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NUCLEAR SCIENCE AND TECHNOLOGY

EUROPEAN COMMUNITY
WATER REACTOR
SAFETY RESEARCH PROJECTS

May 1976

INTRODUCTION

This is the third compilation of Community research formats to be produced by the Commission.

The formats have now been assembled in the order of the revised classification system agreed at the fifth plenary meeting of Working Group No. 2 on Light Water Reactor Safety (Research) of the European Community, held in Brussels in January 1976. Research projects on pressure tube type reactors can now be included.

It has been requested by some contributors of these formats that if the information is distributed outside the European Community then the originator of the format should be informed of the destination.

EUR : 68

Classification system

1. Blowdown and emergency core cooling
   - 1.1 Phenomena prior to ECCS initiation
     - 1.1.1 Dynamic effects of depressurisation (e.g. effects on pressure circuit internals, on fuel, internal stress in fuel)
     - 1.1.2 Thermo-hydraulic aspects
     - 1.1.3 Reactivity effects
     - 1.1.4 Decay heat
   - 1.2 Performance of ECCS
   - 1.3 Behaviour and influence of fuel-elements specifically related to blowdown and ECCS

2. Core meltdown
   - 2.1 Molten material behaviour
   - 2.2 Fuel/coolant interaction
   - 2.3 Effects of molten material on structures

3. External influences
   - 3.1 Seismics effects
   - 3.2 Missiles
   - 3.3 Explosions
   - 3.4 Fire
   - 3.5 Hurricanes and tornadoes

4. Power transients
   - 4.1 Reactivity insertions
   - 4.2 Secondary system effects
   - 4.3 Instability

5. Behaviour, transport and release of radioactive substances
   - 5.1 Release from fuel-elements in normal operation
   - 5.2 Release from overheated fuel-elements (in accident conditions, including LOCA)
   - 5.3 Retention (e.g. plate out, wash-out, filtration)
   - 5.4 Environmental effects
   - 5.5 Detection and measurement
   - 5.6 Doses emanating from released activities

6. Faults and accident combinations

7. Containment and associated systems (for material and mechanical problems : see section 11)
   7.1 Dynamic loading (e.g. pressures, pressure differential, pressure waves, jet forces, internal missiles) and temperature loading
   7.2 Pressure suppression
   7.3 Hydrogen production and limitation
   7.4 Leak tightness assurance

8. Instrumentation, control and computerized protection

9. Other safeguards

10. Core and primary circuit in steady state conditions
    10.1 Physico chemical and materials properties and their effects on fuel elements, core internals, control mechanisms and primary circuit components
    10.2 Reactor physics
    10.3 Thermohydraulics
    10.4 Mechanical effects (e.g. vibration)

11. Materials and mechanical problems in normal and accident conditions (e.g. load following, turbine trip, blowdown, etc.)
    11.1. Fuel elements and core (e.g. fuel densification, fuel pin distortion, cladding ballooning, cladding oxidation, cladding embrittlement, cladding water reaction, rupture)
    11.2 Steel pressure vessel, pressure vessel internals and primary circuit
        11.2.1 Material properties
        11.2.2 Stress-strain analysis
        11.2.3 Non destructive testing, inspection, surveillance
        11.2.4 Destructive testing
    11.3 Prestressed concrete pressure vessel
        idem 11.2
    11.4 Containment
        11.4.1 Concrete structures
            idem 11.2
        11.4.2 Steel structures
    11.5 Coolant channels

12. Quality assurance

    12.1 Formulation of quality assurance system

    12.2 Fabrication methods

    12.3 Non destructive testing, inspection, surveillance (for pressure structures and componants : see under relevant sections 7 and 11)

    12.4 Human factors

13. Systems optimisation, standardisation, new concepts

    (e.g. integrated primary circuit, new containment concepts)

14. Probabilistic methods of safety analysis

15. Interrelation between reactor plant and operating personnel

    15.1 Behaviour of personnel (under normal and accident conditions)

    15.2 Training of personnel

16. Environmental protection

    16.1 Preparation for emergencies

    16.2 Emergency equipment

17. Nuclear accident recovery and decommissioning

    17.1 Decontamination

    17.2 Removal of accident consequences

    17.3 Decommissioning

18. Fuel cycle

    (e.g. fuel production, fuel and waste transport, reprocessing)

19. Economics of safety

1. BLOWDOWN AND ENERGENCY CORE COOLING

Classification 1

Title 1	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 NORHAV - RHC a core heat-up computer program	Project leader: Aksel Olsen
Initiated: November 1971 Completed: Status: progressing Last updating:	Scientists: Jens Andersen H. Abel-Larsen Preben Hansen

1. General aim

Development of a multirod core heat-up computer program, including spray cooling and flooding.

2. Particular objectives

RHC calculates the temperature transient of the fuel and coolant in a multirod cluster geometry evaluating the influence of the emergency core cooling. The program is based on a separate description of the water and steam phase in the primary system and a detailed description of the radiation heat transfer between the fuel rods and the shroud including multiple reflection. The latter involves a determination of the absorption of thermal radiation in the two-phase mixture in the fuel element. Furthermore, decay heat, metal-water reactions, heat transfer due to convection and conduction, creation and propagation of water films on the shroud and the individual fuel rods. The program also takes into account the influence of the primary system.

3. Experimental facilities and programme

4. Project status

1. Progress to date

A version of the program with spray cooling is available for production use.

2. Essential results

5. Next steps

Development of a flooding version of RHC.

6. Relation with other projects

In addition to the present core heat-up program the NORHAV project includes:

a) A one-dimensional blow down computer program for reactor systems under development at IFA, Norway.

b) The Danish subchannel blow down computer program DANBLOW under development at AEC, Risø.

c) Updating of COBRA 3-C and RELAP 3 by STF, Finland and AE, Sweden.

d) A 64-rod (electrically heated) core heat-up experiment by AE, Sweden.

7. Reference documents

Jens Andersen:
REMI/HEAT COOL. A Model for Evaluation of Core Heat-up and Emergency Core Spray Cooling System Performance for Light-Water-Cooled Nuclear Power Reactors.
Risø Report No. 296, September 1973.

8. Degree of availability

Available on exchange basis.

Title 1 (original language)	COUNTRY : FRANCE
Programmes de calcul pour l'étude de l'accident de perte de caloporteur d'un P.W.R.	SPONSOR : C.E.A.
	ORGANIZATION
	C.E.A.
Title 2 (english) Computer codes for the LOCA studies (P.W.R.)	Project leader C.E.A. - DSN/SETS M.GOMOLINSKI Scientists :
Initiated (date) 1972 Status : progressing codes en cours de tests Completed : (date) Last updating (date) janvier 1975	

1. But général

Mise au point de l'ensemble des codes permettant de calculer l'évolution de la température maximale de gaine pendant l'accident de perte du caloporteur des réacteurs P.W.R.

2. Objectifs particuliers

Prévision des essais hors pile et en pile. Ajustement des codes sur ces essais. Utilisation des codes pour les réacteurs de puissance.

3. Installations expérimentales et programmes

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4. Etat du projet

a) phase de décompression

Thermohydraulique du circuit primaire calculée par DANAIDES
Echauffement du coeur calculé par THETA 1 B (qui utilise les résultats de DANAIDES comme conditions aux limites du canal).

b) Phases de remplissage et du renoyage

Thermohydraulique du circuit primaire calculée par CERES
Echauffement du coeur calculée par le code CORINTHE : calcul de la température maximale de gaine dans le plan du point chaud en utilisant les résultats de CERES et les corrélations FLECHT.

5. Prochaines étapes

Remplacement de CORINTHE par FLIRA 2.
Ajustement sur les essais PHEBUS.
Utilisation pour les réacteurs de puissance.

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6. Relation avec d'autres projets

Programmes de calcul pour la phase de décompression DANAIDES et code utilisant la méthode des caractéristiques.

Programmes de calcul pour la phase de renoyage CERES et FLIRA 1 et 2.

7. Documents de référence

Programmes de calcul pour l'étude de l'accident de perte du caloporteur des réacteurs P.W.R.

Rapport SETS n° 31 par M.GOMOLINSKI, D.MENESSIER, N.TELLIER (disponible)

(Version anglaise du rapport présenté au séminaire sur les programmes de calcul de sûreté des réacteurs thermiques - ISPRA 23-25 octobre 1974).

Title 1 (original language) Etude en pile de l'accident de perte de caloporteur (P.W.R.)	COUNTRY : FRANCE SPONSOR : C.E.A. ORGANIZATION C.E.A.
Title 2 (english) In pile study of the loss of coolant accident (P.W.R.)	Project leader CEA/DSN/SETS MmexTELLIER – ~~Collaborateurs~~ : MxxDELxNEGRO M. DEL NEGRO Mme TELLIER
Initiated (date) 1972 (début projet) Status : progressing début de construction	Completed : (date) Last updating (date) janvier 1975

1. But général

Etude en pile de l'accident de perte du caloporteur ainsi que de l'efficacité du refroidissement de secours.

2. Objectifs particuliers

S'ajoutant à des essais fragmentaires et plus fondamentaux portant sur la décompression, le renoyage et le débit critique ces essais doivent permettre d'ajuster les programmes de calcul au déroulement global de l'accident y compris l'intervention de refroidissement de secours. D'une façon plus spécifique ces essais permettront d'étudier le comportement thermo-mécanique des éléments combustibles ainsi que l'influence de l'irradiation.

3. Installations expérimentales

Réacteur PHEBUS (Centre d'Etudes Nucléaires de Cadarache).
Boucle d'essais placée au sein d'un réacteur type piscine.
Principales caractéristiques de la boucle :

nombre de barreaux	:	29 (géométrie 17x17)
hauteur fissile	:	0,80m
puissance linéique maximale	:	500 W/cm (environ)
pression	:	160 bars

4. Etat du projet

Construction commencée en aout 1974

5. Prochaines étapes

Début des essais vers la fin de 1976
Les essais devront comporter des expériences sur 1 crayon puis sur une grappe de 29 barreaux.

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6. Relation avec d'autres projets

Etude des conditions de refroidissement pendant la phase de décompression (OMEGA)
Etude expérimentale de refroidissement de secours (ERSEC)
Mise au point de l'instrumentation pour les études de décompression
Programmes de calcul pour l'étude de la dépressurisation
Programmes de calcul pour l'étude du renoyage.
Programmes de calcul pour l'étude de l'accident de perte de caloporteur-eau d'un P.W.R.

7. Documents de référence

Etude du comportement des éléments combustibles des réacteurs à eau en cas d'accident de dépressurisation. Programme PHEBUS

par J.FONTERAY - L.ROCHE - H.VIDAL

Présenté au CSNI Specialist Meeting - 22-24 octobre 1973 - Saclay/France

	CLASSIFICATION 1
TITLE 1 CLYSTERE. CODE DE CALCUL DES CONSEQUENCES D'UNE RUPTURE DU CIRCUIT PRIMAIRE.	COUNTRY FRANCE SPONSOR E.D.F./SEPTEN ORGANIZATION E.D.F.
TITLE 2 CLYSTERE. CALCULATION CODE OF THE CONSEQUENCES OF A LOCA.	Project Leader E.D.F./SEPTEN/T
Initiated 1973 Completed 1975 Status Last updating : 20.01.75	Scientists M. SUREAU M. HOUDAYER E.D.F.

I - GENERAL AIM

Code de calcul du traitement de l'accident de perte du caloporteur (Rupture du circuit primaire) d'un PWR .

II - PARTICULAR OBJECTIVES

Ce code décrit toutes les phases de l'accident. Sont pris en compte, les matériels du circuit primaire et les phénomènes qui interagissent au cours de l'accident. On a jugé nécessaire de passer :

- du modèle monocanal au multicanal,
- de la modélisation ponctuelle à la modélisation axiale (pour, en particulier, suivre les propagations d'ondes de pression),
- de l'équilibre au déséquilibre entre les deux phases du fluide.

La modélisation s'effectue sur la base d'un modèle à 4 équations, les 3 premières étant les équations classiques de conservation de la masse, de quantité de mouvement et d'énergie. La quatrième caractérise le retard au changement d'état.

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III - EXPERIMENTAL FACILITIES AND PROGRAMME

Toutes études expérimentales de support du programme français.

IV - PROJECT STATUS

4.1 - Progress to date

L'écriture du programme est terminée. L'ensemble tournera de façon satisfaisante et pourra être utilisé pour traiter des problèmes physiques au milieu de 1975.

4.2 - Essential Results

V - NEXT STEPS

- tests globaux à partir du milieu de 1975,
- calage du code, sur des expérimentations particulières.

VI - RELATION WITH OTHER PROJECTS

Toutes études expérimentales et théoriques menées en France relativement aux conséquences de l'accident de perte du caloporteur.

VII - REFERENCE DOCUMENTS

Néant.

VIII - DEGREE OF AVAILABILITY

A partir de 1976, des études appliquées pourront faire l'objet de contrats cas par cas.

CLASSIFICATION 1	
Title 1	Country FRANCE Sponsor Organisation G.A.A.A.
Title 2 KAPCOR : A blowdown code	Project Leader : JC. MEGNIN
Initiated : July 1971 Completed : January 1973 Status : Last updating	

1. GENERAL AIM

Development of a multinode blowdown code simulting specially the core of a water cooled reactor.

This code was initially developped for pressure tubes reactors.

2. PARTICULAR OBJECTIVES

The geometric description of the circuit includes a lower and upper plenum connected by several channels. Each channel can simulate a core assembly, the by pass and if desired a downcomer line including an heat exchanger and a pump.

The core itself, can be simulated with hot, average and cold channels. The boundary conditions, flows and enthalpies entering or leaving the lower and upper plenum are input data versus time.

Heat transfer regimes and heat transfer coefficients between the cladding and the coolant are determined fonction of the thermodynamic conditions of the coolant (nucleate boiling, film boiling).

Heat diffusion within the fuel and the clad is calculated using the finite difference equations approach.

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Special attention has been paid to obtain a code economical to run, easy to modify or complete and using improved numerical methods, integration procedures, automatic choice of time steps.

3. EXPERIMENTAL FACILITIES

Comparison between calculations and experimental data were done at the technology division of the CCR Ispra with the DHTI loop.

4. PROJECT STATUS : Complete

5. RELATIONS WITH OTHER PROJECTS

DHTI loop in Ispra.

6. REFERENCE DOCUMENTS

- G. FRIZ, W. RIEBOLD, JC. MEGNIN, A. RAYNAUD
  A comparison between code calculation and blowdown experiments simulating a loss of coolant accident in a pressurized water reactor, Nuclear Engineering and Design 25 (1973) 193-206

- G. FRIZ, W. RIEBOLD, D. LANGE, JC. MEGNIN
  Calculations compared with experiments simulating different blowdowns,
  European Nuclear Conference, April 21 - 25, 1975 - Paris

- G.A.A.A., code, descriptions and user's manual (internal reports).

7. DEGREE OF AVAILABILITY

To be discussed.

Classification 1, 7.1, 7.2	
Title Calculations of the consequences of pipe breaks in reactor systems	Country The Netherlands Organization KEMA
Status progressing Last updating 1975	Projectleader R.M. van Kuijk Scientists Kloeg Oppentocht Talens

1. General aim

To evaluate the general design and the design of components of reactor systems in the case of pipe breaks in the system.

2. Particular objectives

A - Critical flow rates
- Pressure decrease in the primary system
- Steam water separation
- Heat transfer in the core
- Behaviour of the cladding (ballooning temperatures)
- Initiation and behaviour of core cooling systems

B Forces on internals in the vessel during blow down

C - Pressure and temperature response in containment systems (including pressure suppression)
- Long term behaviour of the containment system.

3. Experimental facilities and programme

- Participation in the Marviken project
- Main computer programmes:

slow down : BRVIS, RELAP
core heat up: CHEMLOC-5, BUBBLE
containment : RIS, DRUKSTUK, ZOCCO
forces : MARC

4. Project status

- Progress to date: operational
- Essential results: complete ECCS analysis of the Dodewaard reactor (BWR).

5. Next steps

- Calculation of Marviken I and II results
- Small break analyses in PWR.

6. Relation with other projects

Not applicable.

7. Reference documents

Internal KEMA reports.

8. Degree of availability

Free on basis of exchange with other programmes and results.

Classification 1.1

Title 1	COUNTRY Denmark
	SPONSOR DAEC Risø
	ORGANIZATION DAEC Risø
Title 2 DINO - Core heat-up during blow down	Project leader: H. Abel-Larsen
Initiated: February 1971 Completed: September 1972 Status: Last updating:	Scientists: H. Abel-Larsen M. Lolk Larsen

1. General aim

Development of a computer programme for the calculation of transient temperatures in a fuel rod during a postulated loss-of-coolant accident.

2. Particular objectives

DINO calculates the transient temperatures in a fuel element rod during a postulated loss-of-coolant accident. The geometric model is cylindrical. The considered rod is concentric surrounded with an equivalent coolant channel and a shroud of fuel and canning consisting of and equivalent to the surrounding rods and possible fuel element box. The equivalent geometry is calculated from the assumption of the same hydraulic diameter.

DINO is a finite difference program, two-dimensional in the fuel system and one-dimensional in the coolant channel. The program contains a steady state option to calculate the initial temperatures. The integration technique used is Peaceman and Rachford's method, the ADI-method. Gas-gap between fuel and canning, different materials, radiation etc. may be taken into account using a calculated equivalent heat conductivity. Temperature dependence of the physical properties is taken into account.

3. Experimental facilities and programme

4. Project status

Completed

5. Next steps

6. Relation with other projects

The DINO program is part of an integrated procedure for calculation of fuel temperature transients during loss-of-coolant accidents. Besides the DINO program, the procedure consists of RHC and a blow down program which calculates the hydraulic conditions during the accident. At present the German program BRUCH-S is used for the blow down calculations.

* 7. Reference documents

H. Abel-Larsen and M. Lolk Larsen
Heating in a reactor fuel element rod under transient conditions. Part I.
Heat conduction program. RISØ-M-1391 (1971)

H. Abel-Larsen and M. Lolk Larsen
Heàting in a reactor fuel elément rod under transient conditions. Part II.
Risø-M-1533 (1972)

8. Degree of availability

Available

Classification 1.1

Title 1	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 NORHAV - P(B)WR core blow down computer program (DANBLOW)	Project leader: Aksel Olsen
Initiated: 1973 Completed: Status: progressing Last updating:	Scientists: Peter Sten Andersen Niels Bech Marilyn Eget

1. General Aim

Development of a 3-dimensional P(B)WR core blow down computer program.

2. Particular objectives

Calculation of the spatial and temporal distributions of coolant mass, -flow, -enthalpy and pressure as well as fuel rod temperature, cladding deformation and rupture in a P(B)WR core during the blow down phase of a loss-of-coolant accident. The model which is based on the subchannel approach includes slip and thermal non-equilibrium between steam and water. Opposite flow directions within a single channel (as well as different channels) are handled by the program.

3. Experimental facilities and programme

4. Project status

1. Progress to date

The physical equations have been formulated and partly tested. Some basic numerical problems (stability and opposite flow directions) have been solved. At present the hydraulic part of the program is being programmed. A fuel- and cladding failure

model is under development

2. Essential results

5. Next steps

6. Relation with other projects

In addition to the present blow down program the NORHAV project includes:

a) A one-dimensional reactor system blow down model under development at IFA, Norway

b) The Danish core heat-up programme RHC under development at AEC, Risø

c) Updating of COBRA 3-C and RELAP 3 by STF, Finland and AE, Sweden

d) A 64-rod (electrically heated) core heat-up experiment carried out by AE, Sweden

7. Reference documents

8. Degree of availability

Available on exchange basis when completed

Title 1 (original language)	COUNTRY : FRANCE
Etude expérimentale des débits critiques	SPONSOR CEA/EDF
	ORGANIZATION C.E.A
Title 2 (english) Expérimental Study of critical flows	Project leader CEN/DTCE/STT M.COURTAUD Scientists :
Initiated (date) 1968 — Completed : (date)	
Status : progressing — Last updating (date) Janvier 1975	

1. **But général**

Détermination des débits critiques et étude des écoulements à fort gradient de pression.

2. **Objectifs particuliers**

Mise en évidence du déséquilibre pouvant exister entre les phases d'un écoulement eau-vapeur, déséquilibre qui a une influence prépondérante sur la valeur du débit critique.

3. **Installations expérimentales**

Boucle Moby Dick (Centre d'Etudes Nucleaires de Grenoble)

- 1ère phase : tube de 2 cm de diamètre, pression maximale de 2 bars et titre maximal de 0,8%.
- 2ème phase : tube de 14 mm de diamètre permettant avec la même installation que précédemment d'attendre des pressions de 10 bars.

4. **Etat du projet**

Essais de la 1ère phase réalisés et interprétés.
Démarrage des essais de la 2ème phase.

5. **Prochaines étapes**

Essais sur l'installation permettant d'atteindre 10 bars .... début 1975
Débits critiques avec des mélanges eau-air .... fev.mars 75
Extension de la gamme de titre (jusqu'à 20%) .... juin 75-Juin76
Influence de la géométrie élargissement brusque .... juin 75
effet de coude .... juin 76
A l'étude : boucle Super Moby Dick de 3 MW permettant d'explorer les régions à forte pression (110 bars) et faibles titres et moyennes pression (20 bars) et fort titres.

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6. Relation avec d'autres projets

Etude des conditions de refroidissement pendant la phase de décompression (OMEGA).
Mise au point de l'instrumentation par les études de décompression.
Programmes de calcul pour l'étude de la décompression.

7. Documents de références

- Thèse de M. REOCREUX : Contribution à l'étude des débits critiques en écoulement diphasique eau-vapeur - Grenoble 1974.

- Choking flows and propagation of disturbances
  European two-phase flow group meeting - Brussels June 4-7 (1973)
  par J. BOURÉ - A.FRITTE - M. GLOT - M.REOCREUX

- Etude expérimentale des débits critiques en écoulement diphasique eau-vapeur à faible titre dans un canal avec divergent de 7 degrès.

  par : M.REOCREUX - G.BARRIERE - B.VERNAY

  CEN/G rapport interne rTT/15 (1973)

Title 1 (original language) Programme de calcul pour l'étude de la dépressurisation	COUNTRY : FRANCE SPONSOR : C.E.A. ORGANIZATION C.E.A.
Title 2 (english) Blow-down computer codes	Project leader C.E.A./DSN/SETS D.MENESSIER Scientists :
Initiated (date) 1972 — Completed : (date) Status : progressing — Last updating (date) Codes à l'état de tests et d'amélioration — Janvier 1975	

1. **But général**

Mise au point de programmes de calcul pour étudier la phase de décompression lors de l'accident de perte de caloporteur d'un P.W.R.

2. **Objectifs particuliers**

Mise au point du code DANAIDES qui étudie la thermohydraulique du circuit primaire pendant la phase de décompression.

Mise au point d'un code traitant un canal par la méthode des caractéristiques. Ajustement de ces codes sur des essais hors-pile et en pile. Application au calcul de l'A.D.R. des réacteurs de puissance.

3. **Installations expérimentales et programmes**

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4. **Etat du projet**

a) code DANAIDES

Le code existe sous plusieurs versions :

DANAIDES H : traite une boucle d'essais sans calculs de thermique (le flux de chaleur délivré au fluide est une donnée).

DANAIDES T : traite une boucle d'essais avec calculs de thermique

DANAIDES R : version réacteur de puissance.

Une forme simplifiée du code (DABBOUS) traite la décompression d'un réservoir.

b) code utilisant la méthode des caractéristiques

Première version en cours de tests.

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5. Prochaines étapes

Amélioration des modèles physiques des diverses versions de DANAIDES 1975

Amélioration de la méthode numérique afin de diminuer les temps de calcul 1975

Poursuite des tests du code des caractéristiques 1975

Dépouillement des essais OMEGA 1976

Dépouillement des essais PHEBUS 1978

6. Relation avec d'autres projets

Essais sur la boucle OMEGA
Décompression du dispositif CANON
Essais sur la boucle MOBY-DICK
Essais sur la boucle PHEBUS

7. Documents de référence

Programmes de calcul pour l'étude de l'accident de perte du caloporteur des réacteurs P.W.R.

Rapport SETS n° 31 par M. GOMOLINSKI, D.MENESSIER, N.TELLIER (disponible)

(Version anglaise du rapport présentée au séminaire sur les programmes de calcul de sûreté des réacteurs thermiques - ISPRA 23-25 octobre 1974).

Title 1 (original language)	COUNTRY: FRANCE
Mise au point de l'instrumentation pour les études de décompression.	SPONSOR: C.E.A.
	ORGANIZATION: C.E.A.
Title 2 (english) Instrumentation development for blow-down studies	Project leader C.E.A.-DTCE/STT M.RICQUE Scientists :
Initiated (date) 1974 Status : progressing Completed : (date) Last updating (date) Janvier 1975	

1. **But général**

Mise au point et étalonnage des moyens de mesure destinés aux études de décompression.

2. **Objectifs particuliers**

Mise au point et étalonnage sur le dispositif DEDIF de débimètres diphasiques (moulinet, Venturi) et mesure de taux de vide.

Amélioration des mesures de pression en transitoire sur le dispositif CANON.

Mise au point sur le même dispositif d'une mesure de taux de vide par neutronographie.

3. **Installations expérimentales**

CANON : réservoir contenant de l'eau à 30 bars et 220°C (soit légèrement au dessus de la saturation).
Géométrie : l = 3m Ø 10 cm.

DEDIF : boucle eau-argon sous pression (92 bars) simulant le mélange eau-vapeur à 155 bars.

4. **Etat du projet**

CANON : la première campagne d'essais est terminée. Les mesures de pression par capteur piézoélectrique decroissent trop lentement au cours de la dépressurisation. Il a été montré que le transitoire de température sur la face sensible des compteurs pouvait fausser la mesure la technique de mesure de pression avec des capteurs à membrane effleurante devra être abandonnée.

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DEDIF : en cours d'essais pour les mesures de taux de vide par absorption de rayons X et les mesures par débitmètre à turbine à paliers et butées fluides.

5. Prochaines étapes

Extension de CANON (P = 155 bars environ)
Poursuite des essais sur DEDIF

6. Relations avec d'autres projets

Essais sur la boucle OMEGA
Essais sur la boucle PHEBUS.

Classification 1.1

Title 1 Investigation of the transient flow response in a BWR core.	Country Italy Sponsor CNEN
Title 2	Organisation CNEN-CISE
Initiated 1.1.74 Completed Dec. 75 Status progressing Last updating Sept.74	Project leaders V. Marinelli

1. General aim
Experimental validation of a computer code for the calculation of the onset of the CHF under LOCA conditions.

2. Particular objectives
Validation of the thermal-hydraulic response of the code by means of mass hold-up measurements during transients simulating LOCA.

3. Experimental facilities and programme
Measurements of mass hold-up during transients of inlet flow stoppage at constant power and pressure and inlet flow stoppage and power decay at constant pressure will be taken on a 16 rod BWR test section 12 ft long; the Plant used is IETI-III (CISE).

4. Project status
   1. The computer code, named DOLCE has been developed and satisfactory results have been so far obtained for the predictions of transient CHF in a BWR during LOCA and other transients.

      The mass hold-up measurements have been initiated.
   2. Both Eulerian and Lagrangian methods used in the DOLCE code give reasonably good results in the evaluation of transient CHF.

7. Reference documents
DOLCE computer code, CNEN Internal Report. Under pubblication
Open

8. Budget
60. MLit - Personnel 50 MM

	Classification 1.1
Title 1 Investigation of flow blockage effects in a sub-channel array.	**Country** Italy **Sponsor** CNEN **Organisation** CNEN-A.B.Atom-energi (Sweden)
Title 2	
Initiated June 73 **Completed** Dec. 74 **Status** progressing **Last updating** End 73	**Project leaders** V. Marinelli

1. General aim

   Study the flow redistribution of the coolant in blocked subchannels.

2. Particular objectives

   Validate the theoretical methods as LEUCIPPO, COBRA, in their ability to predict the flow distribution in blocked subchannels by extensive comparison with experimental data.

3. Experimental facilities and programme

   Single phase flow is completed in a test section of 4x4 rods 2.5 m long where the blockages are installed. Measurements of the three components of the velocity are taken, by means of a special 5 Pitot-tubes probe, as well as radial and axial pressure drops. The blockages consist in 100% and 70% obstructions within the center subchannel, in a large blockage of one half test section, and also in ballooning type blockage of four rods.

4. Projects status

   1. Progress to date

      The measurements have been completed at 90%

   2. Essential results

      Preliminary indications show the existence of long relaxation (50 D) length before the normal flow redistribution is obtained downstream a blockage.

5. Next steps

   Analitycal work will be done to reduce the data and to compare them with the subchannel code predictions.

6. Relation with other projects

   None

7. Reference documents

   None

8. Degree of availability

   Budget

   40 M/lit / Personnel: 36 MM

Classification
1.1.
(1.2.)

Title 1 Experimentelle Untersuchungen des Einflusses der DWR-Umwälzschleifen auf den Blowdown	Country : JRC Sponsors: BMFT-Bonn, CEC Organization : JRC ISPRA Establishment
Title 2 Experimental Investigation of the Influence of PWR-Loops on Blowdown	Project leader: W. Riebold
Initiated : December 1973 Status : progressing Completed : December 1977 (BMFT part A) Last updating : March 1975	

1) General aim

Design and construction of a rather large blowdown loop system. Performance of loss of coolant experiments by simulating tube ruptures within a model PWR primary cooling circuit system.

2) Particular objectives

Experimental investigation of the role of the different components of a model PWR primary cooling circuit system during a blowdown by the measurement of the main thermohydraulic quantities at all important positions in the loop. The experimental results will be used for the checking and development of blowdown codes and associated theories used in LWR safety assessment.

3) Experimental facilities and programme

A 4-loop primary cooling circuit of a 1300 MW(e) PWR reference plant will be simulated by a 2-loop experimental system, one loop representing the three "intact" reactor loops and the

other representing the "broken" reactor loop. Both experimental loops contain pumps and steam generators. Tube ruptures (double-ended and smaller) will be simulated at three different positions in the "broken" loop.
Applying a scaling factor with respect to the reference plant of about 1/700 for the thermal power, mass flow rate and volume led to a 5 MW power input to a 64 heater rod bundle simulation of the reactor core.
The distribution around the loop of pressure drop, fluid temperature and component volumes will be carefully matched to the same distributions in the real PWR circuit.
The relative heights of the components and the lengths of the heat transfer regions (core rod bundle, steam generator tubes) will also be the same as in the real circuit. Size reductions will be made under the constraint that the power to volume ratio is maintained equal to that of the real system.

Two different experimental programmes are envisaged :

Programme A specified by the BMFT-Bonn, is concerned with the investigation of the influence on the blowdown of the rupture size at three different positions, the pumps characteristics in both loops, the initial power level, the time dependence of the heat input, the strength of the heat sink (steam generator secondary side), the downcomer resistance and the ECC water injection positions.

Programme B formulated by the CEC, is mainly concerned with studies of variations of geometry and components. These studies foresee the modification of certain components and certain aspects of loop geometry (shape and component height). This programme will take certain reference tests from programme A (in fact they will be repeated) so that the consequences of the loop variations can be assessed in a clear manner. Seven loop variations have been agreed on for programme B :

- variation of the depth of the loop seal (U-tube between steam generator and pump) within the intact loop;

- **variation of the steam generator height in the intact loop;**
- variation of the volume of the lower plenum (higher l/d ratio);
- two separate accumulators, one for each loop, instead of one accumulator for both loops;
- primary tube rupture within the steam generator (of the intact loop);
- small rupture within the lower plenum;
- ECC water injection into the upper plenum.

4) Project status

The project work started in January 1974 with the revision of the preliminary loop design and will be completed at the beginning of 1975.
The work involved in this revision became more extensive because of two major modifications in the concept of the loop. The first modification resulted from the change of the reference plant from a 600 MW(e) to a 1300 MW(e) PWR which necessitated the provision of one steam generator for each loop instead of one common steam generator for both loops and a more appropriate design for the double-ended rupture device. The second modification was concerned with the design of the downcomer as an annulus instead of the previously conceived circular tube.
The final loop design was concluded in November 1974 and the corresponding orders for construction are being placed in 1975. Three pumps of the same type and performance have been ordered. Two of them will be directly delivered to Ispra and will be operated at different speeds in the two loops so as to account for the different loop mass flow rates. The third will be used for establishing the two-phase pump characteristics in the framework of a separate R&D-contract of the BMFT-Bonn; thereafter it will be available as a spare pump.

The specifications of the electrical power supply and the loop regulating and control systems as well as those of the data acquisition systems have been completed and the orders for all these systems are being placed.

Extensive theoretical work has been necessary in the loop design and for the specifications of the different auxiliary systems. This has been done partially by the project group itself and partially by the LRA-Garching using the BRUCH-D blowdown code.

5) Next steps : Orders for all parts of the loop are being placed and preparations for the mounting of the loop are being made during 1975. Prototypes of signal transducers and measuring chains will be tested and calibration facilities will be prepared.
The final specification of the instrumentation system will be made in the near future, taking into consideration the philosophy adopted for the Semiscale and LOFT instrumentation.
Programmes for the digital minicomputer will be set up for the different process control, the data acquisition and evaluation tasks.
Pre-prediction blowdown calculations will be carried out with the BRUCH-D and the RELAP-4 codes for the programme A version of the test rig, and with RELAP-3 and RELAP-4 codes for the programme B version (with several component modifications).

6) Relation with other projects :

There is a close relation with the following BMFT contracts(RS):

RS- 16/2 : Investigation of decompression phenomena of LWR's. Model tests with a steel vessel with core internals.

RS- 36 : Emergency cooling programme - low pressure tests of core refilling of LWR's after MCA :

36/1 : Evaluation of flooding tests with single tube and rod bundle

36/2 : Refill tests with primary loop influences

RS- 37 : Emergency cooling programme - high pressure tests :

37/1 : Investigation of phenomena within the core during loss of cooling and emergency cooling

37/2 : Determination of heat transfer coefficients

RS- 48 : Theoretical and experimental investigations on scaling laws for transient heat transfer conditions in LWR-ECC

RS- 50 : Investigation of phenomena in a widely subdivided containment in cooling tube rupture accidents of LWR's.

RS- 62 : Tube experiments for setting up a theory for rewetting of fuel rods heated up to high temperatures

RS- 77 : Investigation of the thermohydraulic non-equilibrium

RS- 81 : Mixing phenomena in parallel flow channels

RS-111 : Investigation of reactor pump behaviour during blowdown

RS-144 : Investigation of RS-109 experimental pump behaviour during blowdown

RS- 64 : Investigations of steady-state and transient CHFs with multiple rod bundles of PWRs and BWRs with R 12 as model fluid

RS : Development of measuring techniques for density and mass flow rate in water-vapour two-phase flow

7) <u>Reference documents</u> :

1. Tender to the BMFT-Bonn for the execution of the project "Experimental Investigation of the Influence of the PWR-Loops on Blowdown" in the EURATOM JRC at Ispra, elaborated by the Technology Division of the JRC, May 1973

2. I. Trimestrial Report 1974, IRS - F - 20 (July 1974)

3. II.Trimestrial Report 1974, IRS - F - 21

4. III. Trimestrial Report 1974, IRS – F – 22 (December 1974)
5. IV. Trimestrial Report 1974, IRS – F

JRC Safety Programme Progress Report 1974

8) Degree of availability :

The references mentioned above are available form the IRS-Köln, Glockengasse 2.

9) Budget :

Provisional estimates of the total costs (manpower and investments) considered in the contract BMFT/CEC are as follows :

BMFT : about 4 MUA
CEC : about 4 MUA

10) Personnel :

BMFT : 10 men/year
CEC : 15 men/year

11) Additional information :

The time schedule of whole the project according to the planning made during the elaboration of the tender for the BMFT-Bonn is as follows :

Project phase I : Elaboration of the preliminary project and of the tender for the BMFT-Bonn for the execution of this project at the Ispra Establishment of the JRC :
Nov. 1972 – April 1973

Project phase II : Revision of the preliminary project, request for confirmation of existing offers and for new offers, placing of orders :
January 1974 – September 1974

Project phase III : Construction and mounting of the loop; preparation of computer programmes for process control, data acquisition and evaluation; prototype testing and preparation of calibration facilities for instrumentation; pre-prediction calculations with different blowdown computer codes : October 1974 - December 1975

Project phase IV-1: Commissioning of the loop with all auxiliary systems; performance of preliminary tests : January 1976 - December 1976

Project phase IV-2: Execution of tests for the experimental programme A : January 1977 - December 1977

Project phase V : Execution of tests for the experimental programme B : January 1978 - December 1978

Time slippages accumulated : Because of extensive project revision (see § 4) and new delivery times, especially of those parts determining the critical path of the planning, the beginning of the project phase IV-1 will certainly be delayed by 10 months.

Classification: 1.1.1		
Title 1 (Original Language): Untersuchung der Vorgänge bei Druckentlastung wassergekühlter Reaktoren. Modellversuche mit einem 11,2 m hohen Stahlbehälter (RS 0016 B - I.1.1., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Battelle-Inst. Frankfurt/Main
Title 2 (english): Investigation into the Phenomena Involved in the Depressurization of Water Cooled Reactors. Experiments Using a Steel Vessel 11.2 m in Height with Internals		Project Leader: B. Hölzer
Initiated (Date): 15th July, 1972	Completed (Date): 30th April, 1979	
Status: continuing	Last Updating (Date): December 1975	

1. General Aim

The aim of this project is to investigate by means of large-scale experiments the loads on reactor vessel internals of pressurized and boiling-water reactors occurring in the case of rupture of the primary cooling circuit.

All experimental results will be compared with the results of model-calculations to show the applicability of the computer codes used and, if possible, to improve them.

2. Particular Objectives

The loads on reactor vessel internals under BWR and PWR conditions and the phenomena in the discharge nozzle during the initial phase of blowdown is to be investigated. By means of comparing the results of the preliminary experiments without internals with the data of the main tests the influence of the internals on the discharge process shall be obtained.

3.1. Experimental Facilities

The experimental facilities are described in the previous annual report IRS-F-24. In the planned BWR experiments only the electrical power of the heater will change from nominally 350 kW to 600 kW.

.2. Research Program

Simulation of the maximum credible accident under BWR and PWR conditions without and with internals.

- The preliminary program includes four experiments without internals,
- The PWR program is concerned with the investigation of the influence on the blowdown of

-- cross-section and site of rupture
-- time of rupture
-- break of inlet or outlet nozzle ("cold" or "hot" leg)
-- type of internals ("rigid" or "flexible")
-- L/D ratio of the discharge nozzle
-- initial thermal stratification.

- The BWR program is concerned with the investigation of the influence on the blowdown of

-- type of rupture (steam line, feed-water line)
-- cross-section of rupture
-- time of rupture
-- initial water level.

4. Project Status

4.1. Progress to Date

The first five of 18 PWR experiments have been performed. The variation in the parameters and the initial conditions are shown in the following table:

	PWR1	PWR2	PWR3	PWR4	PWR5
Cross-section of rupture (dia/mm)	50	100	145	145	145
Rupture time (ms)	~3	~3	~110	~30	~3
Length of discharge nozzle (mm)	550	550	920	920	350
Initial temperature (°C)	←		285		→
Initial pressure (bar)	141	141	141	138	142

All experimental data are recorded on computer tapes.

4.2. Essential Results

The experimental results compared well with the results of the calculations, but the magnitude of the measured stresses on the internals was generally lower than calculated.

The magnitudes of the measured stresses reflect the different cross-sections of rupture (results of PWR1, PWR2, PWR5) and show the influence of rupture time (PWR3, PWR4, PWR5).
In particular the experimental results of the mass flow-rate showed a significant influence of the internals on the discharge process. The maximum peak of the mass flow-rate was higher in the PWR experiments with internals but occured later than in experiments without internals.

5. Next Steps

Evaluation of the results of the performed experiments will be continued and a report will be prepared.
When the instrumentation of the "BWR internals" and the final preparation for the BWR experiments are finished, eight BWR experiments with internals will be carried out. Three of them will simulate a steam-line break, five a break of a feed-water line.

6. Relation to other projects

RS 0123A Safety Investigations performed at the decommissioned HDR plant.

7. Reference Documents

(1-4) Quarterly reports in the series "IRS-Forschungsberichte" (in German)

IRS-F-25	January to March 1975
IRS-F-26	April to June 1975
IRS-F-27	July to September 1975
IRS-F-28	October to December 1975

(5) IRS-F-24 Research Reports (Annual Reports) (in English)

(6) Technischer Bericht RS16/2
Versuchsergebnisse vom Druckentlastungsvorgang im Druckbehälter mit flexiblen DWR-Einbauten
(Versuche KWR01, KWR02, DWR1, DWR2)
April/Mai 1975

(7) Technischer Bericht RS16/2
Versuchsergebnisse vom Druckentlastungsvorgang im Druckbehälter mit flexiblen DWR-Einbauten.
Versuch DWR3
Juli 1975

(8) Technischer Bericht RS16/2
Versuchsergebnisse vom Druckentlastungsvorgang im Druckbehälter mit flexiblen DWR-Einbauten.
Versuch DWR4
August 1975

(9) Technischer Bericht RS16/B
Versuchsergebnisse vom Druckentlastungvorgang im Druckbehälter mit flexiblen DWR-Einbauten
Versuch DWR5
August 1975

(10) Technischer Bericht RS16/B
Versuchsergebnisse vom Druckentlastungsvorgang im Druckbehälter mit flexiblen DWR-Einbauten.
Versuchswiederholung DWR2
September 1975

8. Degree of Availability

Documents are available through the

Institut für Reaktorsicherheit
Forschungsbetreuung
Glockengasse 2, 5000 Köln 1

Documents (6) to (10) are available under special agreement.

PROJECT TITLE : Experimental blow-down tests by PIPER facility	CLASSIFICATION 1.1.1
SPONSORING COUNTRY : ITALY	ORGANISATION : UNIVERSITY OF PISA
DATE INITIATED : 1972 DATE COMPLETED : 1976	PROJECT LEADER : P. VIGNI

Description :

Research program:

Goals are: to acquire specific knowledges on basic blow-down problems; to analyse causes of possible disagreement between experimental results and RELAP-3 calculations, with special reference to model data transfer to full scale plants. Up to date a set of 18 blow-down experiments were carried out. Test pressure was varied from 20 up to 70 Kg/cm^2; blow-down was operated from either water or steam zone through openings of 50 and 15 mm diameter.

In the next future structures will be located into the vessel to simulate geometry of BWR internals; blow-down tests which will be scheduled will have the purpose of studying pressure transients and fluid forces on these structures.

Facilities:

PIPER apparatus which is a pressure vessel, equipped with electrical heating device, rupture disks and instrumentation for pressure and temperature transient measurements. Five measurement points are allowable along the height of the vessel.

Design features of the vessel are:

pressure 110 Kg/cm^2

temperature 310 °C

internal height: two values can be used: 1,8 m and 3 m

internal diameter: 0,194 m

outlet nozzles: two, both of which have diameter of 50 mm and lenght of 400 mm

Reference documents:

1. P. VIGNI et alii
Esperienze preliminari sull'efflusso rapido di miscela acqua-vapore, inizialmente allo stato saturo (P.I.P.E.R.).
Istituto di Impianti Nucleari, RL 149(73).

2. N. CERULLO et alii
Analisi dell'incidente di perdita di refrigerante nel circuito primario di un reattore nucleare. Ricerca teorica e sperimentale sul transitorio di efflusso rapido di miscele acqua-vapore inizialmente allo stato saturo.
Paper presented at the 29th Congresso ATI - Firenze 25-27 Sept. 1974.

3. N. CERULLO et alii
Blow-down Activity Performed at the Scalbatraio Center of the Pisa University, Comparison between Experimental Results and RELAP-3 Calculations.
Meeting on Computer Programs for the analysis of certain problems in thermal reactors safety. NEA CPL - ISPRA - 23-25 October 1974.

Related projects:

7.2

Remark:

Up to date financial support was given by C.N.R. and C.N.E.N.

Classification
1.1.1. (1.1.2.)

Title 1 Untersuchung des thermodynamischen Ungleichgewichts	Country : JRC Sponsor : BMFT and CEC Organization : JRC ISPRA Establishment
Title 2 Investigation of the thermodynamic non-equilibrium	Project leader : G. Friz
Initiated 1.12.1972 Completed : 31.12.1975 Status : progressing Last updating : March 1975	

1) General aim

To provide experimental data for theoretical models describing the deviation from thermodynamic equilibrium of the water-vapour mixture in a primary PWR-circuit during a blowdown.

2) Particular objectives

Measurement of the deviation from thermodynamic equilibrium between the phases caused by :

- a sudden expansion of water
- a periodic volume variation
- injection of cold water in a vappur atmosphere.

The deviation is obtained by observing the time behaviour of pressure.

3) Experimental facilities and programme

The experimental programme consists of :

- 44 tests with a sudden expansion. Parameters are : temperature, initial pressure step and initial void and water quality,
- 25 tests with periodic volume variation, new parameter :frequency

- **36 tests with cold water injection, parameters : injection quantity, state of the vapour atmosphere (pressure and temperature).**

4) Project status

A series of about 20 flashing tests at 200, 250, 280, 300, 315°C has been carried out. The main results are :

- The measured half-value times t_h of return to equilibrium after a stepwise volume increase lie between 20 and 80 ms.
- The pressure time curves fit well with the theoretical calculations. The experiments indicate bubble numbers from N=10 to N= 1000 bubbles per cm^3.
- The dependence of t_h on the initial pressure step and the temperature follows quite well the theoretical curves. The theory describes quite well the return to the equilibrium pressure.

5) Next steps : Completation of the flashing test series. Preparation of the injection tests.

6) Relation with other projects :

RS 36	: "Experiments on Refilling and Emergency Cooling of the Reactor Core of light Water Cooled Power Reactors after an MCA" (SIEMENS-KWU)
RS 37 and RS 37/1	: "Investigations of the Events within the Reactor Core under Loss of Coolant and Emergency Cooling Conditions, High Pressure Experiments" (AEG-KWU)
At T 85 a	: "Emergency cooling-theoretical studies in connection with a pressure fall in the primary system (blowdown)" (LRA-Garching)
RS 109	: "Experimental Investigation of the Influence of PWR-Loops on Blowdown"

7) Reference documents

G.Friz, W. Riebold
Pressure history during flashing caused by a sudden expansion
EUR 5039.e.

Quarterly reports (German) and annual report (English) in the series IRS-Forschungsberichte IRS F 15 to IRS F 22.

JRC Safety Programme Progress Report 1974.

8) Degree of availability : Freely available

9) Budget : Total investment and running costs are :
BMFT : 13660 UA
CEC : 21000 UA

10) Personnel : 2.5 men/year

11) Additional information :

Classification 1.1.2

Title 1 (original language)	COUNTRY : FRANCE
Etude des conditions de refroidissement pendant la phase de décompression d'un accident de perte du caloporteur d'un réacteur P.W.R. Boucle OMEGA.	SPONSOR : C.E.A. ORGANIZATION: C.E.A.
Title 2 (english) Study of the cooling conditions during the blow-down phase of a LOCA for a PWR - OMEGA loop.	Project leader C.E.A. -DTCE/STT M.RICQUE Scientists :
Initiated (date) Septembre 1974 — Completed : (date) Status : progressing — Last updating (date) Janvier 1975	

1. But général

Etude des conditions de refroidissement pendant la phase de décompression lors d'un accident de perte de caloporteur d'un P.W.R.

2. Objectifs particuliers

Etablir des corrélations de coefficients d'échange ainsi que de burn-out dans les conditions de dépressurisation.

3. Installations expérimentales et programme

Boucle hors-pile OMEGA (au Centre d'Etudes Nucléaires de Grenoble)

Principales caractéristiques :

puissance électrique	:	4,5 MW
pression	:	170 bars
débit	:	0,2 à 18 kg/s
nombre de barreaux	:	1 à 36
hauteur chauffante	:	3,65 m

4. Etat du projet

Démarrage de l'installation fin 1974 sur des essais de régime nominal.

5. Prochaines étapes

Essais de dépressurisation sur un tube refrodi intérieurement
Essais sur une grappe.

6. Relations avec d'autres projets

Essais d'instrumentation sur CANON et DEDIF
Essais de débits critiques sur MOBY DICK
Essais en pile PHEBUS
Mise au point des programmes de calcul.DANAIDES

..../....

7. Documents de référence

- Dépressurisation et refroidissement de secours (boucles OMEGA et PHEBUS) par MM. FONTERAY, COURTAUD, PELTIER, BAILLY présentées au colloque AIEA des 5-9 février 1973 - Julich (R.F.A.) (disponible)

CLASSIFICATION

1.1.2

TITLE 1	PWR. COMPORTEMENT DES POMPES DU CIRCUIT PRIMAIRE (EN MONOPHASIQUE ET DIPHASIQUE) AU COURS D'UN ACCIDENT DE DEPRESSURISATION	COUNTRY FRANCE SPONSOR E.D.F. ORGANIZATION E.D.F.
TITLE 2	PWR. BEHAVIOUR OF PRIMARY PUMPS DURING A LOCA.	Project Leader E.D.F./DER/MAP
Initiated octobre 1973	Completed Decembre 1977	Scientists M. BLANC-PERRAUD M. SUREAU
Status	Last updating : 20.01.75	

I - GENERAL AIM

Introduction des courbes caractéristiques des pompes dans les codes de calcul d'accident de rupture de tuyauterie primaire.

II - PARTICULAR OBJECTIVES

Obtention des courbes caractéristiques des pompes primaires en fonction des paramètres mécaniques et thermodynamiques caractérisant leur fonctionnement.

III - EXPERIMENTAL FACILITIES AND PROGRAMME

Tous les essais seront effectués en régime stationnaire et comportent des phases suivantes :

a) Essais sur maquettes "froides", Plate-forme d'essais Pompes Turbines de CHATOU. La maquette de type 93 D d'échelle géométrique 1/2,2 sera faite en 1975 par WESTINGHOUSE. Les essais seront effectués en monophasique eau ; quelques essais complémentaires en émulsion air-eau seront peut être exécutés. Tous les domaines de fonctionnement répartis dans les 4 quadrants seront explorés.

b) Essais sur maquettes 'chaudes'. Boucle IHP7 CHATOU. Ces essais doivent respecter l'évolution des phases pendant la traversée de la machine tant en titre qu'en répartition des phases. Ces conditions étant incompatibles, deux types d'essais complémentaires sont envisagés.

...

La maquette, d'échelle géométrique 1/1[illegible], permettra d'effectuer les essais respectant l'évolution de la phase gazeuse. Si possible, cette même maquette sera utilisée pour effectuer les essais nécessitant le respect de la répartition spatiale des phases. Cependant, il est envisagé, si nécessaire, une maquette d'échelle géométrique 1/6.

IV - PROJECT STATUS

4.1 - Progress to date

Maquette froide : plans acquis. Programme d'essais en préparation.

Maquette chaude : étude complémentaire de similitude en cours.

4.2 - Essential Results

V - NEXT STEPS

Réception maquette froide	:	milieu 1975
Fin des essais maquette froide	:	1er semestre 1976
Réception maquette chaude	:	1er semestre 1976
Essais maquette chaude	:	1976 - 1977.

VI - RELATION WITH OTHER PROJECTS

Code CLYSTERE.

VII - REFERENCE DOCUMENTS

Néant.

VIII - DEGREE OF AVAILABILITY

Confidentiel.

CLASSIFICATION

1.1.2

TITLE 1	DEPRESSURISATION DE LA CUVE D'UN PWR.	COUNTRY FRANCE SPONSOR E.D.F./SEPTEN ORGANIZATION E.D.F.
TITLE 2	PWR REACTOR VESSEL'S BLOW DOWN.	**Project Leader** E.D.F./DER/LNH
Initiated avril 1974 **Status** 1ère phase en cours	**Completed** octobre 1975 **Last updating**: 20.01.75	**Scientists** M. DAUBERT M. SUREAU

I - GENERAL AIM

Rupture d'une canalisation du circuit primaire, étude de la dépressurisation de la cuve.

II - PARTICULAR OBJECTIVES

La rupture d'une canalisation du circuit primaire entraîne la propagation d'une onde de dépressurisation dans tout le circuit et en particulier dans la cuve du réacteur. Pendant les tous premiers instant (1 à 2 sec) l'eau reste sous phase liquide puis commence à se vaporiser, le fluide devenant diphasique. Le but de cette étude est la description et l'analyse de cette première phase pendant laquelle l'eau est encore liquide.

III - EXPERIMENTAL FACILITIES AND PROGRAMME

1ère phase : étude sur modèle physique de la propagation de l'onde de dépression dans la cuve.

2ème phase : schématisation du phénomène et mise au point d'un modèle mathématique devant s'intégrer à une modèle plus global prenant en compte tout le réacteur.

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IV - PROJECT STATUS

4.1 - Progress to date

Première phase de l'étude en cours jusqu'en avril 1975.

4.2 - Essential Results

V - NEXT STEPS

Avril 1975 - octobre 1975 : 2ème phase de l'étude.

VI - RELATION WITH OTHER PROJECTS

Cette étude pourra se prolonger par celle de la phase de dépressurisation pendant laquelle le fluide se vaporise.

. Code CLYSTERE.

VII - REFERENCE DOCUMENTS

Néant.

VIII - DEGREE OF AVAILABILITY

Rapport d'essais d'accès libre.

CLASSIFICATION
1.1.2

TITLE 1 MELANGE DANS LA CUVE D'UN PAWR DES ECOULEMENTS PROVENANT DES DIVERSES BOUCLES.

COUNTRY FRANCE

SPONSOR E.D.F./SEPTEN

ORGANIZATION E.D.F.

TITLE 2 MIXING OF THE FLOWS OF THE DIFFERENT LOOPS ENTERING THE REACTOR VESSEL.

Project Leader
E.D.F./DER/LNH

Scientists
M. PUGNET
M. LARMINAUX

Initiated Mai 1974 **Completed** mai 1975

Status Étude en cours **Last updating :** 20.01.75

I - GENERAL AIM

Déterminer ce que la différence de température des débits primaires des différentes boucles au niveau des entrées dans la cuve devient à l'entrée du coeur et dans les conduites de sortie.

II - PARTICULAR OBJECTIVES

Coefficients de mélange intervenant dans les codes de calcul de fonctionnement.

III - EXPERIMENTAL FACILITIES AND PROGRAMME

Cette étude est menée sur modèle physique à échelle réduite. Les différences de températures sont simulées sur le modèle par des différences de salinité.

IV - PROJECT STATUS

Etude en cours.

V - NEXT STEPS

Fin des essais en janvier 1975.

...

VI - RELATION WITH OTHER PROJECTS

Codes de calculs des transitoires accidentels.

VII - REFERENCE DOCUMENTS

Néant.

VIII - DEGREE OF AVAILABILITY

Rapport d'essais d'accès libre.

	Classification : 1.1.2 7.1
Title 1 (original language) AQUITAINE 2 PROGRAM	Country : FRANCE Sponsor : FRAMATOME CEA Organization FRAMATOME CEA
Title 2 (english) Dynamic studies of the mechanical and thermal effects which occur on primary piping during a LOCA.	Project leader: M. CAMPAN CEA M. TROUBLE FRA Scientists :
Initiated (date): JANUARY 1975 Completed (date): JUNE 1977 Status: PROGRESSING Last updating (date): JUNE 1975	

1. OBJECTIVES

The objectives of this test program consist of studying on reduced scale model under dynamic conditions the mechanical effects which happen at the level of primary piping and on surrounding structures in a case of LOCA.

The following effects will be studied :

(i) Measurement of reaction leads and pipe whip

(ii) Study of plastic hinge of an elbow

(iii) Study of conditions resulting from impact between a pipe and rigid structure

(iiii) Study of plastic deflection of a straight pipe in the event of a lateral break

(iiiii) Measurement of impact forces and jet thrust on surrounding structures

The results obtained will permit the calibration of computer programs which deal with problems related to the behavior of structures.

The test facility will further be apt to be used as testing stand for the calibration of fast transient two phase flow instrumentation.

2. PROJECT STATUS

A theoritical study for sizing the pressurised capacity has been done by FRAMATOME. The loop will represent a 3 loop PWR and the similitude ratio will be 1/10 of the full scale.

Preliminary studies of the instrumentation of the test section have been carried out jointly between C.E.A and FRAMATOME.

The explosive techniques used in the Space will be used for initiating break in a very short time.

3. NEAR TERM PLANNING

The construction of the test facility and the procurement of long delivery items will start in Fall 1975. In parallel some qualification tests of both instrumentation and explosive system will be carried out on a simplified loop. .

The test program will start Mid 1976.

4. RELATIONS WITH OTHER PROJECTS

NONE

5. AVAILABILITY OF "RESULTS"

Joint property of CEA and FRAMATOME.

Classification: 1.1.2	
Title 1 (Original Language): Notkühlprogramm-Hochdruckversuche Teilvorhaben: DWR-Post DNB Hauptversuche mit einem 25-Stabbündel (RS 0037 C - I.1.2, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Emergency Core Cooling Program - PWR Post DNB Experiments with a bundle of 25 fuel rods	Project Leader: Dr. Schad
Initiated (Date): 1. 1. 75 Status: Continuing Completed (Date): 31. 3. 76 Last Updating (Date): 31. 12. 1975	

General Aim

The thermohydraulic and heat transfer behaviour during instationary blow down processes have to be studied. Characteristic are two phases of transients after large leaks: in the first seconds of a postulated rupture in the primary coolant system, steep transients of pressure, mass flow in the core and heat flux density of the fuel rods occur; in the later part of the blowdown a phase of flat gradients follows.

Particular Objectives

In order to reach sufficient high accuracy of the measurements, the region of flat pressure gradients has to be investigated seperately from the initial phase of large pressure gradients; of special interest is the heat transfer during high pressure gradients (begin of the blowdown). The experimental investigations (hot channel conditions) require special adaption of the test facility.

Experimental Facilities

The experimental facilities are described under "test equipment for controlled bundle measurements (BET)", Project RS 37, 37/1 and 37/2 I.2.2, Annual Reports No. A 74.
The facility was modified by adjusting a cold water injection system down stream of the bundle. Thus fast transients could be achieved at

the beginning of the blowdown with hot channel conditions.
In addition the volume upstream of the test section was reduced, which means less flashing volume.

Research Program
The research program is divided into two parts: in the first part DNB-tests will be run, which investigate the problem of DNB delay. The second part contains DNB tests for the determination of the heat transfer in the Post DNB region.

Project Status/ Progress to Date
A computer program was developed and tested, which calculates the control and drive values of the testloop.

The first part of the program has been completed, 21 DNB-tests were run under 11 different conditions, corresponding to primary coolant system ruptures between RPV and steam generator or steam generator and pump.

Before the tests started the stationary initial values were evaluated. The heat power, the pressure, the lifting of the feedwater valve and the inlet area of the control valve were determined as a function of time. The blowdown tests were started automatically and the run was computer controlled.

During 40 sec the computer had to store data from 140 instrumented spots. The maximum frequency was 100 Hz per channel.

Project Status/Essential Results
The first tests yielded experimental data of

a) the DNB delay time
b) the DNB position
c) the shorttime Post-DNB-phase

The electrical power of the bundle was simulated corresponding to hot channel conditions in the reactor. The pressure drop in the bundle was reduced from 141 bar to 116 bar within 0,3 sec. All values were in good agreement with the specified ones.

Next Steps

Longtime DNB-tests will be performed. The adaption of the test loop has already been started. The computer program must be modified with respect to the prediction of some control data.

Relation with Other Projects

See RS 37 - A 74

Reference Documents/Degree of Availability

No reports available.

Classification: 1.1.2

Title 1 (Original Language): Theoretische und experimentelle Untersuchungen über Modellgesetze für instationäre Wärmeübergangsbedingungen in wassergekühlten Reaktoren bei Notkühlung (RS 48 - I.1.1., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** TU-Hannover
Title 2 (english): Theoretical and Experimental Investigations on Model Laws for Instationary Heat Transfer Conditions in Water Cooled Reactors under Emergency Cooling Conditions	**Project Leader:** Prof. Dr. Mayinger
Initiated (Date): 1970 **Status:** finished **Completed (Date):** 1975 **Last Updating (Date):** March 1975	

1. General aim

The general aim of these activities was to evaluate scaling laws for the transient behaviour in reactors under loss of coolant conditions by experimental and theoretical investigations.

2. Particular objectives

The main problems were the investigations of burnout delay time, heat transfer mechanism in the post burnout region and flowpattern under blowdown conditions.

3. Experimental facilities and program

Two loops had been available for the experimental part of this investigation. The different aggregates of the circuit and the used measuring techniques had already been described in detail in the reports IRS-F-12 and IRS-F-18.
The test program during the whole period was separated into parts:

1. Tests for investigating dryout delay and post-DNB-heat transfer. The corresponding parameter ranges had been given in the last report A74.
2. Model experiments corresponding to conducted water tests, e.g. by KWU /1/.
3. Tests for investigating the influence of loop components on the heat transfer behaviour in the heated section.
4. Visual and X-ray-tests to investigate the flow patterns during blowdown.

4. Project status

4.1 Progress to date

Within this report period the investigations had been finished. The evaluated calculation models for the onset of dryout and the post-DNB heat transfer had been verified and improved, especially the dryout delay model. From these models the scaling laws for translating the experimental results to original water conditions had been deducted.

Based on these scaling laws the final model blowdown tests had been carried out corresponding to original water blowdown tests with an inside cooled tube and a four-rod-bundle conducted by KWU /1,2/.
All test results as well the own model experiments as the corresponding water tests had been recalculated with the developed theoretical models. A good agreement between measured and calculated values as well for model tests as for original ones could be recognized.

4.2 Essential results

The dryout delay model - roughly described in the report A74 - started from a mass and energy balance for the two-phase flow within a heated channel. Combining both balances one gets an equation for the cross section of the liquid film A' in dependance of energy input $\frac{\dot{q}(z)\cdot U_h}{\rho'\cdot r\cdot A_{tot}}$, evaporation because of pressure decrease

$$\frac{1}{\rho'}\cdot\frac{\partial\rho'}{\partial t} + \frac{1}{r}\left(\frac{\rho''}{\rho'}\cdot\frac{\partial h''}{\partial t} - \frac{\partial h'}{\partial t}\right)$$

and of entering and escaping mass flows $\frac{\partial\dot{M}'}{\partial z} + \frac{\partial\dot{M}_E}{\partial z}$ in the film ($\dot{M}'$) and entrained droplets in the gas core ($\dot{M}_E$).

$$\frac{1}{\rho'}\left(\frac{\partial\dot{M}'}{\partial z} + \frac{\partial\dot{M}_E}{\partial z}\right) + \frac{\partial A'}{\partial t} + A'\left(\frac{1}{\rho'}\cdot\frac{\partial\rho'}{\partial t} + \frac{1}{r}\left(\frac{\rho''}{\rho'}\cdot\frac{\partial h''}{\partial t} + \frac{\partial h'}{\partial t}\right)\right) + A_{ges}\left(\frac{\dot{q}(z)\cdot U_h}{\rho'\cdot r\cdot A_{ges}} - \frac{\rho''}{\rho'}\cdot\frac{1}{r}\cdot\frac{\partial h'}{\partial t}\right) = 0$$

This equation can be integrated by neglecting the length-dependance of the film thickness. Then only the mass flow rates at the exit of the heated section are unknown, while all other values like pressure decrease, heat flux and entrance mass flow rate must be given by experimental results or by a thermohydraulic code like Relap or Bruch.

The mass flow rates in the film and the droplets may be calculated by the definitions:

$$\dot{M}' = \rho' \cdot A_{ges} \cdot (1-\varepsilon) \cdot (1-e) \cdot w'$$

$$\dot{M}_E = \rho' \cdot A_{ges} \cdot (1-\varepsilon) \cdot e \cdot w_E$$

where the phase velocities may be calculated from the transient pressure drop.

$$w' = \sqrt{(1-\varepsilon)\left(\frac{dp}{dz}\right)_{2ph} \cdot \frac{2 \cdot d_h}{\zeta' \rho'}}$$

$$w'' = \sqrt{\frac{\sqrt{A_{ges}(A_{ges}-A')}}{A_{ges} - 75\, A'} \cdot \left(\frac{dp}{dz}\right)_{2ph} \cdot \frac{2 \cdot d_h}{\zeta'' \rho''}}$$

The cross section of the area covered by droplets may be gained by a dimensionless number

$$A_E(L,t) = A_E(L,0) \cdot \left[1 + C \cdot \left(\frac{\frac{dp}{dz} \cdot \sqrt{A'}}{\rho' \cdot w_E^2}\right)^n\right]$$

With these substitutions the first equation can be solved by a computer program. Within the solution the constant and the exponent for calculating transient entrainment have to be fit to measurements.

By introducing characteristic values for pressures, lengths time and velocities, the initial equation may be given in a dimensionless form, from which the scaling numbers for modelling the dryout delay can be taken. Applying all numbers as well for model as for original conditions one gets the necessary ratios of the system describing parameters to conduct model tests:

$$\frac{\dot{m}_M}{\dot{m}_O} = \frac{n'_M}{n'_O} \qquad \frac{\Delta p_M}{\Delta p_O} = \frac{\rho''_O}{\rho''_M} \cdot \frac{\left(\frac{\rho''}{\rho'}\right)_O}{\left(\frac{\rho''}{\rho'}\right)_M} \cdot \left(\frac{\dot{m}_M}{\dot{m}_O}\right)^2$$

$$\frac{\dot{q}(z)_M}{\dot{q}(z)_O} = \frac{r_M}{r_O} \cdot \frac{\dot{m}_M}{\dot{m}_O} \qquad \frac{p_{S,M}}{p_{S,O}} = \frac{\rho'_M}{\rho'_O} \cdot \frac{\left(\frac{\partial \rho'}{\partial p}\right)_O}{\left(\frac{\partial \rho'}{\partial p}\right)_M} \text{ oder } \frac{r_M}{r_O} \cdot \frac{\left(\frac{\partial r}{\partial p}\right)_O}{\left(\frac{\partial r}{\partial p}\right)_M}$$

The post-DNB heat transfer model had already been presented in the report A74. From this model the scaling laws for model tests are:

$$K_1 = Nu \qquad K_4 = \eta'/\eta'_W$$

$$K_2 = Re' \qquad K_5 = d_{Tr}/D$$

$$K_3 = Pr' \qquad K_6 = \varepsilon$$

In the fig. 1 - 4 four examples for the agreement between experimental and theoretical results are given. A comparison of measured and calculated dryout-delay times in dependance of variied steady state heat flux shows a deviation of less than 20% for a hot leg break and less than 25 % for a cold leg break. The same agreement can be recognized for a comparison of water blowdown tests by KWU /1/ in fig. 3.
The comparison of the post-DNB heat transfer model with experimental results, given in fig. 4, shows a good accuracy for the developed model (I.f.V.), but also for the model of Dougall-Rohsenow /3/ used in the thermohydraulic codes.

References

/1/ KWU (AEG), Notkühlprogramm DWR and SWR, AEG-E3-1941, Mai (1971)

/2/ KWU/R5-2882, Abschlußbericht über die Abblaseversuche mit 4-Stabbündeln Großwelzheim, Jan. 1974

/3/ Rohsenow W., Griffith P.: Correlation of maximum heat transfer data for boiling of saturated liquids
Chem. Engng. Progr. Symp. Ser. 52, Nr. 18, 47 (1956)

5. Next steps

The investigations had finished in March 1975.

6. Relation with other projects

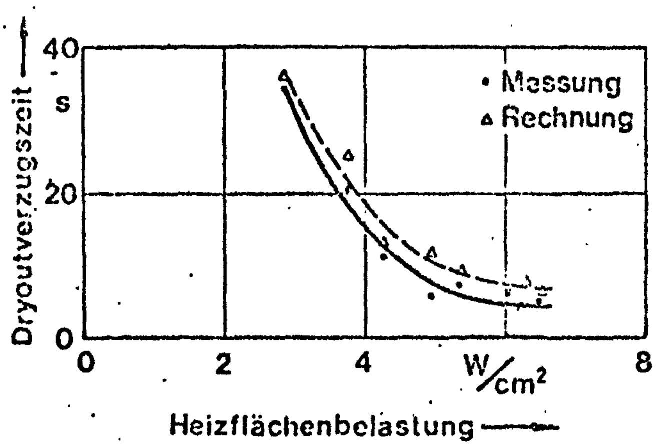
RS 37, RS 37-1 and RS 37-2

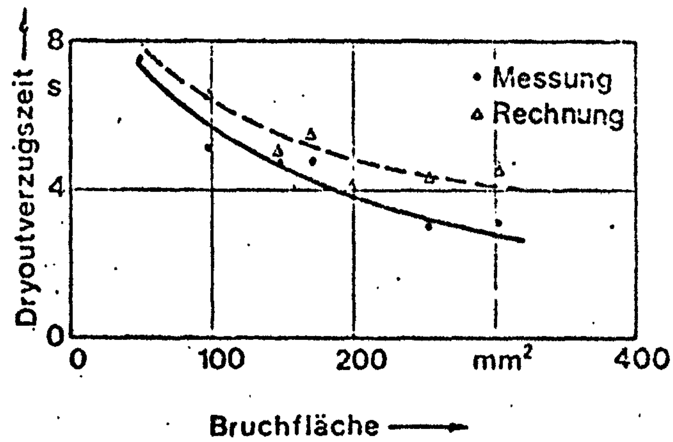
"Investigations of the Events within the Reactor Core under loss of Coolant and Emergency Cooling Conditions, High Pressure Experiments" at KWU, Großwelzheim.

RS 64: "Investigations of the steady and transient Critical Heat Flux of Multirod Bundles for PWRs and BWRs with Freon as Model Fluid".

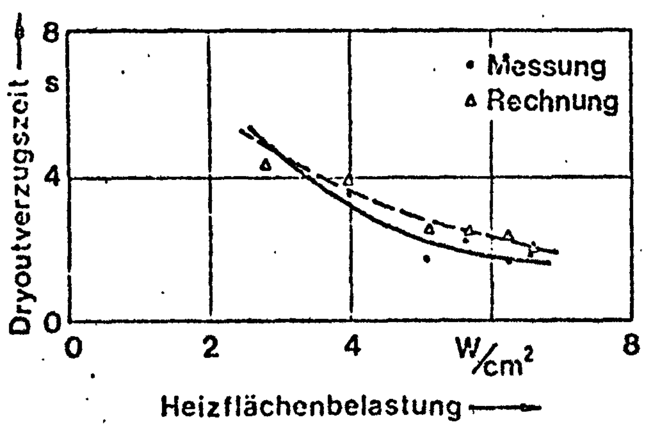
Comparison of measured and calculated dryout delay (4-rod bundle) as function of

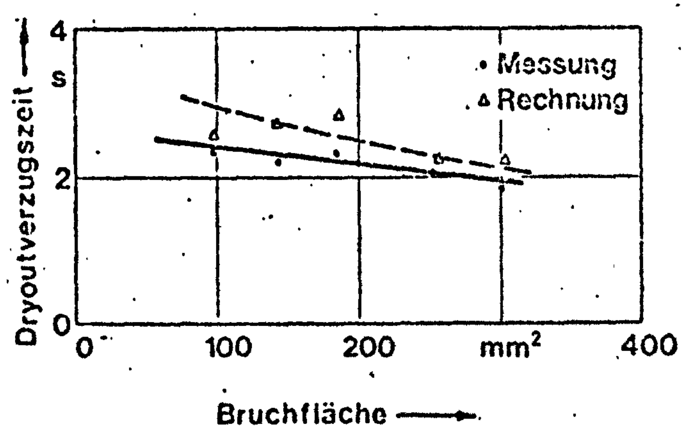
hot leg break: heat flux ↓ cross section of break area ↓

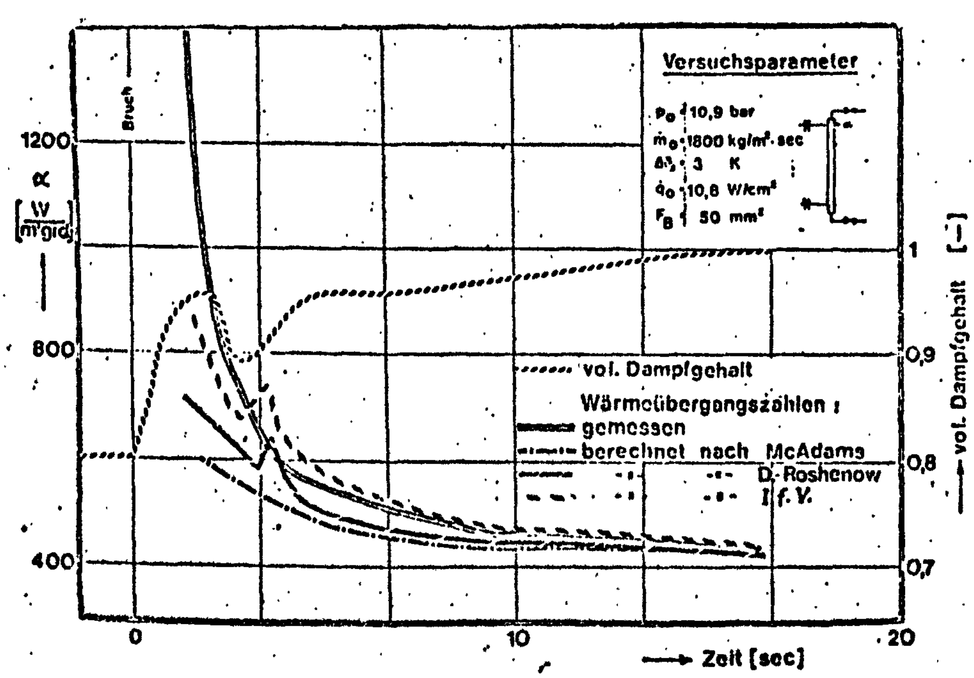




cold leg break: heat flux ↓ cross section of break area ↓







comparison of measured and calculated heat transfer coefficients

7. Reference documents

/1/ T.U. Hannover; 1. Jahresbericht BMFT-FB-48-01, Juli 1971
/2/ T.U. Hannover; 2. Jahresbericht BMFT-FB-48-02, August 1973
/3/ T.U. Hannover; 3. Jahresbericht BMFT-FB-48-03, Mai 1974
/4/ T.U. Hannover; 4. Jahresbericht BMFT-FB-48-04, Juni 1975
/5/ T.U. Hannover, 5. Jahresbericht BMFT-FB-48-05, Juli 1975
/6/ W. Belda, F. Mayinger, ETPFGM 1973, paper C3, Brussels 1973
/7/ W. Belda, F. Mayinger, ETPFGM 1974, paper E3, Harwell 1974
/8/ W. Belda, F. Mayinger, ETPFGM 1975, Haifa 1975
/9/ W. Belda, F. Mayinger, Reaktortagung 1974, Paper 102, April 1974
/10/ W. Belda, F. Mayinger, Reaktortagung 1975, Paper 118, April 1975
/11/ K.P. Viert, F. Mayinger, Reaktortagung 1975, Paper 117, April 1975
/12/ Quarterly and annual reports in the series IRS-F

8. Degree of availability

The annual reports, BMFT-FB and the IRS-F are available by IRS, the other ones are free.

Classification: 1.1.2	
Title 1 (Original Language): Experimentelle und theoretische Untersuchungen zum thermohydraulischen Verhalten des Cores in der ersten Blowdownphase (RS 163 - I.1.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: TU Hannover
Title 2 (english): Theoretical and Experimental Investigations on Thermo- and Fluiddynamic Behaviour of the Reactor Core during the First Blowdown Phase	Project Leader: Prof. Dr.-Ing. F. Mayinger
Initiated (Date): 1975 — Completed (Date): 1979	
Status: continuing — Last Updating (Date): December 1975	

1. General aim

The general aim of these investigations is to predict the thermo- and hydrodynamic behaviour in the first blowdown period during the loss of coolant accident and the following emergency cooling phase.

2. Particular objectives

The main problems are the experimental and theoretical investigations of dryout-delay-time under consideration of entrainment- and mixing processes. Furthermore, the influence of loop components on the heat transfer behaviour in the test section during LOCA-conditions must be tested.

3. Experimental facilities and program

During this period an entrainment- and a mixing-test-section was constructed. The entrainment-test-section is an inside cooled tube with a heated length of 5 m (16,4 ft). At the end of the heated length there is a slit-device, which separates the liquid film at the inside of the tube wall and the liquid droplets flying in the gas-core of a two-phase annular-flow (fig. 1). Through a glass-window at the top of the slit-device, pictures can be taken from the droplets with a miniature camera and with a high-speed camera. In the first tests, the entrainment behaviour of a steady state annular flow has been investigated. Parameter range for steady state entrainment investigations are:

initial mass flow rate	200 ... 900 kg/m² sec
system pressure	8 ... 14 bar
(system pressure reduced with p_{crit}	0,2 ... 0,33
initial subcooling	5 ... 15 K
heat flux	6 ... 8 KW

At the mixing-test-section the transverse-exchange between two parallel subchannels with equal and different mass flow has been tested by measuring the dimming of a coloured single phase water flow with aid of a photoelectric diode. First investigations about the transverse exchange with air water mixtures has been conducted.

During the last period of this report, a second mixing-test-section for Freon-12 tests has been constructed.
Furthermore, the computer program COBRA II was accomodated to the CDC-Cyber 76 computer at the RRZ Hannover.

4. Project status

The following investigations had been carried out during this report period:

1. Construction of a special entrainment test-section
2. Entrainment experiments at steady state with a miniature camera and with a high speed camera
3. Analyse of the test results
4. Tests to take pictures of an annular flow with liquid entrainment with aid of a glass fibre optic
5. Evaluation of dimensionless numbers to describe the entrainment behaviour at steady state
6. Construction of a 16-rod-bundle test-section
7. Mixing investigations at parallel subchannels

After testing the photo-technique to take pictures of the droplets in direction of the flow on air-water mixture, quantitative measurements with the modelfluid Freon 12 had been taken. The test parameters are systematically variied in the range described in chapter 3.

For the entrainment investigations in the rod bundle a new photographic method was tested parallel. With a glass fibre optic, as shown in fig. 2, photographic investigations even in complicated subchannels of rod bundles can be carried out.

During the analysis of the test results, four dimensionless numbers were evaluated to describe the steady-state entrainment behaviour of a two-phase annular flow. A momentum balance at a differential part of a wavy annular flow surface was applyied in direction of the flow as well as cross to it. A comparison of the forces which attack the wave at the surface of the liquid film lead to an equation which includes four dimensionless numbers. These numbers are appropriate to describe the entrainment behaviour of a steady state annular two-phase flow.

4.2 Essential results

An entrainment photo gained in the first tests is shown in fig. 3 at the end of this report. By measuring and counting the droplets we find e.g. a decreasing entrained mass flow with increasing total mass flow (fig. 4). But even at steady state conditions the entrainment mass flow has no constant value. We found fluctuation about 60 % related to the calculated average value. Fig. 5 shows the oscillations of the entrainment mass flow as a function of time: 30 minutes after reaching the steady state conditions, 36 photos were taken with a frequency of 3 pictures/sec.

The photo technique with the fibre optic was tested in tube geometries applying different illumination techniques. The results of these tests are shown in fig. 6.

In the theoretical part of this work the following dimensionless numbers were evaluated:

$$\Pi_1 = \frac{\dot{m}_{tot\ inlet} \cdot \varepsilon \cdot d''}{\nu'' \cdot \rho''} \cdot Re''$$ considers the influence of mass flow

$$\Pi_2 = \frac{\partial p/\partial z \cdot dz - \tau}{\sigma} \cdot \ell$$ considers the influence of test section geometrie and the most important physical properties of the fluid

$$\Pi_3 = \frac{\dot{q}_0}{\dot{m}_{tot.\ inlet} \cdot r_0} \left(1 - \frac{\Delta h}{r_0}\right)$$ considers the influence of heat flux and initial subcooling

$$\Pi_4 = \frac{\dot{M}_{ENTR}}{\dot{M}_{Fl\ tot}} = E$$

Fig. 7 and 8 show the experimental results of Bennet et al (ref. 2) carried out at a heated test section as a function of Π_1, Π_2, Π_3 and Π_4.

5. Next steps

The next steps will be steady state entrainment investigations at higher mass flow rates (BWR-conditions: 1800 kg/m²sec). Entrainment and mixing tests under blowdown conditions simulated at the inside cooled tube will follow. The photo technique with the glass fibre optic must be optimated till the end of the construction of the 16-rod bundle.

6. Relations with other projects

RS 37; 37-1; 37-2

Investigations of the events within the reactor core under loss of coolant and emergency cooling conditions at KWU, Großwelzheim

RS 48

Theoretical and Experimental Investigations on Model laws for Instationary Heat Transfer Conditions in Water Cooled Reactors under Emergency Cooling Conditions

RS 64

Investigations of the steady and transient Critical Heat Flux of Multirod Bundles for PWR's and BWR's with Freon as Model Fluid

RS 179

Phaseseparation

7. Reference dominant

/1/ W. Belda, F. Mayinger: Calculation model and experimental results concerning dryout delay time during blowdown; European Two-Phase Flow Group Meeting Haifa, June 2-5, 1975

/2/ W. Belda, F. Mayinger: Some deliberations and measurements concerning dryout delay time; European Two-Phase Flow Group Meeting, Harwell, June 3-7, 1974

/3/ Quarterly reports BMFT-RS 163 V 75/2, V 75/4 in the series IRS-Forschungsberichte

8. Degree of availability

The annular reports, BMFT-FB and the IRS-Forschungsberichte are available by IRS, the other ones are free.

9. Budget

1 year (10 month) DM 284.600.00

10. Personell

References

/1/ Arnold C., Hewitt G.F.: Further developments in the photography of two-phase gas-liquid flow; AERE-R 5318 (1967)

/2/ Hewitt G.F., Bennet A.W.: Studies on burnout in a uniformly heated round tube AERE-R 5072 (1966)

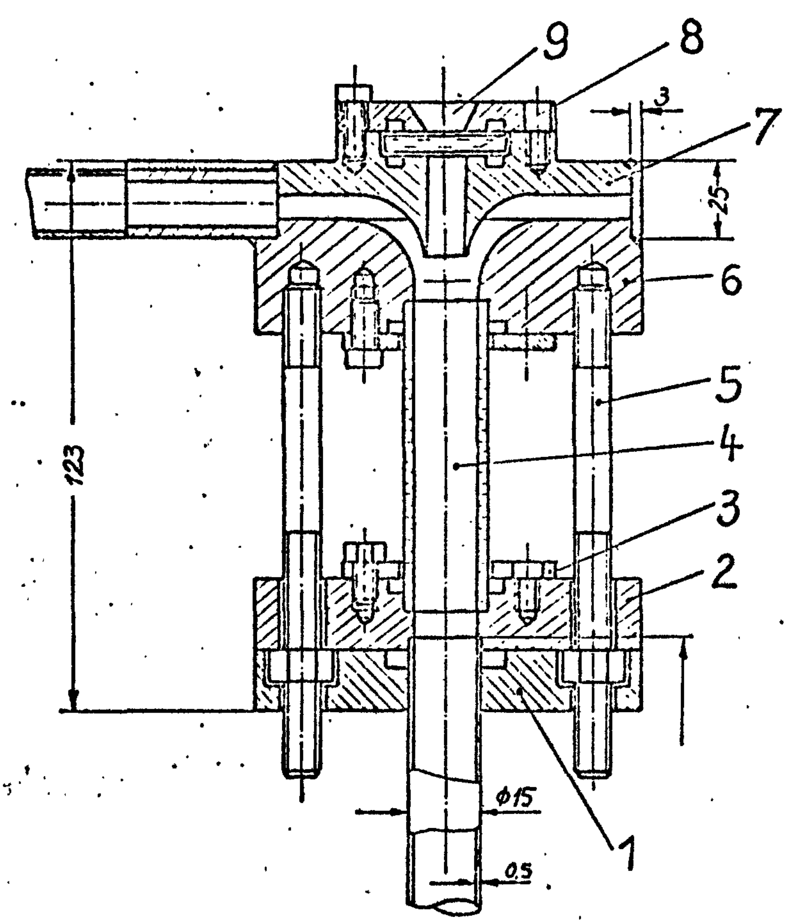


Fig. 1: Slit device at the end of the heated test-section

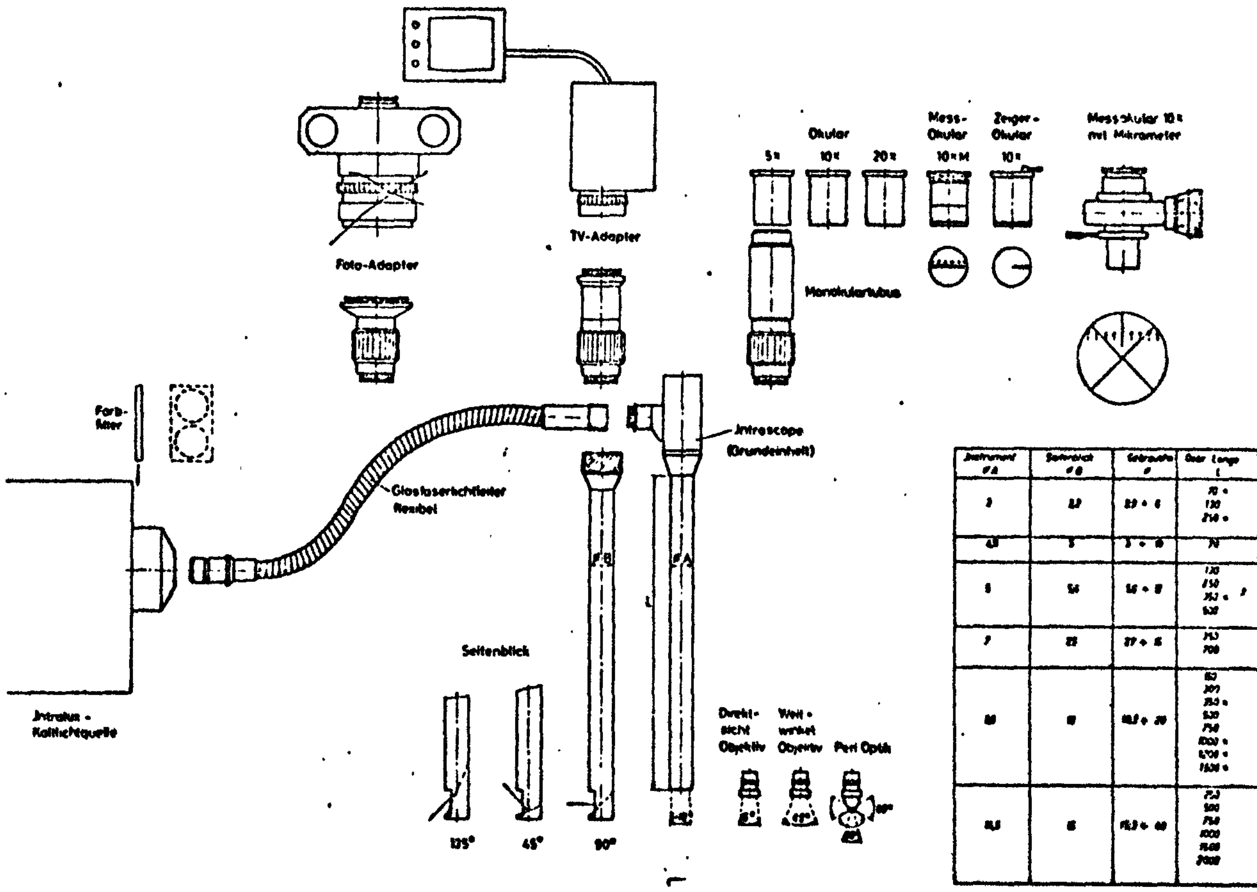
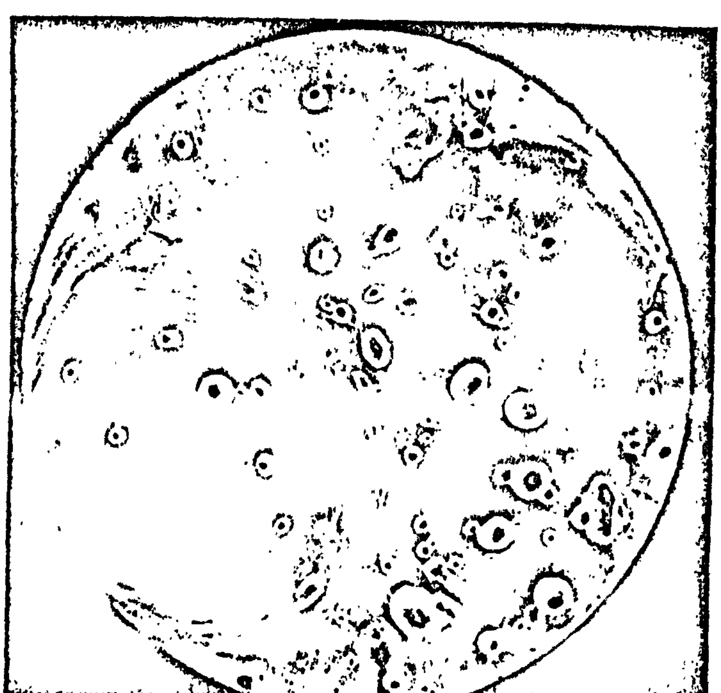


Fig. 2: Glass fibre optic equipment



$\dot{m}_{inlet}$ = 250 kg/m² sec
p = 8 bar
l_{heated} = 5 m
ε_{out} = 0,9
ϑ_{out} = 308 K
$\dot{Q}$ = 5,4 KW

Fig. 3

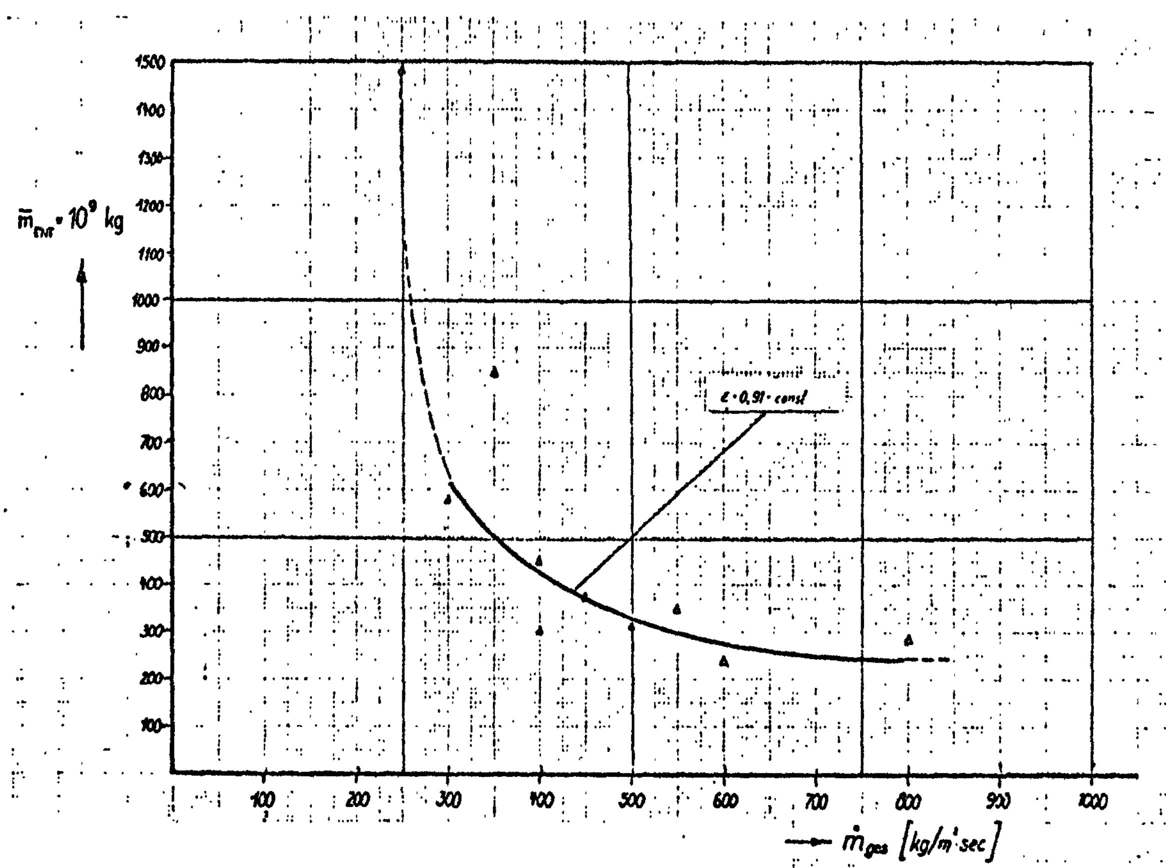


Fig. 4: Entrainment as a function of total mass flux

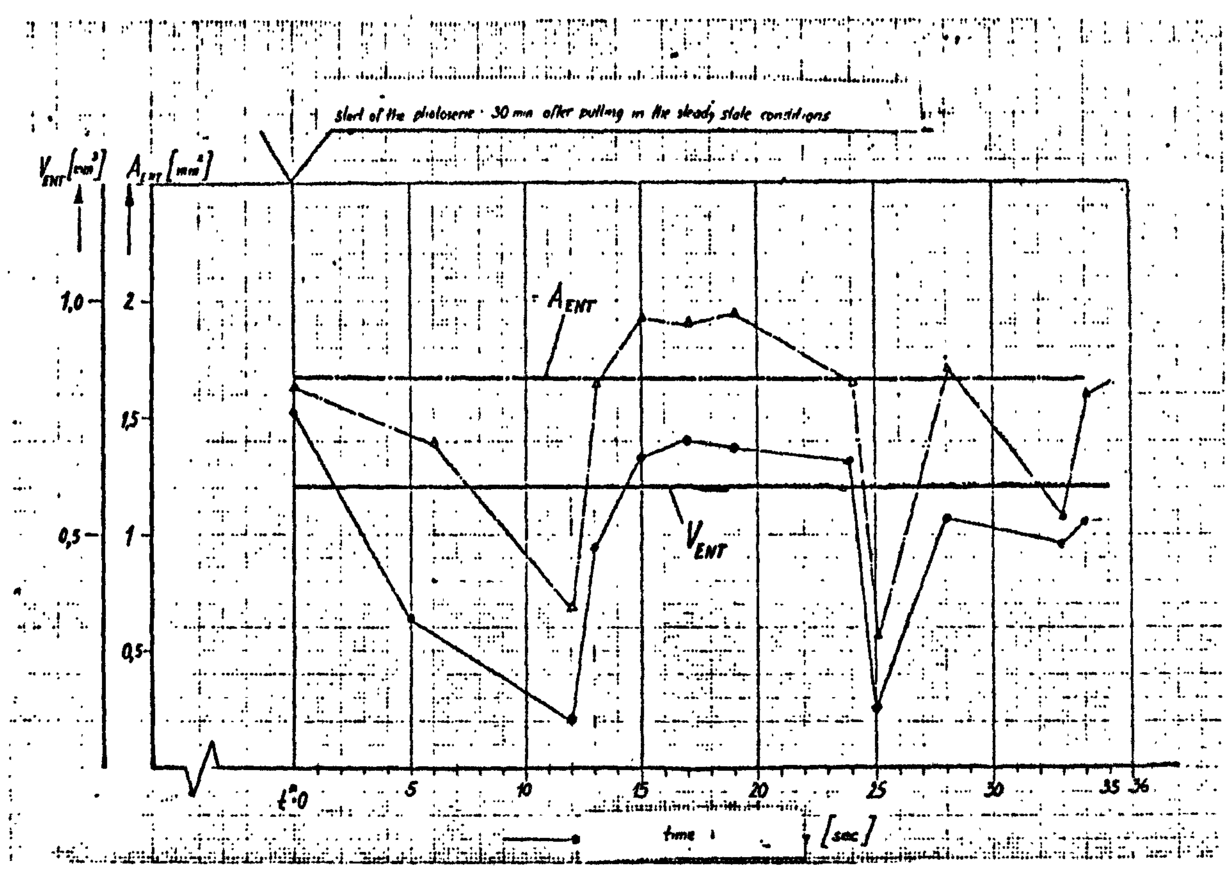


Fig. 5: Unsteadiness of entrainment mass flux

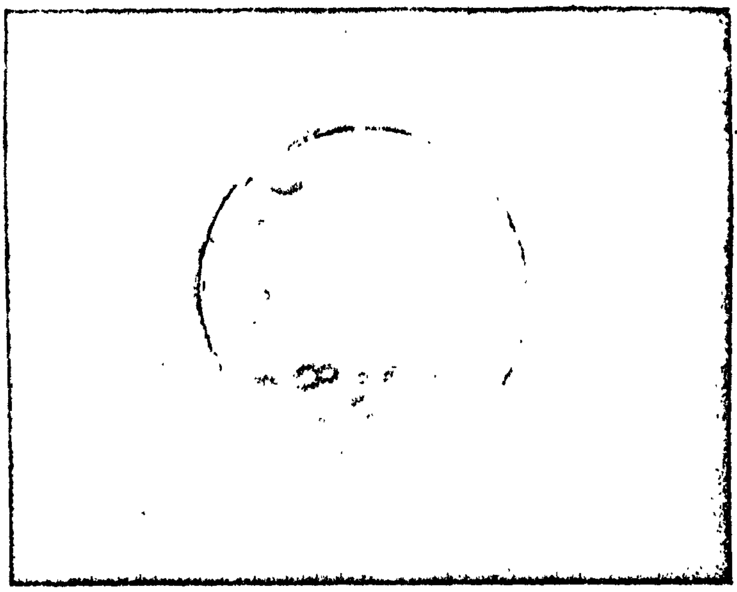


Fig. 6: Bubble flow - picture taken with a glass fibre optic

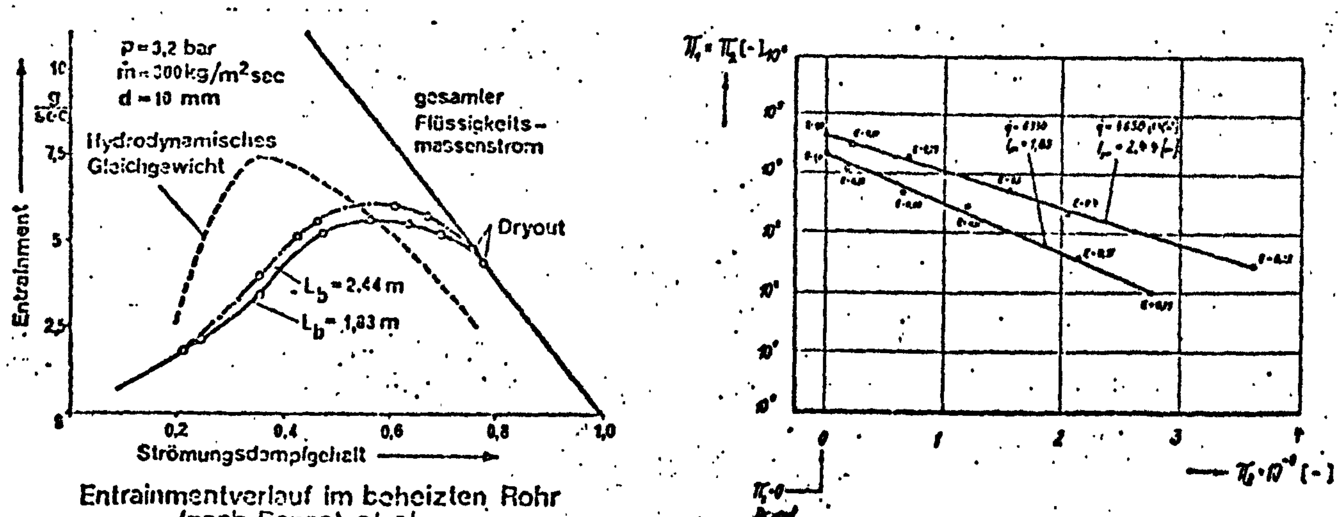


Fig. 7/8: Entrainment measurements of Bennet /2/

Classification: 1.1.2	
Title 1 (Original Language): Entwicklung von Meßverfahren zur Bestimmung transienter Massenströme (Dampf/Wasser) durch Signalkorrelation (RS 135 - I.1.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: TU Berlin
Title 2 (english): Development of Methods for Measuring Transient Two-Phase Mass Flows (Steam/Water) by Signal Correlation	Project Leader: Prof. Dr. U. Wesser
Initiated (Date): January 1, 1975 Completed (Date): December 31, 1977 Status: continuing Last Updating (Date): December 1975	

1. General Aim

The purpose of this project is the accurate measurement of the cross section average mass flow of a steam-water mixture flowing in a pipe as a function of time during blow-down experiments. The investigations are part of experiments in the field of loss-of-coolant accidents in nuclear reactor power plants.

2. Particular objectives

To detect the cross section average mass flow of a steam-water mixture it is necessary to measure either the average density of the fluid or its average velocity in the pipe.
The measurement of the cross section average fluid density is based on the attenuation of multiple gamma or x-ray beams.
The determination of the fluid velocity is based on measuring the transit time of variations in fluid temperature between fixed points along the direction of flow. The transit time is determined by using cross correlation techniques, while the temperature fluctuations are detected by thermocouples.
An alternative method to measure the fluid velocity is the use of the signals due to density fluctuations detected at two points along the flow path.

3. Experimental facilities and program

3.1 Experimental facilities

There is a one- and two-phase-flow water loop for low fluid velocities up to 5 m/s which is used to study and calibrate the measurement apparatus.
A small blowdown facility (50 bar) was built up and used first for mechanical testing of the thermocouple probes and second to run the apparatus under real blow-down conditions.

3.2 Research Program

The estimated schedule and the envisaged developments are shown in figure 1.

4. Project Status

4.1 Progress to date

4.11 Density Measurement

A theoretical analysis was made to calculate the estimated measurement errors of the multi-beam and scattered beam methods. These two densitometer principles were analysed under different fluid saturation pressures with beam energies from 10 keV up to 1.3 MeV and different numbers and angles of the beams.
To verify these theoretical approaches 'mock up' measurements were made with Cs-137-, Co-60-beams and 30 keV x-rays.

4.12 Velocity Measurement

Fluid velocity transients with velocity ratios of 1:10:1 within a time greater one second were recorded at the two-phase flow water loop and then computed with a Time-Data time series analyser. The result of this computation is the plot of the function of fluid velocity versus time without any operator interaction.

4.2 Essential Results

4.21 Density Measurement

In the lower beam energy regime the 'scattered beam technique' seems to be more accurate than multi-beam systems (result of theoretical analysis). The possibility of using x-ray tubes is advantageous because of less problems with shielding and other security aspects.

4.22 Velocity Measurement

The cross correlation technique applied to the signals due to natural temperature fluctuations measured at two fixed points along the flow path proves to be a successful method for determining fluid velocity in a pipe.

5. Next Steps

Velocity measurements will be made in two-phase flows with phase change during the velocity transient in the next future. These experiments especially emphasize the possibility of separating the two peaks in the cross-correlation function (probably due to the two different velocities of phases) to determine the phase slip. Later the experiments are continued under blow-down conditions.

6. Relations with other Projects

It is intended to test the developed mass-flow-meter under nuclear reactor conditions at the two-phase flow loop at GfK-Karlsruhe. The purpose is to install the mass-flow-meter in the LOCA-Loop at EURATOM-Ispra.

7. References

1.) D. Lübbesmeyer; Wesser, U.: Theoretische Untersuchungen von Dichtemeßmethoden mit Gamma-Strahlung. TUBIK 45, Report of the Institut für Kerntechnik, TU Berlin (Oct. 1975)

8. Degree of availability

1.) free

<table>
<tr><td colspan="3">Classification: 1.1.2</td></tr>
<tr><td colspan="2" rowspan="3">Title 1 (Original Language):
Entwicklung einer Massendurchsatz-Meßmethode für transiente Zweiphasen-Strömungszustände mittels der magn. Kernspinresonanz
(RS 136 + 161 - I.1.1, Jahresbericht A 75)</td><td>COUNTRY:
BRD</td></tr>
<tr><td>SPONSOR:
BMFT</td></tr>
<tr><td>ORGANIZATION:
KWU, Erlangen</td></tr>
<tr><td colspan="2">Title 2 (english):
Development of a Method for the Determination of Transient Two-Phase-Massflow by Means of NMR-Techniques</td><td rowspan="2">Project Leader:
F. Winkler</td></tr>
<tr><td>Initiated (Date):
1. 8. 1974
Status:
Completed</td><td>Completed (Date):
30. 9. 1975
Last Updating (Date):
31. 12. 1975</td></tr>
</table>

General Aim

A study was conducted on the feasibility for developing a method for the determination of coherent flow velocity spectra of quasi-stationary and transient fast two-phase flows with speeds up to about 10^4 cm/s and under conditions approaching the critical point of water. This should be practicable by presently available NMR-techniques and magnet technology.

Particular Objectives

It was proposed to observe the spin-echo-time-series resulting from the discharge of a magnetically labled amount of fluid leaving the measuring coil. From its envelope the velocity spectrum may be derived by an integral transform. This spectrum describes the distribution of the mass and their state of phase as function of their velocity. Integration with respect to the velocity yields the mass-flow rate. This measuring technique was derived theoretically in a first order approximation.

Experimental Facilities/Research Program

In preparation of flow experiments in conjunction with the derivation of the transform, the longitudinal and transversal relaxation times as well as the diffusion coefficients of water and especially steam have to be determined with appropriate precision. The development of a prototype flow-spectrometer of 50 mm ID was described as well as measuring problems discussed at simultaneous operation with a hydro- and thermodynamic experimental facility like the blow-down experiment RS 109 at Euratom-CCR at Ispra.

Project Status

The possibilities of different NMR methods were discussed and analyzed in cooperation with Prof. Dr. R. Kosfeld and his coworkers from the RWTH Aachen and with Mr. B. Knüttel from Bruker-Physik Company. Preliminary tests were conducted. The influence on the NMR-signal of the relaxation time and of temperature- and velocity transients in the flow were estimated as well as the transmitter-frequency and the S/N ratio determined. The influence of the nonuniform high frequency field on the signal was derived and calculated for test conditions. The magnetic shielding for a prototype-flow-spectrometer at the test-facility RS 109 was calculated.

Project Status/Essential Results

A study on the development of a mass-flow rate measuring technique for two-phase flows applying NMR-techniques was worked out and presented to the "AG Meßmethoden". It demonstrated the development of NMR-Methods for the observation of the hydrodynamics of stationary and transient two-phase flows to be promising.

Next Steps

The development of this measirung technique has been proposed under the auspices of Prof. Kosfeld, RWTH Aachen.

Relation with Other Projects

RS 36: Emergency Core Cooling Program - Low Pressure Experiments
RS 109: Influence of the PWR Loops on the Blowdown
RS 93: Impingement Forces on Structures Caused by a Two-Phase Jet
RS 16: Dynamic Effects on Internals of Pressure Vessels
RS 50: Phenomena Occurring within a Containment during Blowdown

Reference Documents/Degree of Availability

W.H. Bergmann "Studie zur Entwicklung einer Massendurchsatz-Meßmethode für transiente Zweiphasen-Strömungszustände unter Anwendung der magnetischen Kernspinresonanz (Nuclear Magnetic Resonance)"
Abschlußbericht zum Förderungsvorhaben BMFT RS 136 + 161
KWU-Erlangen (Okt. 1975)
Company Confidential

Classification: 1.1.2		
Title 1 (Original Language): Gemeinsamer Versuchsstand zum Testen und Kalibrieren verschiedener Zweiphasen-Massenstrommeßverfahren (RS 145, PNS 4215 - I.1.1., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Joint Test Rig for Tests and Calibration of Different Methods of Two-Phase Mass Flow Measurement		**Project Leader:** J. Reimann
Initiated (Date): 1.10.1974	**Completed (Date):** 31.12.1976	
Status: continuing	**Last Updating (Date):** December 1975	

1. General aim

On behalf of the Federal Ministry of Research and Technology, the Institute of Reactor Components (IRB) is building a test rig which will be used for testing and calibrating different methods for measuring unsteady-state two-phase mass flows.

2. Particular objectives

The accuracy of the different measuring methods will be dependant on the flow regime. Therefore, it is necessary to have a method to identify the flow pattern. Two methods will be used :

- an optical technique (sapphire windows and high speed photography)
- an impedance probe

3. Experimental facilities and program

The test rig is designed for the generation of a steady-state steam-water flow with a quality between 1 and 0 and a maximum pressure of 160 ata. Depending on the quality, the mass flow will reach a maximum level of 10 to 20 t/h. The test section has a total length of 8 m and consists of exchangeable tube sections. The tube diameters are 20,50 and 80 mm.

4. Project status

4.1 Progress to date

The most components like valves, pipes, fittings and the components of the control-and measuring-system are delivered. The mounting of the control-and measuring-system has been started, the mounting of the pipes has been nearly finished.

4.2 Essential results

To develop the electronic equipment for the impedance probe and to compare the two methods of flow regime identification a water-air loop had been built up. Experiments have shown that the signal of the impedance probe is excellent suited also in flow regimes in which the visualization technique gives results which are difficult to interprete.

5. Next steps

After the completion of the loop some time will be necessary for the setting in operation. The begin of the experiments is expected to be in April 1976.
Conceptional work is started to extend the test-facility for unsteady-state conditions.

6. Relation with other projects

RS 147	Improvement of a Drag Body for the Mass Flow Measurement in Blowdown Experiments, BATTELLE-Institut, Frankfurt, 1974-1975
RS 146 (PNS 4214)	Development of a Radionuclide Method of Mass Flow Measurement in Unsteady-State Multiphase Flows, GfK-LIT, Karlsruhe, 1974-1979
RS 136	Development of a Method for the Determination of Transient Two-Phase-Mass-Flow by Means of Nuclear Magnetic Resonance Techniques,KWU-Erlangen, 1974-1975

RS 135 Development of a Method of Mass Flow Measurement by Means of Temperature-Signal-Correlation,
TU-Berlin, planned in 1975

RS 1o9 Experimental Investigation of the Influence of PWR-Loops on Blowdown,
Euratom-Ispra

7. Reference documents

Report KFK 213o (1974) p. 163 (German)
Report KFK 2195 (1975) p. 18o (German)
KFK-Nachrichten 3/1975 p. 22 (German)

Reports in the series IRS-FORSCHUNGSBERICHTE

Quarterly reports:

Report period	Sept.-Dec.	1974	IRS-F-23	(German)
	Jan.-March	1975	IRS-F-25	(German)
	Apr.-June	1975	IRS-F-25	(German)
Annual report	Jan.-Dec.	1974	IRS-F-24	(English)

8. Degree of availability

Unrestricted distribution.

Classification: 1.1.2	
Title 1 (Original Language): Entwicklung eines Radionuklidmeßverfahrens zur Massenstrommessung in instationären Merhphasenströmungen (RS 146, PNS 4214 - I.1.1, Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Development of a Radionuclide Method of Mass Flow Measurement in Non-Steady State Multiphase Flows	**Project Leader:** R. Löffel
Initiated (Date): 1974 — **Completed (Date):** 1979 **Status:** continuing — **Last Updating (Date):** December 1975	

1. General Aim

A radionuclide technique is presently developed which allows to determine the mass flow of non-steady-state two-phase flows. This is done by measurements of the

- velocity of the gas phase) using a radiotracer technique,
- velocity of the liquid phase)
- density of the two-phase mixture, using a gamma-absorption method.

It is intended to measure simultaneously although separately the velocities of the two phase by means of two different radiotracers. Moreover, the method will be combined with an absorption density measurement. The method is to attain a very high time resolution (better than 100 msec) and shall be also applicable above all at pressures between 1 and 160 bar and temperatures from 20 to 350 °C. Besides, efforts must be made to keep the measuring sections as short as possible so that the method can also be used for short tubes (e.g. the rupture pipes of the Groß-Welzheim Superheat Reactor HDR).

2. Particular Objectives

2.1 Method of Radiotracer Velocity Measurement

Based on the familiar time-of-flight method a measuring technique is being developed which is suitable for studying steady-state and non-steady-state two-phase flows also in tubes of larger dimensions. Having passed an initial section, the radioactive tracer injected into the flow is recorded by an activity distribution plot at two measuring points placed in a staggered arrangement along the tube. the flow rate is determined from the distance between the two measuring points and the time-of-flight ass[illegible] of the radiotracer. Periodic injection allows also a quasi-continuous observation of the non-steady-state flows. The short-lived radionuclides Ar-41 (gas phase) and Mn-56 (liquid phase) are used to mark the two gas/liquid phases.

2.2 Gamma Absorption Density Measurement Technique

The gamma-absorption density measuring technique is coupled with the measurement technique for determination of radiotracer velocities such that the density and velocity are measured with the same accuracy and time resolution. Since both direct and scattered radiations are emitted from the Ar-41 and Mn-56 radiotracers injected into the flow, the scattered radiation must be eleminated in the energy range of the gamma-absorption density measurement selected to allow proper measurement of densities.

3.1 Experimental Facilities

A composition of a 12-detector velocity measuring system has been completed. A water-air test facility was built up and commissioned in mid-June. For testing the injection techniques for the liquid and gaseous phase in the range of blowdowns a high pressure autoclave facility of GfK-IRB has been used as well as the RS 50 containment facility of the Battelle Institut during the C7 blowdown experiment.

3.2 Research Program

Measurements of gas velocity in the blowdown channel during the MARVIKEN MX-II CRT experiments. Testing the radionuclide method by measuring gas- and water velocities, respectively, in the MARVIKEN top pipe. Preparation of the method for the ISPRA blowdown experiments, and for the HDR-experiments.

4. Project Status

4.1 Progress to Date

Preliminary experiments on a high pressure autoclave facility have been started. Tracing and injection techniques should be studied for the liquid and gas phases in the range of high pressures and temperatures. Additionly, the injection system was tested on the RS 50 containment facility of the Battelle Institute during the C7 blowdown test of March 26.

4.2 Essential Results

With a view to develop and test the radionuclide method of two-phase mass flow measurements a water-air test section was set up and progress has already been made regarding the separation of energies and time resolution, viz.,

- perfect recording of the Ar-41 and Mn-56 radiotracers;
- decisive improvement of time resolution of the method from 1 sec to 20 msec by increasing the injection frequency from 1 Hz to 50 Hz for liquid and gas injections.

5. Next Steps

In 1976 the method will undergo final testing under conditions resembling blowdown. The tests will be performed at the "Joint Test Bench for Testing and Calibrating Differenz Two-phase Mass Flow Measuring Techniques" (PNS 4215, RS 145). Further applications of the radionuclide method are planned for the MARVIKEN II experiments in 1976 and the HDR blowdown experiments from 1977 until 1980.

6. Relation with Other Projects

RS 33	Joint Reactor Safety Experiments in the Power Station of Marviken, Sweden GKSS, Geesthacht, 1971 - 1977
RS 50	Investigation of the Phenomena Occurring within a Multicomparted Containment after Rupture of the Primary Cooling Circuit in Water-Cooled Reactors Battelle-Institut, Frankfurt, 1971 - 1976
RS 145 (PNS 4215)	Joint Test Rig for Tests and Calibration of Different Methods of Two-Phase Mass Flow Measurement GfK-IRB, Karlsruhe, 1974 - 1976

7. Reference Documents

Report KFK 1859 (1973) (German)
Report KFK 2050 (1974) (German)
Report KFK 2130 (1975) (German)
Report KFK 2195 (1975) (German)

Reports in the IRS-FORSCHUNGSBERICHTE series

Quarterly reports:

Report period	Jan. - March 1975	IRS-F-25 (German)
	April - June 1975	IRS-F-26 (German)

Semiannual report:

Report period	July - Dec. 1975	IRS-F-28 (German)

8. Degree of Availability

Unrestricted distribution.

Classification: 1.1.2		
Title 1 (Original Language): Weiterentwicklung eines Drag-body für die Massenstrommessung bei Blowdown-Untersuchungen des Vorhabens RS 109 (RS 147 - I.1.1., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Battelle-Inst. Frankfurt/M.
Title 2 (english): Improvement of a Drag Body for the Mass Flow Measurement on Blowdown Experiments of the Research Project RS 109		**Project Leader:** G. Hampel
Initiated (Date): September 15, 1974	**Completed (Date):** May 31, 1976	
Status: continuing	**Last Updating (Date):** December 1975	

1. General Aim

For measuring the discharge rate of a transient two-phase flow, a measuring method has been developed at Battelle which combines momentum measurement (ρw^2) using a drag body and density measurement (ρ). This mass flow measuring method has already been used in the blowdown experiments of the research projects RS 16/2 and RS 50.

The RS 147 project is aimed in particular at optimising the design of the drag body used for the mass flow measurements within research project RS 109, i.e., at adapting it to the specific conditions of these experiments in which the effect of the PWR loops on blowdown are to be investigated.

2. Particular Objectives

In addition to the optimisation of the design data for the drag body with regard to measuring range and dynamic behaviour, an estimate is to be made of the range of application of the drag body as a mean for measuring the momentum in an unsteady two-phase flow. These investigations will include

- a theoretical determination of the magnitude of the drag coefficient characteristic of the drag body in

-- an accelerated incompressible and
-- a (steady) compressible

flow medium, and

- an analysis of the effect of the drag body on the thermodynamic state of flow.

3.1. Experimental Facilities

A laboratory-scale test apparatus was built for investigating the transient behaviour of the drag body. The firmly clamped drag body is deflected by means of an eccenter and then suddenly released. The signal of the strain gauges is stored on an oscillograph. The damping coefficient of the drag body can be determined from the curves of the measured values.

3.2. Research Program

The transient behaviour of the drag body as a function of the deflection was determined in water and in air.

4. Project Status

4.1. Progress to Date

a) The differential equation which describes the oscillation behaviour of a drag body was programmed so that the optimum design data of a drag body with respect to measurement range, sensitivity and frequency response characteristics can be calculated by means of a digital computer. The modular programme system consists of several modules and allows rough calculations of the design data taking into account geometry and flow conditions, as well as calculations and plottings of the eigenfrequency, the vibrational mode the transient process for a defined deflection, the extension of the strain gauges and the frequency response curve.

   The transient behaviour of an existing drag body (which is being used in other research projects) for different initial deflections in water and air was investigated with the above described laboratory test apparatus. The damping coefficient was determined.

Calculations were made for two drag bodies to be used in the two experimental loops of project RS 109 with a nominal diameter of 45 mm (ND 45) and 75 mm (ND 75) respectively.

Construction of one drag body was completed.

b) The drag coefficient of a flat plate - the drag body - exposed to incompressible unsteady flow normal to the plate was determined theoretically using a modified computational model based on potential flow theory. This model takes into account the greater separation velocity (relative to the free stream velocity) at the edges of the plate and the variable shape of the wake past the plate associated with the velocity.

c) The analysis of the effect of the drag body on the thermodynamic state of the flow, and the theoretical determination of the drag coefficient of the drag body in compressible steady flow have been started.

4.2. Essential Results

a) A comparison of the calculations with the transient oscillation experiments shows that calculating the drag body, one can assume as a first approximation that the damping effect is proportional to the flow velocity. A variation in the damping coefficient by a factor of ten has no substantial influence on the frequency response curve in the frequency range of interest (0-1 kHz for the drag body for ND 45).

The frequency range of the drag body for ND 45 shown in Fig. 1, that of the drag body for ND 75 in Fig. 2.

b) The drag coefficient for an accelerated flow against the plate is composed of the value for steady flow and an unsteady component. The difference from the steady flow value grows with increasing acceleration and decreasing velocity. Example: With a flow velocity of 10 to 100 m/s and an acceleration of 100 to 1000 m/s^2 (typical conditions for the inital phase of blow down), the drag coefficient for accelerated flow deviates from that for steady flow by less than 1 %.

c) It was found, based on a literature survey, that the drag coefficient must be expected to rise in compressible flow if the flow velocity reaches the velocity of sound. The increase in the drag coefficient depends on the computational model chosen for calculating the sound velocity of the two-phase flow. The density variations in the flow around the drag body must be taken into account.

5. Next Steps

One model each of the drag body versions for ND 45 and ND 75 will be built.

The theoretical investigations will be continued.

A subsequent agreement provides for an extension of the RS-project the development of an additional drag body for use in the lower plenum of the experimental facility of RS 109 and of another one for bidirectional measurements.

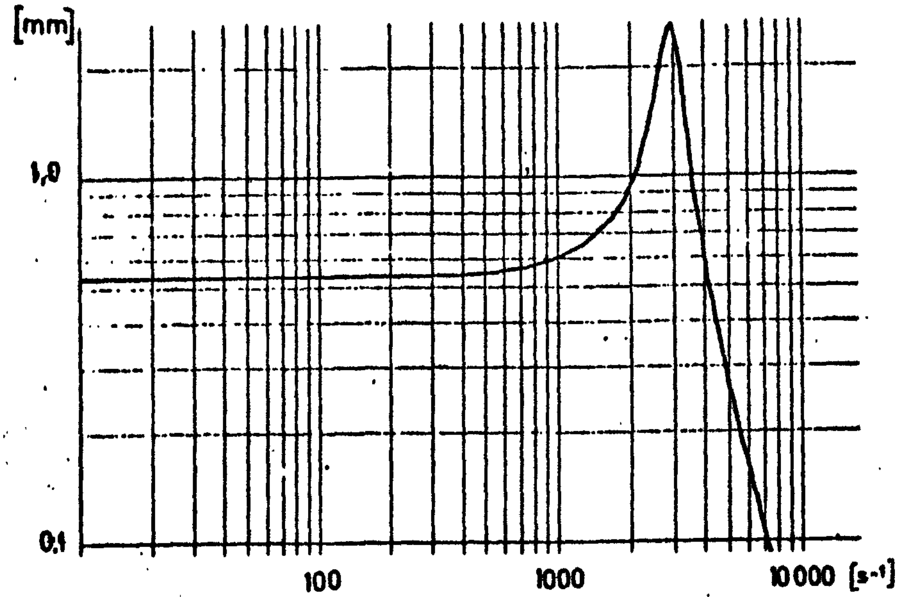


Fig. 1: Frequency response curve of the drag body for ND 45 (deflection at the end of the plate)

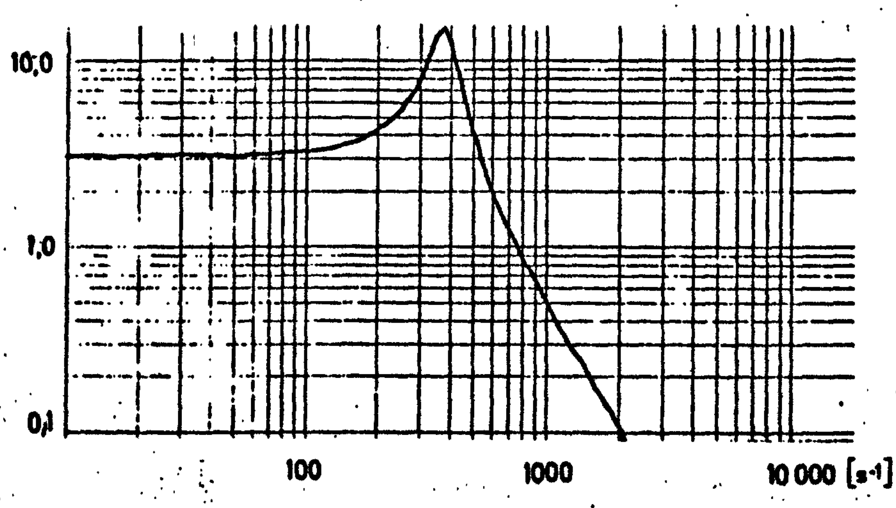


Fig. 2: Frequency response curve of the drag body for ND 75 (deflection at the end of the plate)

Classification: 1.1.2		
Title 1 (Original Language): Phasenseparation (RS 179 - I.1.1., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: TU Hannover
Title 2 (english): Phaseseparation		Project Leader: Prof. Dr.- Ing. F. Mayinger
Initiated (Date): 1975	Completed (Date): 1978	
Status: continuing	Last Updating (Date): December 1975	

1. General aim

Because of the great uncertainty in giving a detailed prediction of flow pattern void fraction and phase velocity of liquid and steam during blowdown conditions in the pressure vessel and in the core it is necessary to investigate the phase separation in a two-phase flow. This is valid especially at the end of blowdown and for the reflood and refilling period. Special interest has to be given to the mutual influence of gas and liquid phase in void fraction ranges of about 0,4 to 0,6, since there exist only a few empirical models which have to be adapted to the original conditions by use of a number of empirical parameters. For the intended investigations reported here the general aim is to develop a theoretical phase-separation-model which is able to describe the hydrodynamic conditions in the reactor pressure vessel for most of the expected flow regimes. Therefore it is necessary to investigate the behaviour of slip ratio depending upon quality, mass flow rate, phase velocity of steam and liquid and pressure. Experimental analysis has to be done with different geometries. These experiments are performed with the model fluid R 12 (CF_2Cl_2).

2. Particular objectives

The main problems are the investigations of mass discharge out of the reactor vessel during LOCA, the liquid level settlement and the separating mechanisms of liquid and steam in a two-phase mixture. These investigations have to be performed for two different geometrical conditions such as an pressure vessel firstly without core structure and a free liquid surface and secondly including a rod bundle simulating the reactor core. Two different separation-models have to be developed considering the different geometries. Additional to these investigations Freon-water scaling laws have to evaluated because of the model fluid R 12 (CF_2Cl_2)used.

3.1 Experimental facilities

Within the experimental part of these investigations phase separation in a two-phase mixture and droplet entrainment from a liquid surface have to be determined by an optical measuring method. With aid of the high speed kinematographie fluid behaviour in the pressure vessel can be investigated directly. So it is possible to get detailed informations about droplet entrainment by counting and planimetering the recorded droplets from the pictures. From a single photo liquid hold up can be got. By comparing two following high speed photos it is possible to calculate droplet velocity. Additionally an X-ray equipment and a gamma ray-source will be used to determine the void fraction. Furthermore to determine the mass flow rate out of the pressure vessel a so-called True Mass Flow Meter /1/ is planned to be placed behind the outlet pipe.

For realizing the provided investigations it is necessary to construct two testing devices: a small test section where the measuring techniques can be tested and a bigger one which reproduces geometrical reactor conditions in a better way. The two test sections differ only in their size. The small one is a 28:1 and the bigger one a 7:1 scaled-down version of a reactor pressure vessel.

The loops feeding the test sections consist in their main parts of an evaporator, an superheater, a steam injector, a condensor, a pressurizer and a pump. Additionally a junction from the outlet of the test section to a cold trap has to be provided to simulate blowdown conditions.

3.2 Research program

After successful testing the kinematographic methods and techniques at the small loop first experiments will be conducted. Based on the received results a second test circuit will be designed in a bigger manner to give better agreement to reactor conditions. For these investigations all essential conditions can be realized even at the small loop, which are developing a residual liquid volume in the reactor vessel and also conditions which are necessary to simulate liquid entrainment from the upper grid plate during spray cooling. In the bigger one small flow rates will be simulated instead of the quiet liquid surface in the small one. Furthermore a

4-rod-bundle representing a section of the reactor core will be installed to investigate water discharge out of a ring slot, and phase separation during reflood and refilling period.
The test program for the investigations of phase-separation, mass discharge and liquid level settlement will be conducted by a systematical variation of those conditions which are expected to have a strong influence.

Variied conditions resp. parameters:
quiet liquid surface - small changed flow rates
heated or adiabatic conditions
injected void rate
initial liquid volume in the vessel
initial subcooling of liquid - initial overheating of steam
saturation pressure
steady state - blowdown conditions

4. Project status

4.1 Progress to date

Within this report period starting in August 1975 the following investigations were conducted.

1. Studies of the relevant literature
2. Design and assemblage of the small test facility
3. Theoretical investigations and comparison of up to date published phase separation models.

For testing the measurement techniques and to start with first steady experiments the small 28:1 scaled-down version of the pressure vessel has been designed. Because of the used modelling fluid Freon 12 system pressure simulating reactor conditions can be kept below 10 bar. So it was possible to use a glass tube for the simulated reactor vessel, to allow a visuel observation of the phase interface and the fluid behaviour in the whole pressure vessel. The loop consists of the equipment described in 3.1. The arrangement of these components is shown in fig. 1 placed below.

Besides constructing the experimental circuit, the appropriate literature was examined of available phase separation models. There are four possible models distinguishable to their region of validity.

- Totally phase separation, i.e. only saturated steam rises above the phase interface because of pressure release. This model may only realize small leaks with small pressure charges versus time.

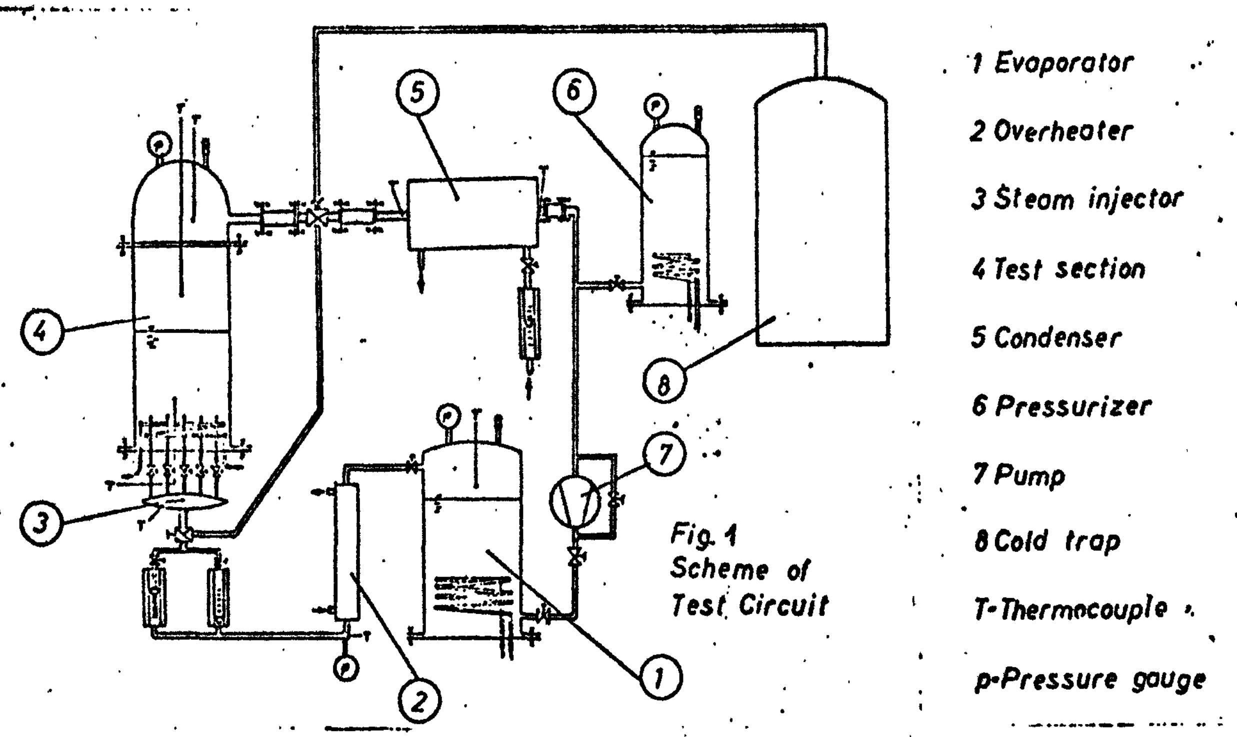


Fig. 1
Scheme of
Test Circuit

- A homogeneous model as critical case of phase separation described by W.H. Rettig at al /2/.

  At sudden release of pressure the whole volume starts to flash and a homogeneous mixture may be formed filling up the whole pressure vessel. This model may be relevant for a 2 A-break in the primary circuit with expected great pressure gradients.
- Phase separation following the bubble rising model by J.C. Wilson et al /3/. Void fraction ε at the interface is calculated as a function of the terminal velocity of steam bubbles rising through saturated water and void velocity above the interface following the equation:

$$\varepsilon \cdot V_B \cdot A = V_S A$$

V_B = terminal velocity of the bubble
V_S = velocity of steam above the two-phase interface
A = total liquid surface area

By this equation void flow at the interface is determinable. Measuring the mass flow leaving the pressure vessel and local quality, phase separation at the interface and in the upper plenum can be recalculated.
- A semi-empirical phase separation model used in RELAP-3 described by W.H. Rettig et al /2/. The quantities necessary to describe phase separation are a measured

local void velocity and the partial density of steam bubbles in the whole two-phase mixture. The shape of that partial density curve is approximated to increase linearly within height of the mixture by use of an empirical parameter.

Connecting a constant bubble rising velocity given by experimental results and the partial density curve, void flow rate contained in the mixture can be got from a differential equation considering post-evaporation of entrained droplets from results of outlet mass flow.

Furthermore by knowledge of the partial density at the interface a local void flow rate is determinable.

4.2 Results

The work was started in August 1975. First results are expected in March 1976.

5. Next steps

The test-loop will be set to work to test the measuring techniques and to perform first steady-state experiments.

6. Relation with other projects

RS 48, RS 48/1

Freon-water scaling laws (T.U. Hannover)

RS 163

Theoretical and experimental investigations on thermohydraulic behaviour in reactor cores during the first blowdown phase

RS 147

Two-Phase Measurements - Drag body - (Battelle Institute Frankfurt)

7. Reference documents

/1/ Class, G.: True Mass Flow Meter (TMFM) - Entwicklung von Verfahren zur Massenstrommessung instationärer Zweiphasenströmung, Reaktortagung, Nürnberg 1975, paper no. 347

/2/ Rettig, W.H. et al: Relap-3 A Computer program for reactor blowdown analysis IN 1321 June 1970

/3/ Wilson, John F., Grenda, Ronald J., Patterson, John F.: The velocity of rising steam in a bubbling two-phase mixture, ANS Transactions, Vol. 5, No. 1, Pg. 151 (1962)

/4/ F.J. Moody: Maximum two-phase vessel blowdown from pipes, Journal of Heat Transfer, August 1966

/5/ F.J. Moody: Maximum flowrate of a single component two-phase mixture; Journal of Heat Transfer, Feb. 1965, Transactions of the ASME
/6/ Quarterly reports in the series IRS-Forschungsberichte

8. Degree of availability

The IRS-Forschungsberichte are available by IRS.

Classification: 1.1.2	
Title 1 (Original Language): Untersuchungen über das Verhalten von Hauptkühlmittelpumpen bei Kühlmittelverluststörfällen Phase A (RS 111 - I.1.1, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Investigations of the Behaviour of Main Coolant Pumps under MCA Conditions (Phase A)	Project Leader: Dr. Riedle
Initiated (Date): September 1974 — Completed (Date): December 1976 Status: Continuing — Last Updating (Date): December 1975	

General Aim

During blowdown the flowrate through the core and the temperature of the fuel rods depend on the behaviour of the main coolant pumps. The behaviour of the pumps under two phase flow conditions will be studied, in order to improve the theoretical and experimental knowledge and to dismantle rather conservative assumptions, which are used in the safety analysis.

Particular Objectives

Using the experimental results of pump tests under simulated MCA conditions, the physical models of the pump behaviour will be improved in order to replace the existing assumptions of the blowdown calculations.

Experimental Facilities / Research Program

Two model pumps will be built in the scale 1 : 4 and 1 : 5 of the main coolant pumps of GKN. The singlephase characteristic will be measured by the manufacturer.

A testloop at C-E will be rebuilt in order to measure the interesting parameter variations of pressure, flowrate and vapour content. After this, the two model pumps will be installed and about 390 stationary points will be recorded in the twophase region. Finally with 10 blowdown tests the extent of the results on the transient blowdown process will be checked.

Project Status / Progress to Date

For the transient tests precalculations were conducted with the C-E code FLASH; the proper loop operation under steep gradients was assured. The data recording and evaluation was planned.

The instrumentation had to be improved: transducers, drag discs and turbine flow meters were installed. First loop-tests were run at 30 bar pressure. Slug flow conditions were observed, which will be investigated. Other problems were associated with the control of some components.

A small testloop was built, which simulates well defined, quickly changing conditions in order to test the time behaviour of the instrumentation.

The model pump manufacturer has designed details of the pump casing and bearings. The auxilary systems for the oil and seal water provision have been worked out.

Project Status / Essential Results

The blowdown calculations have shown, that the test loop is well suited for simulation of the transients, which allow extent of the stationary results on transient reactor MCA-conditions.

The testloop is ready for use, the necessary mass steam for stationary tests can be reached. The instrumentation has been tested successfully.

Next Steps

First tests will be conducted during the first quarter 1976.

Relation with Other Projects

See RS 111 A 74.

Reference Documents / Degree of Availability

No reports available.

Classification: 1.1.2	
Title 1 (Original Language): Einfluß der DWR-Umwälzschleifen auf den Blowdown (RS 1o9-I.1.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Euratom, Ispra
Title 2 (english): LOBI (Loop Blowdown Investigation), Influence of PWR Primary Loops on Blowdown	Project Leader: Riebold
Initiated (Date): January 1974 Status: continuing	Completed (Date): December 1977 Last Updating (Date): December 1975

1. General aim

Design and construction of a large scale blowdown loop system.

Performance of loss of coolant experiments by simulating tube ruptures within a PWR primary cooling circuit system.

2. Particular objectives

Experimental investigation of the role of the diverse components of a PWR primary cooling circuit system during a blowdown by measuring the main thermal hydraulic quantities which are influencing the core cooling, in particular the fluid flow situation, the heat transfer and the pressure differences.
The experimental results will be applied for the checking an improving of the blowdown computer codes and associated theories used for the safety analysis of LWRs.

3.1 Experimental facilities

A 4-loop primary cooling circuit system of a 13oo MWe PWR reference plant is simulated by a 2-loop experimental system, one loop representing three intact "reactor" loops and the other one representing the broken "reactor" loop. Both experimental loops are active loops containing pumps and steam generators.
Tube ruptures of various rupture sections are to be simulated at three different positions within the broken loop.

The scaling factor of 712 for power, mass flow and volume led to 5 MW heating power input to a 64 heater rod bundle as reactor core simulator.

The loop system and component design has been done for 16o bar and 325° C operating pressure and temperature respectively, maintaining

- the power to volume ratio for the size reduction
- the pressure drop and fluid temperature distribution along the flow paths
- the volume ratios among, and the elevations of the components
- the lengths of the heat transfer surfaces (core rod bundle, steam generator)

equal to the corresponding reactor values.

Two different accumulators (6o bar and 3o° C operating pressure and temperature respectively) of a 1 : 3 volume content for the two loops, are providing ECC water for both separate and combined cold leg and hot leg injection into both loops.

3.2 Research programme

Two different experimental programmes are to be performed with this test facility:

Programme A, defined by the BMFT-Bonn and to be performed in the framework for the R&D contract RS-1o9/143-73-PIHOD, concluded between the BMFT-Bonn and the C.E.C., will be concerned with the investigation of the influence on the blowdown of

- rupture size and position
- pumps performances
- initial power level
- power-time function of the heat input
- strength of heat sink (steam generator secondary side)
- downcomer resistance and volume
- ECC water injection positions.

Programme B, still to be defined by the C.E.C. and to be executed after conclusion of programme A, will be concerned with the

- performance of component studies, to be done with this test rig, after modification of certain components, and aiming at the investigation of the influence of the geometrical shape (volume) and/or the elevation of these components on blowdown

- performance of some reference tests (repeated tests of programme A) holding at the same time as reproducibility tests.

The following seven component modifications have been chosen up to now:

- variation of loop seal depth (U-tube between steam generator and pump) within the intact loop
- variation of the steam generator elevation within the intact loop
- variation of the lower plenum volume (higher L/d ratio)
- two separate accumulators, one for each loop, instead of one accumulator for both loops
- primary U-tube rupture within the steam generator of the broken loop
- small rupture within the lower plenum
- ECC water injection into the upper plenum

4. Project status

During 1975 the project activities mainly delt with works of the last part of phase II and the first part of phase III of the project planning:
After an extended revision of the preliminar project a few orders could already be placed during 1974 especially the order of the pumps having the greatest delivery time of all components and determining therefore the critical path of the time planning. Most of the orders for the various test facility parts have been placed during 1975; these were, in chronological sequence, the orders for the mechanical loop components including the scaffolding, a time series analyzer for the signal evaluation programs, the 5,5 MW rectifier plant, the reactor-model consisting of the pressure vessel with internals and of three heater rod bundles, the signal processing and data acquisition system, a first part of the measuring devices, the building with two rooms for housing the loop regulation and control instruments and panel and the signal processing and data acquisition system, the foundations for the loop system and the rectifier system, and the concrete bunkers for the containment simulation. Only a few orders have remained for being placed during the first months of 1976; they are con-

cerned with the rest of the measuring devices having rather short delivery times, and with some further auxiliary materials. So, during 1975, there has been a large overlapping of the placing of orders (phase II) and the construction of the last facility components (phase III).
The rather comprehensive administrative work, required previous to placing orders and caused by heavy procedures, unfortunately has been incombent on the technical project staff, too.

4.1, 4.2 Progress to date and essential results
These activities have been accompanied by technical and theoretioal works concerned with further and final details of the design of the various systems and components of the test facility, which then finally has been agreed with the respective furnishers and manufacturers.

The experimental investigation of the solubility of different ceramic materials in water under operation pressure and temperature conditions showed, that 99,7 % Al_2O_3 ceramic is the best suited material to be used for fillers and insulators within the reactor-model. Thermal shock test with annuli of the same material revealed, that only open annuli could be applied for the downcomer gap-widths variation by slipping them over the core barrel tube.
Several absolute and differential pressure transducers, made available to us from different manufacturers, have been tested under operation conditions in order to examine their steady-state performances, especially with respect to the measuring precision and to the overload capabilities.
The experimental investigation of the flow resistance distribution within the downcomer region of the reactor pressure vessel, performed with a reactor mock-up by the KWU-Erlangen in the framework of a subcontract from Ispra, has been concluded; the results have been evaluated for the downcomer design of the LOBI facility reactor-model.

A special small-scale facility has been constructed for testing and calibrating the pressure and temperature transducers in steady-state and transient conditions.

Preparations have been completed for an experimental investigation of the influence of the heater rod current and the magnetic fields on the temperature signals of the heater rod thermocouples; the test will be started in January 1976.
Theoretical work was mainly concerned with calculations supporting the various loop systems and components design and specification work.

Theoretical assessments showed that the influence of asymmetric flow towards the rupture section is negligible; the same special inserts for simulating 2-A (double-ended) ruptures can therefore also be used for simulating 1-A and smaller breaks.
RELAP4-code calculations have revealed that closing times of the insulating valve between the two break branches of 100 ms and less do not influence the blowdown time history during the first second.

The influence of stored wall-heat release on the fluid enthalpy-time history within the reactor-model has been determined to get a criterion for the material choice for the downcomer gap-width variation.

Special heater rod calculations, done by the LRA-Garching yielded the time function for the electrical heat input to the rod bundle during blowdown required for simulating decay heat and stored fuel-heat release. To smooth the power-time curves, optimisation calculations have been performed leading to some deviations from exactly binary-graded power subgroups of the 5 MW rectifier system.

Blowdown calculations with the BRUCH-D-code (done by LRA-Garching for the reference reactor) and with RELAP4-code (done by the project staff for the LOBI facility) yielded the variation range and the time gradients of the different thermalhydraulic quantities to be measured and recorded at several positions over the loop system during the blowdown. These informations were required for specifying the appropriate performances of the various measuring devices with respect to their measuring range and dynamic behaviour (response time).
Theoretical considerations revealed the rather strong influence of the fluid velocity and density profiles on the readings of turbine flow meters and hence on the volume flow rate measurements.

From BERSAFE-code calculations the stresses have been obtained, which are to be expected due to thermal and mechanical loads in the upper power connecting plate of the reactor-model; they do not exceed the admissible limit of 800 kp/cm^2.
By STRUDL-II-code calculations the magnitude and direction of displacements at different loop positions have been determined which will occur due to thermal dilatation only. Preparative work is actually under way for similar calculations accounting for combined thermal and mechanical loads.
From DAPSY-code calculations by the LRA-Garching the mechanical loads on the pressure vessel internals during blowdown have been obtained; the time history of the pressure increase within the concrete bunkers simulating the containment during blowdown has been determined by ZOCO-code calculations at the same institute.
Several modifications have been applied to the RELAP4-code for rendering it more flexible to various user needs. The most essential ones are concerned with the change to metrical units for the input and output data, with the improvement of the plot program and with a supplementary program allowing the redimension of all variables in the 64 COMMON blocks.

5. Next steps

Completion of construction and starting of mounting works of the test facility:
Mounting of loop and rectifier system after completion of the foundations.
Installation of data acquisition and loop regulation & control system in the computer and control rooms. Commissioning of the test facility.

6. Relation with other projects

RS 37/1:	Blowdown Heat Transfer and Evaluation
RS 36/1/2/3/A:	Reflood Heat Transfer and Evaluation
RS 36 B:	Reflood Thermohydraulics
RS 77:	Thermodynamic Non-Equilibrium
RS 81:	Boiling Mixing
RS 163:	Blowdown Thermohydraulics with Freon

RS 179:	Phase separation
RS 111:	Pump behaviour
RS 144:	Pump behaviour for RS-1o9 pumps Semiscale Blowdown and ECC Project (ANC, Idaho Falls, USA)

Two-Phase Measurement Methods:

RS 135:	Temperature Noise Correlation
RS 136:	NMR-Study
RS 145:	Joint Test Rig
RS 146:	Radiotracer Method
RS 147:	Drag-body development
AtT o85 A:	Analysis Development
PNS 423o:	Fuel Behaviour

7. Reference Documents

Quarterly Reports of 1974 and 1975, IRS-F-2o-28

8. Degree of availability

All references are available from IRS-Köln, Glockengasse 2, 5 Köln 1.

Classification: 1.1.2.

Title 1 (Original Language): GKSS-Pumpenuntersuchungen (RS 144 - I.1.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GKSS, Geesthacht
Title 2 (english): GKSS-Pumptests for the Primary Pumps of RS 109	Project Leader: Katsaounis
Initiated (Date): ..11.1974 — Completed (Date): 1977/78 Status: continuing — Last Updating (Date): December 1975	

1. General aim

The aim of these tests is to take up the complete working diagramm for single- and two-phase flow of the primary pumps for the blowdown facility, which is projected by EURATOM in ISPRA (see project RS 109). The pumps have a great influence on the blowdown process. Dependent on the position of rupture in loop, any operation point at I, II and IV quadrant of pumps-characteristics by positive or negative speed is possible. Usually the pumps-characteristics are well known only by single-phase flow in I-quadrant. The behavior of pumps by two-phase flow ($0 \leq X \leq 1,0$), different speeds ($0,2 \leq n/n_N \leq 1,2$), and by different operation pressures is until this day totally unknown. The influence of the pumps on the blowdown process can be estimated by experiment only.

2. Particular objectives

The following points are mentionend. The experiments for this project are unusual in pump tests, because

a) all operation-points of the testpump are analysed by operation at varying, positive and negative speed and with normal and reverse flow (i.e. working points in each of the possible quadrants of the working field),

b) the inlet flow of the pump is not subcooled and of one phase, but a saturated two-phase flow with several qualities.

3.1 Experimental facilities

Steam and water flow rates are seperately metered and then combined in a mixing device MK. The resultant slightly sub-cooled or net quality two -phase mixture passes a vertical inlet-tube of 1,5 m length and enteres the test-pump (NW 65). At the end of the inlet-tube on the suction-side of the test pump an experimental chamber ist installed. Here are measured all parameters of the experiment, the differential pressure across the pump, the absolute pressure and temperature. The aim of the pressurizer is to stabilize the pressure on the upstream side of the pump. The mixing chamber is constructed for low pressure losses.

According to the working conditions the test-pump increases or decreases the pressure of the steamwater mixture.

After the test-pump the mixture flows through an outlet tube of 1.5 m length (NW 65). Then the flow passes a condensor with a power of 4.6 MW, where the steam part of the mixture is condensed. A cooler after the condenser with a power of 300 kW has to subcool the water for the circulation pump because of its required NPSH.

The condensed part of the mixture after the condenser passes the pressurizer and enteres to a storage tank (VB).

The pressure in the pressurizer is maintained by a regulated electrical heating and an injection cooling.

The flow of water going to mixing chamber then passes the circulating pump UP. This pump is working if the test-pump is not able to compensate the pressure losses of the circuit.
The characteristics of the circulating pump are:

total head H_{up} = 245 m Fls
total flow $\dot{V}_{up}$ = 250 m^3/h
connect load N_{el} = 315 kW

After the circulating pump surplus water flows back regulated by a bypass to the cooler. The heat due to energy losses is also eliminated in this cooler and the required NPSH of the circulating pump is made ready.

The recirculating water flows are then seperately controlled and measured before entering a preheater. Here the water is preheated nearly to the saturation point.

The required steam flow comes from the pressure vessel of the "PVS" after passing through a superheater with a regulated connect load of 250 kW. The superheating of steam is nescessary on account of pressure losses in the orifice, valve etc. The pressure vessel has the function of a steam accumulator. There is installed an electric heating of 600 kW, in some test point there is needed a steam flow with a power out put of 8.5 MW.

3.2 Research program

The aim of the tests is to obtain the complete working diagram of the test pump for stationary conditions with varied flow, including the reverse. By that way experiments are represented by points in all possible quadrants of the so-called Q-H-diagram (the third is excepted because the pump is a radial one).

The tested field in Q-H-coordinates is limited by the conditions $-1H_N \leq H \leq +2H_N$

$$-\dot{V}_N \leq \dot{V} \leq +2{,}5\ \dot{V}_n,$$

and, concerning the speed: $0.2 \leq u \leq 1{,}2\ u_N$,

where N indices nominal working conditions. The quality x is varied in the limits $0 \leq x \leq 1$.

A test-series with sufficiently subcooled water is planned.

4.1 Progress to date

1) The flow-sheet is concluded. The set-up disposition of the pressurier is a problem. In the selected set-up disposition of the pressurizer the problem of instabilities obtained priority. A theoretical study on the dynamical behaved of the controlled system "mixing-chamber-pressurizer" is made of a esteemed industrial firm.

2) A report of our possibilities in testing the pump is terminated. It still has to be written.

3) A scedule of instrumentation for the normal operation conditions is finished.

4) To eliminate large fluctuations of pressure the steam /water-mixture is condensed inside the tubes, There are necessary many studies before ordering the condenser with a capacity of 4.6 MW. (The condenser is ordered by Fa. Gessner). The pressure losses of the first of all provided horizontal cooler have been too large. The now provided version is a non-circulating water cooler, a part of the pipe is integrated. The cooler is to be ordered.

The preheater was constructed alternatively with a heating windings for direct heating or heating elements for indirect heating.

5) The sketches of electrical supply give priority to the heating windings for directly heating. Several components of this plant are to be used for a general direct current electric plant.

6) The high technical resources for the control system is nescessary because the installed heating power of 600 kW in our pressure vessel, which is used for steam production, is less than the 8.5 MW needed for some experiments. The pressure vessel therefore is used as a steam-store. At the beginning of experiment the pressure will be 105 bar. During experiments with maximum required steam-flow the pressure of the pressure vessel decreases within 3 min. to 85 bar.
The special design features of the controlling valve is the range of regulating

7) The volume of the pressurizer and the volume of the remaining system is the same because of possible pressure gradients. The construction of the pressurizer is ready for inquiry.

8) The layout of pipes is constructed. Some special informations for the control system are given. Part lits are made for inquiry.

9) The great problem is the mixing-chamber. GKSS has no experiences in this field and the informations of the avaible literature is very small. Ther fore only the information and experiences of CE in USA can be used for construction. There will be built one mixing chamber with two replaceable insets for low and high steam qualities. The optimization of the replaceable insets has to be done during the time of beginning of operation.

4.2 Essential Results

Of the activities in the past year is the rough description of additional limits of the test program, resulting from:

1) extrapolated graph, as was calculated by the characteristic of the pump up to the value of about $\dot{V} \approx 1,2\ \dot{V}_n$,

2) approach to velocity of sound,

3) limitation of power available,

4) controllability of small vaporous and aqueous mass-flows.

Concerning:

1) This reason lies in the test model itself and does not mean a real limitation.
2) Because the loop at ISPRA has the same widths as ours the same limit is to be expected there.
3) By that, especially for low nominal pressures and for aqueous flow-test in a wide range cannot be performed. This limitation can be tolerated, as
   - α) tests with single-phase-aqueous flow already will have been performed by the furnisher
   - β) the slope of the graph in that region is not expected to change markedly,
   - γ) this limit with increasing nominal pressures is shifting to the border of the acquired test field.

4) Those limitations result from economic reasons and mark only a narrow region at both sides of the ordinate-axis. Every slight further minimization of this stripe would mean a multiplication of prices for regulating apparatus and is regarded senseless.

5. Next steps

1) Order of control-system. Preparation of special informations.
2) Order of subcooler, superheater, pressurizer, tubes and flanges.
3) Concluding the construction of the mxing chamber. Order of the mxing chamber with two replaceable insets.
4) Concluding the sketch of electrical supply.
   Inquiry and order of the components.
5) Fixing the final test points in the possible field of experiments.
6) Projecting the instrumentation for the research-program and order of the instrumentation.

6. Relation with other projects

The test of this project takes up the complete characteristics curves in single and two-phase flow at the primary pumps for the blowdown facility, which is projected by EURATOM in ISPRA (RS 109)

7. Reference documents

IRS annual report A 74	(english)
IRS quarterly report V 75/1	(german)
IRS quarterly report V 75/2	(german)
IRS quarterly report V 75/3	(german)
IRS quarterly report V 75/4	(german)

8. Degree of availability

These reports are available by GKSS with permission of the IRS.

Classification: 1.1.2	
Title 1 (Original Language): Untersuchung des thermohydraulischen Ungleichgewichts (RS 77 - I.1.1., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** CEC, Ispra
Title 2 (english): Investigation of the Thermodynamic Non-Equilibrium	**Project Leader:** G. Fritz, W. Riebold
Initiated (Date): .12.1972 — **Completed (Date):** 31.12.1975 **Status:** finished — **Last Updating (Date):** December 1975	

1. General aim

The aim of the studies is to provide experimental data for theoretical models describing the deviation from thermodynamic equilibrium of the water-vapour mixture in a primary PWR-circuit during a blowdown.

2. Particular objectives

The deviation from thermodynamic equilibrium between the phases caused by:

- a sudden expansion of water
- injection of cold water in vapour atmosphere

The time behaviour of pressure is observed as indicator for the deviation from thermodynamic equilibrium.

3. Experimental facilities and program

Fig. 1 in the last Annual Report gives an impression of the experimental device. By technical reasons the test-program had to be reduced to:

- 37 tests with flashing by sudden expansion. Parameters: Temperature and initial pressure step.
- 88 tests with cold water injection. Parameters: injection quantity, state of the vapour atmosphere.

4. Project status

The experiments are finished.

5. Next steps

Completation of the final report.

Point 6., 7., 8. see A 74

Classification: 1.1.2		
Title 1 (Original Language): Untersuchungen der stationären und instationären kritischen Heizflächenbelastung an Vielstabbündeln von Druck- und Siedewasserreaktoren mit Frigen als Modellflüssigkeit (RS 64 - II.1.3., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GKSS, Geesthacht
Title 2 (english): Investigation of the Steady and Transient Critical Heat Flux of Multi-Rod-Bundles for PWRs and BWRs with Freon as Model Fluid		Project Leader: Katsaounis
Initiated (Date): 1.9.1972	Completed (Date): 30.4.1978	
Status: continuing	Last Updating (Date): December 1975	

1. General aim

The aim of this research program is to conduct burnout experiments at steady state conditions and at massflow, power and pressure transients for PWR and BWR using freon as model fluid.

2. Particular objektives

The experimental program is divided into the following three parts, each part will be carried out for PWR and BWR:

I - Burnout experiments at steady state conditions to obtain reference values for transient experiments,

II - Burnout experiments (Burnout-time-delay) at massflow and or power transients

III - Burnout tests (time delay and heat transfer by filmboiling condition) during pressure transients (blow down). These experiments will be carried out in coordination and as supplement to the research program RS 37-

3.1 Experimetal facilities

The experiments of part I and II are carried out in a steady state Burnout-facility (Fig. 1). For the tests during pressure transients (blow down) the loop from fig.1 is reconstructed according to the facility for RS 37 research program (Fig.2). The extension and the conformity of the freon loop to the

facility for RS 37 is necessary for the equivalence of the experimental results between freon and water.
The calculation of the characteristics of the loop for blow down experiments was made using the computing program RELAP 3. At beginning of the blow down process the expansion valve S.1 will open in less then 50 m sec. The fluid flows through the valves E, which simulate the area of fracture, to the condensation vessel (Kühlfalle). This vessel has a temperature less then -70 °C. The valve H 1 regulate the massflow at the inlet of the testsection according to a given curve.
The experiment can be carried out for leaks in hot (Fig 2) an cold leg (Fig3) of reactor respectively.

3.2 Research program

For two different reactor types are provided two test sections for the core configuration incl. the spacers of PWR and BWR respectively. Each test section has 7 x 7 rods with axial and radial non uniform heat flux distributions.

The experiments have been started with the test section for PWR beginning with part I and II followed by BWR-bundle with essentially the same experimental program.

The research program is very extensive, so it is impossible to explain it completely here. In this program are inclu ted:

Mixingmeasurements, Burnout experiments at different radial power distribution, pressure conditions, massflow rates and inlet subcooling at steady state conditions, as well as at massflow-, power and pressure transients. The experiments at transient conditions will be carried out for different combinations of power massflow and pressure curves (dependent curves) according to the calculated reactor characteristics.

4. Project status

4.1 Progress to date

The project is approximately 10 months behind shedule. The main reason for the delay is caused by difficulties with the electrical insulation of the original reactor spacer for the test sections. The experiments type I and II had been started with the PWR-test section.

The loop for the blow down experiments is laid out, and its construction is finished (see Fig.2 and 3). While waiting for the granting of the increased casts from the BMFT some calculations using the RELAP 3 computing program had been made.

4.2 Essential results

a) Mixing experiments are measured for the PWR-test section using thermocouples, which are located at the outlet of the test section. Heated are only the 8 central rods of the test section. Fig.4 shows for two different powers the estimated subchannel temperatures. These mixing experiments are made under variation of the parameters pressure, massflow and inlet subcooling over a large range.

b) Burnout experiments at steady state condition using the PWR- test section have been started. Fig.5 and 6 show respectively for two different pressures and three inlet subcouled temperatures the measured burnout-values vs massflowrate. The position of the burnout accured at different places.

5. Next steps

a) Finishing the experiments type I and II with the PWR-test section
b) Carry-out the experiments type I and II with the BWR-test-section
c) Ordering the components for the blow down loop, and finishing the calculation of the loop characteristics and specially defining the controlcurves for the regulating valves, which are very essential for an automation of the loop.

6. Relations with other projects

see report IRS-F-24
RS 164
RS 176

7. Reference documents

/ 1 / Internal GKSS-Note 72 05 AT C-03 (German)
"Ergebnisse der Durchmischungsversuche (teilweise) DWR/Phase I/RS 64"

/ 2 / Int. GKSS-Note 72 05 AT C-04 (German)
"Ergebnisse der Durchmischungsversuche (teilweise) DWR/Phase I/RS 64"

/ 3 / Int. GKSS-Note 72 05 AT C-05 (German)
"Ergebnisse der stationären Referenzversuche (1.Abschnitt) DWR/Phase I/RS 64"

/ 4 / 3 d Annual report IRS-F-24 (English)
Quarterly report in the series IRS-Forschungsberichte

Jan. - March	75	IRS - F - 25
April- June	75	IRS - F - 26
July - Sept.	75	IRS - F - 27
Oct. - Dec.	75	IRS - F - 28

8. Degree of availability

All reports are available with the allowance of IRS department "Forschungsbetreuung"

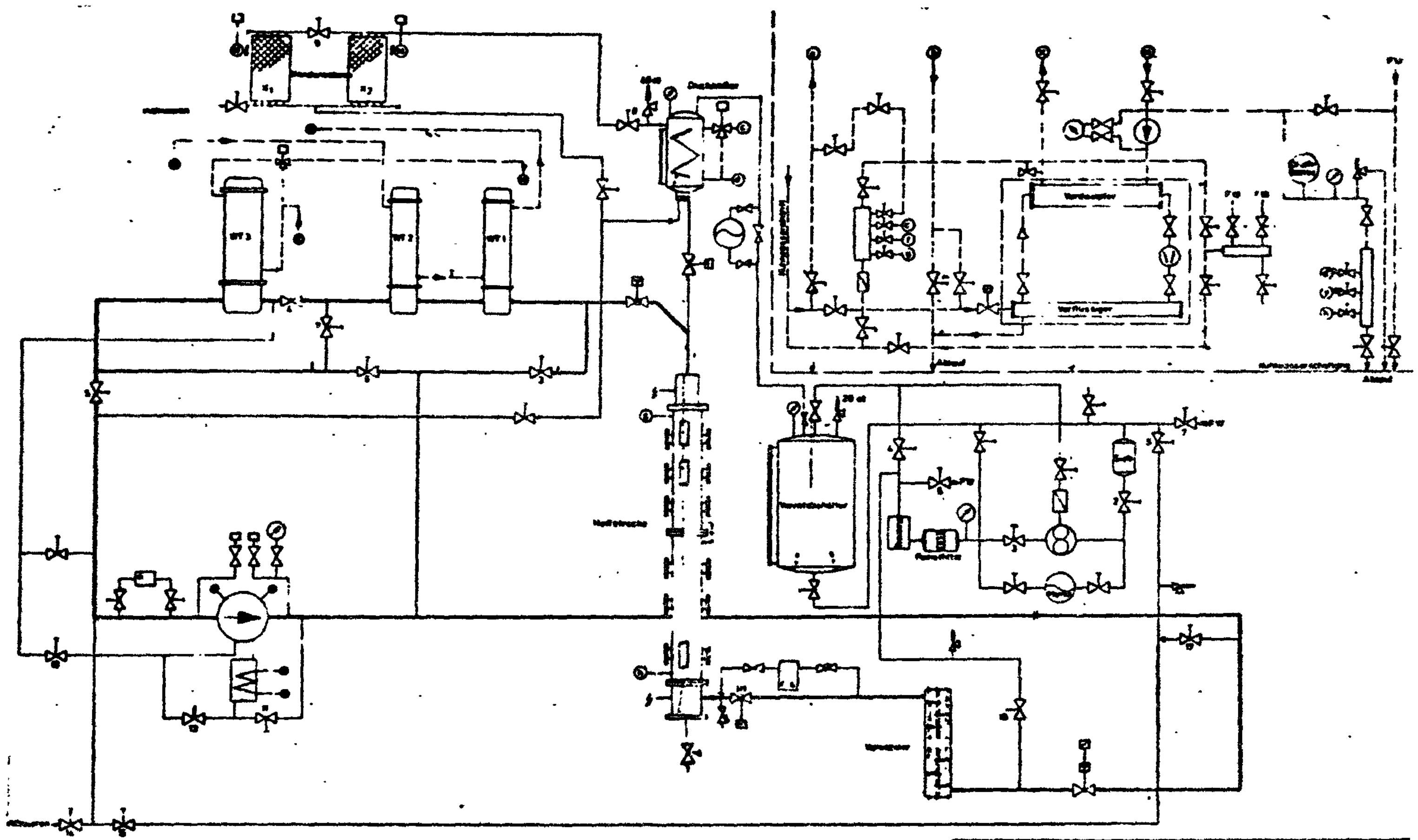


Fig. 1: Freon - Testfacility
shedule of the loop for steady state,
Massflow- and Powertransient experiments
for PWR and BWR

Frigen - Versuchsstand
Ausbaustufe SWR-DWR Reaktor
KERNENERGIEVERWERTUNG
C-2 3-264

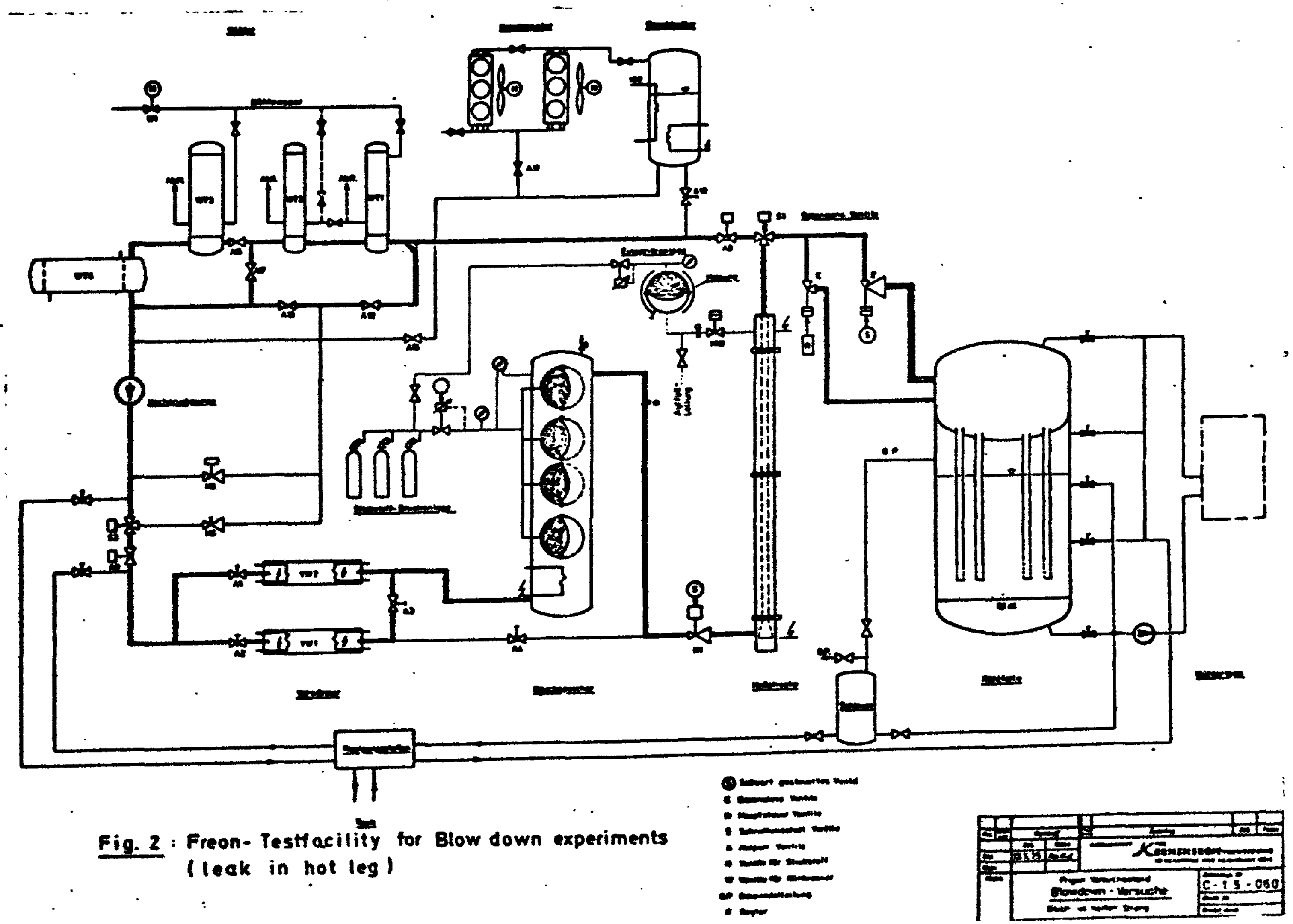


Fig. 2 : Freon- Testfacility for Blow down experiments
(leak in hot leg)

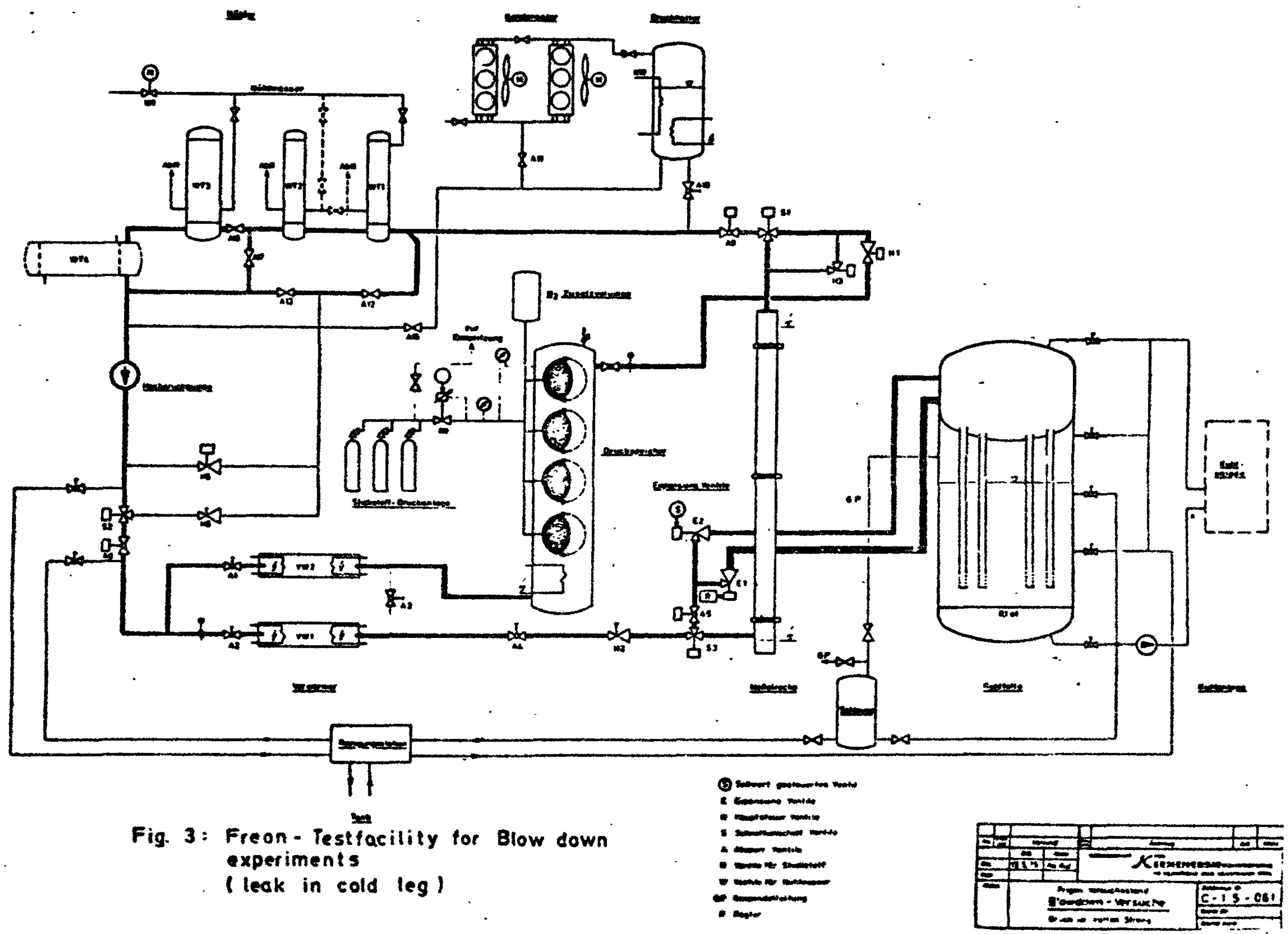
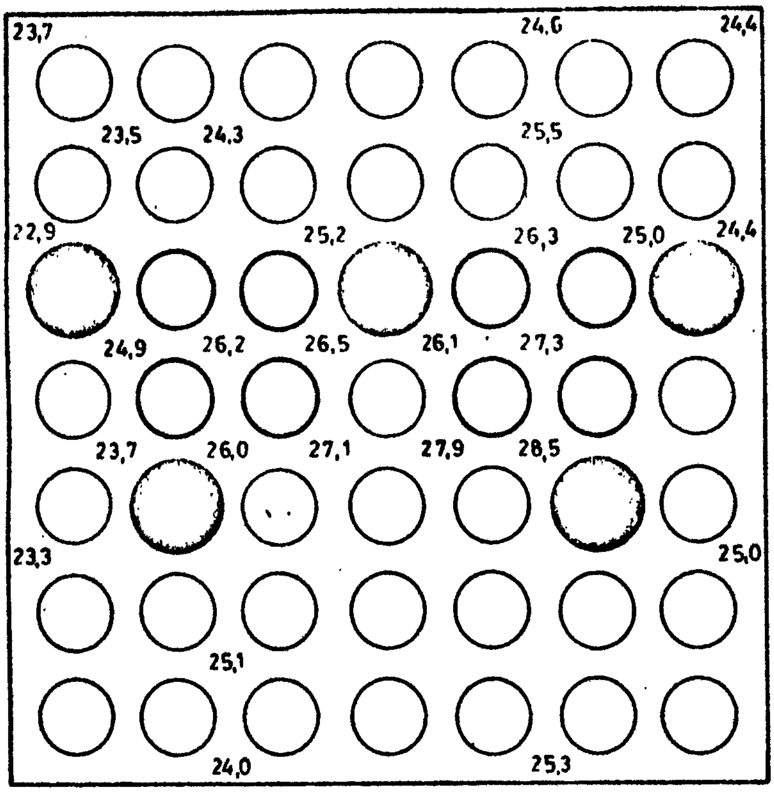
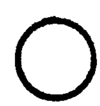


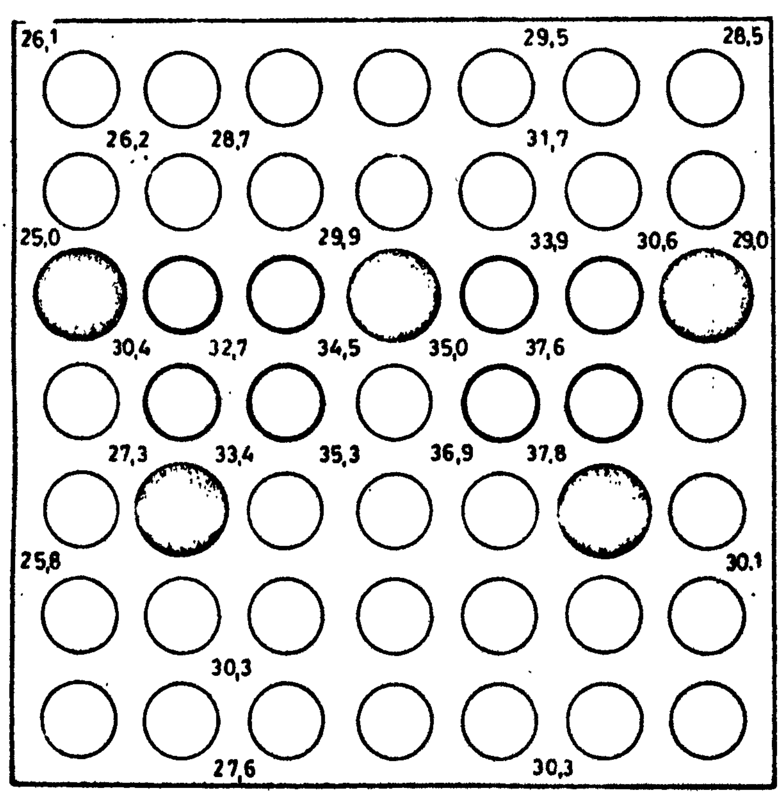
Fig. 3: Freon - Testfacility for Blow down experiments
(leak in cold leg)

PICT. 4 SUBCHANNEL EXIT TEMPERATURES (DEG C)

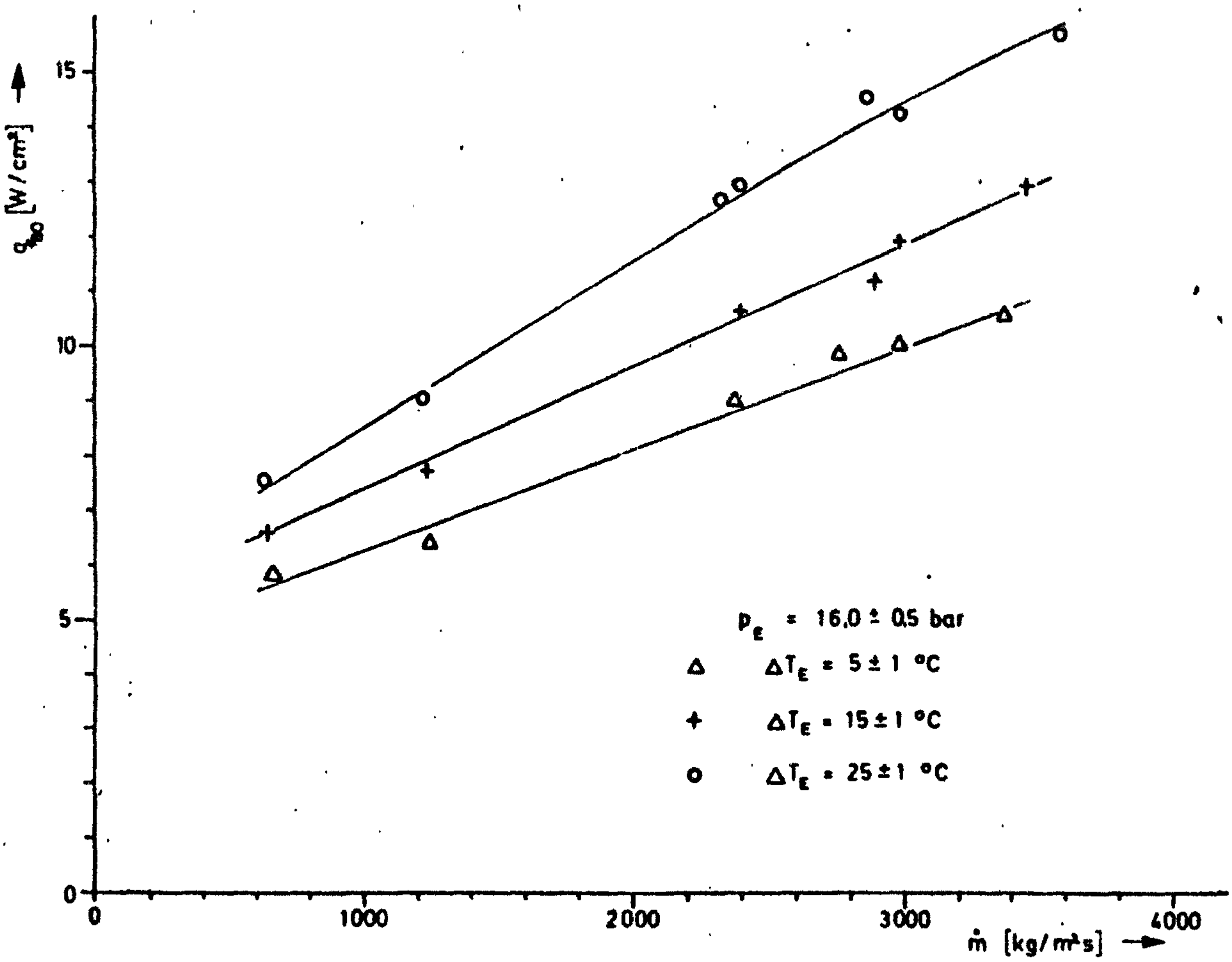


ROD POWER	= 5,16	(KW)
INLET PRESSURE	= 10,77	(BAR)
EXIT PRESSURE	= 9,51	(BAR)
MASSFLOW RATE	= 3684.	(KG/M2/SEC)
INLET TEMPERATURE	= 22,79	(DEG C)

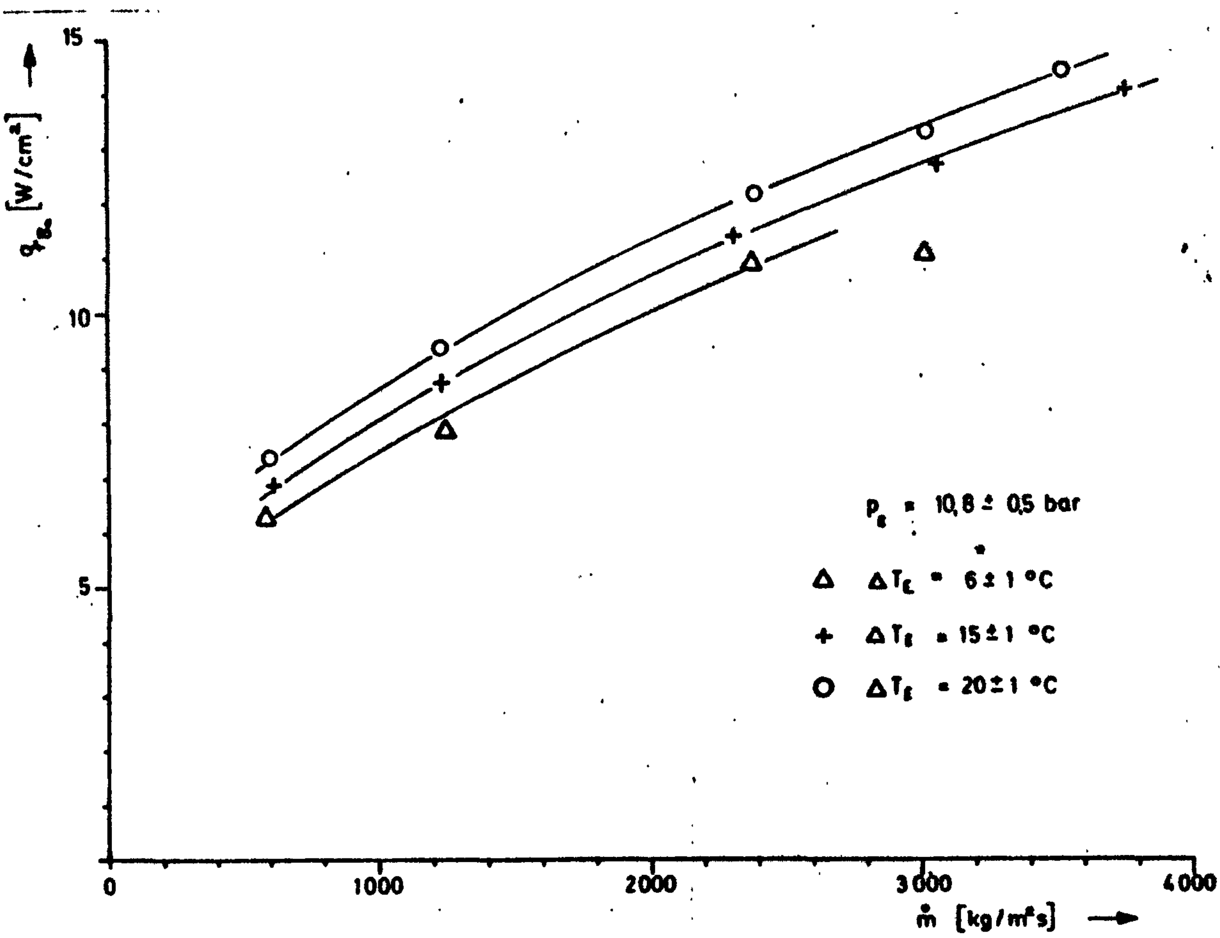

heated rods



ROD POWER	= 13,35	(KW)
INLET PRESSURE	= 10,75	(BAR)
EXIT PRESSURE	= 9,47	(BAR)
MASSFLOW RATE	= 3697.	(KG/M2/SEC)
INLET TEMPERATURE	= 23.89	(DEG C)



Pict. 5 Critical heat flux vs massflow rate for different inlet temperatures



Pict. 6 Critical heat flux vs massflow rate for different inlet temperatures

Classification: 1.1.2		
Title 1 (Original Language): Stationäre DNB-Messungen in Frigen mit komplexer Abstandshaltergeometrie (RS 176 - II.1.3., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GKSS, Geesthacht
Title 2 (english): Steady DNB Measurements in Freon with Complex Spacer Geometry		**Project Leader:** Fulfs Katsaounis
Initiated (Date): 1.9.1975	**Completed (Date):** 30.7.1977	
Status: continuing	**Last Updating (Date):** December 1975	

1. General aim

In addition to the research program RS 164 this program will be the basis for an experimental study of the modelling laws between water and freon as for DNB-measurements.

2. Particular objectives

The program is divided in the following three parts

- I cold flow pressure drop measurements
- II mixing experiments with power, measurements of subchannel exit temperature
- III Critical heat flux measurements

In addition and supplement to the research program RS 164 these experiments will be used for recalculating the results from research program RS 64 Part I and II PWR to water conditions.

3.1 Experimental facilities

The geometry of the test section is squarely arrayed by 25 rods with a pitch of $14.3 \cdot 10^{-3}$ m. 16 of them are electrically-heated with a relative power of 1.0 and 6 with a relative power of 1.2 while 3 are unheated. All heated rods are instrumented with thermocouples near the exit end of the heated length, which is 2.985 m long. The outer diameter of the heated rods is $10.75 \cdot 10^{-3}$m, that of unheated rods $1{,}372 \cdot 10^{-3}$m. Wall spacing of the heated rods is $4.544 \cdot 10^{-3}$m, grid spacing is 0.534 m. 4 pressure taps give information about inlet and outlet pressure and pressure drop across spacers.

The experiments will be carried out at the steady state burnout-facility of GKSS research department, see also research program RS 64 in this report.

3.2 Research program

In order to check the condition of test section and the accuracy of methods for pressure drop prediction cold flow pressure drop measurements will be carried out.
To get information about mixing effectiveness subchannel exit temperature measurements will be made at different levels of bundle power and inlet enthalpy. Critical heat flux tests will be carried out over a large range of inlet conditions e.g. massflow rate, inlet temperature, system pressure , , mostly valid for PWRs. For some aspects CHF-points at a pressure of 70 bar will be investigated.

4. Project status

4.1 Progress to date

The test section is under construction now and the mechanical and electrical components will be ordered soon.
To assure that test conditions will be the same for RS 176 and RS 164 part of the CHF-tests of RS 164 at the high pressure water loop of Columbia University, New York N.Y. had been joined as well as loop and test section been inspected.

4.2 Essential results

The program has started 4 month ago. Therefore no results are available. There is no delay in shedule but as the tests themselves will be run in between RS 64 - approximately at the end of 1976 - any delay from RS 64 will forward the shedule of RS 176.

5. Next steps

a) Complete ordering components (mechanical and electrical)
b) Set up of test conditions according to RS 164
c) Investigation of present similarity laws freon-water

6. Relation with other projects

RS 64 Investigation of the Steady and Transient Critical Heat Flux of Multi-Rod Bundles for PWRs and BWRs with Freon as Model Fluid.
GKSS; 1972 - 1978

RS 164 Steady DNB-Measurements in Water with Complex Spacer Geometry
KWU-Erlangen; 1975 - 1976

7. Reference documents

No documents or reports till now.

8. Degree of availability

All future reports will be available with the allowence of IRS, department Forschungsbetreuung.

Classification: 1.1.2		
Title 1 (Original Language): Stationäre DNB-Messungen an Brennstabbündeln mit komplexer Abstandshalter-Geometrie in Wasser (RS 164 - II.1.3., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Stationary DNB Experiments on Fuel Rods with Complex Spacer Geometry in Water		Project Leader: H. Ulrych
Initiated (Date): 1. 3. 75	Completed (Date): 31. 8. 76	
Status: Continuing	Last Updating (Date): 31. 12. 75	

General Aim and Particular Objectives

The transferibility of DNB-measurements in freon as a model fluid to water will have to be investigated, especially for the complex geometry of KWU-spacers. With respect to an extended experimental test at GKSS with freon (RS 64) it will be essential to work out a statement, how the results can be transferred to water conditions and which accuracy can be reached. The comparative tests in water and freon will confirm the results of the transient test program at GKSS.

Experimental Facilities

a) 5.3 MW - Water loop at Columbia University, New York, N.Y., USA

b) 1.1 MW - Freon loop at "Gesellschaft für Kernenergieverwertung in Schiffbau und Schiffahrt", Geesthacht, Germany

Research Program

For two bundles with the same geometry in the freon loop at GKSS (RS 176) and in the water loop of the Columbia-University (RS 164) measurements of the critical heat flux will be conducted. The investigations will be carried out with a 5 x 5 bundle in PWR-geometry with KWU spacers and control rod guide tubes; the bundle will be heated uniformly in axial direction.

Project Status/Progress to Date/Essential Results

Bundle 1 and bundle 2 were run in water for the specified data points. The critical bundle power differed only slightly with respect to the spacers A and B with different mixing grame orientations. Comparative calculations with the subchannel analysis code THERMOHYDRAULIK gave good agreement.

Next Steps

The test runs will be evaluated. The results will be analysed carefully.

Relation with Other Projects

RS 64 Investigation of the Steady and Transient Critical Heat Flux of Multi-Rod Bundles for PWR's and BWR's with Freon as Model Fluid

RS 176 Freon tests with the same bundle geometry

Reference Documents/Degree of Availability

No reports available.

Classification 1.1.2

Title 1 Depressurisation Discharge Rate	**Country** UK **Sponsor** UKAEA
Title 2	**Organisation** AWRE Foulness
Initiated 1968 **Completed** **Status** Continuing **Last updating**	**Project leaders** A.R.Edwards

1 **General Aim**

To enable flows, temperatures and forces to be predicted following accidental depressurisation of a water reactor through a large break.

2. **Particular Objectives**

To establish a suitable way of calculating the flow in a pipe discharging to atmosphere from a broken end.

3. **Experimental Facilities**

Pipes of different lengths and diameters are pressurised to PWR/BWR conditions, then allowed to discharge through a rapidly broken bursting disc. Pressures, flow and voidages are measured.

4. **Project Status**

Progress to Date: Measurements of transient pressure, temperature and voidage have been made in three constant diameter pipe systems, each 4m long and of 32, 73 and 200 mm diameter. In all cases the pipes were initially completely liquid filled, generally with 35 bar overpressure. Initial temperatures corresponding to 35,70,105, 140 bar saturation pressure were used for the two smaller pipes and 35 bar for the largest pipe. Results obtained may be compared with predictions from depressurisation codes. In addition a limited programme of work to examine the blowdown of a vessel, through a pipe into a containment vessel, has also been carried out to provide data for checking the validity of various critical flow discharge models. These tests started from 50 bar saturation pressure in the reservoir.

Work has continued to measure transient pressure, temperature and density changes in steam/water mixtures during the blowdown of an 8 inch diameter pipe 12 ft. long. These tests incorporate a multi-beam X-ray system to make a detailed examination of the changing void distribution at one particular cross section during the blowdown. The report on the first test has now been published. A paper describing the multi beam X-ray system has been published in the BNES journal.

A repeat test has been carried out and a preliminary examination of the results indicates very good agreement with the previous test results. The X-ray system has now been moved to the discharge end of the pipe and the final alignment checks have been nearly completed. It is hoped to carry out two further tests before the experimental work is terminated at the end of March 1975.

Reference Documents

Aldermaston Report AWRE/44/86/97 (SRD/R/29)

Heat transfer during blowdown	1.1.2 Thermal Hydraulics
	COUNTRY UK SPONSOR UK - NII ORGANISATION Univ. of Manchester
PWR BLOCKAGE EXPERIMENT: An investigation into the effects on heat transfer of a region of swollen fuel cladding causing a partial flow blockage in the core of a Pressurised Water Reactor (PWR)	<u>Project Leader</u> Prof. W. B. Hall
Initiated October 1975 Status progressing	<u>Scientists</u> J. T. Turner G. P. Ioannu

1,2. General Aims and Particular Objectives

The objective is to provide experimental data to be useful in assessing the influence of a region of swollen fuel-rod cladding on a loss of coolant accident in a pressurised water reactor. Particular attention will be given to the temperature changes which might occur at the boundaries of the swollen region. Detailed flow and heat transfer data within the rod bundle will be obtained from a scale model and airflow facility.

3. Experimental facilities

An airflow rig has been developed to permit the measurement of heat transfer and flow behaviour within the fuel rod bundle. The bundle consists of an 18x18 rectangular array of 12.7 mm diameter rods on a 17 mm pitch and a central 7x7 array of swollen rods. Within this swollen region, there is a 5x5 array which can be heated electrically under conditions of constant heat flux.

Thermocouples placed on the heated rods permit the measurement of surface temperatures. Instrumentation has also been developed to enable flow velocity and static pressure distributions within the rod bundle to be established. Data logging and digital computer methods are being employed so that changes in the extent of the blockage, the influence of Reynolds number and the heat transfer rates can be readily studied.

4. Project Status

The apparatus is now virtually completed and much of the computer software has been developed.

It is anticipated that detailed experimental work will commence shortly.

contd.

5. <u>Next steps</u>

Examination of experimental data. Long-term development of a prediction technique yielding heat transfer behaviour under accident conditions.

6. <u>Relation with other projects</u>

Linked to range of research projects on Reactor Safety at the University.

7. <u>Reference documents</u> None

8. <u>Degree of availability</u>

On application to the NII when available

Heat transfer during blowdown	1.1.2	Thermal Hydraulics

	COUNTRY UK SPONSOR UK - NII ORGANISATION Univ. of Manchester
Transition to film boiling induced by a pressure reduction	Project Leader Prof. W. B. Hall
Initiated October 1974 Status: progressing	Scientists A. WATSON H. V. ERSOZ

1. General aim

Experimental measurements of heat transfer from a wire to water during a rapid depressurisation.

2. Particular objectives

The fluid used is water. Stage 1 of the program is to depressurise from 20 bar and 180°C to atmosphere. Stage 2 is to depressurise from 150 bar and 340°C.

3. Experimental facilities and programme

Stage 1. A pressure vessel of approx. 1 litre capacity is fitted with a platinum wire 0.1 mm diameter, 20 mm long, which is heated at approximately constant uniform heat flux. A double bursting disc arrangement is used to achieve depressurisation from a fixed pressure within the vessel. An intermediate water filled chamber lies between the pressure vessel and the atmosphere and is separated from each by a bursting disc. Increase in pressure in the intermediate chamber causes the discs to burst in sequence, the outer one first. Transient measurements of power to the wire, wire temperature and pressure are measured with a high speed digital system.

Stage 2. No apparatus has yet been built.

4. Project status

1. Progress to date. The bursting disc technique has been developed. Depressurisation times of 1 ms have been achieved using ambient temperature water at 20 bar.

2. Essential results. None

5. Next steps

Continuation with Stage 1. Selection of geometry and initial conditions required for Stage 2.

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6. Relation with other projects

Thermal boundary conditions like those of a PWR fuel element may be simulated.

Ref. Simulation of the thermal dynamics of a heated surface (sodium contract)

7. Reference documents

None

8. Degree of availability

On application to the NII when available.

1. Budget

£3358	Equipment + overheads	Totals for 2 yrs.
£2260	Research student salary (H. V. Ersoz)	

2. Personnel

Research student	H. V. Ersoz
Academic staff	Prof. W.B. Hall, Dr. A. Watson
Technicians	1, shared with other projects.

3. Additional information

Time schedule: Stage 1 planned completion Oct 1976.

Stage 2 Construction during 1976.

	COUNTRY UK
	SPONSOR UK - NII
	ORGANISATION Univ. of Manchester
The Thermal Dynamics of a Heated Surface	PROJECT LEADER Prof. W. B. Hall
Initiated: October 1974 Status: Progressing	SCIENTISTS C. Tye J. O. Oyinloye

1. <u>General aim</u>

The development of an experimental technique to control the power of an electrically heated surface so that it simulates the behaviour of a reactor fuel element.

2. <u>Particular objective</u>

Simulating the correct boundary conditions on a heater surface for experimental heat transfer studies during depressurisation and re-wetting.

3. <u>Experimental facilities and programme</u>

A digital computer operating in real time is used to simulate the dynamic behaviour of a reactor fuel element by numerical solution of a one dimensional time dependent heat conduction equation. The computed surface heat flux is fed to a control system that regulates the power of heater (currently a thin platignum wire). The surface temperature of the heater is then fed back to the digital computer as a boundary condition for the solution of the conduction equation. Using this technique, fuel elements with a wide variety of physical properties, temperature profiles, heat generation etc may be simulated to provide more realistic boundary conditions in experimental heat transfer studies.

4. <u>Project Status</u>

(i) A control system has been developed to accurately regulate the

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4(i) contd.

power in a thin platinum strip or wire and to give stable operation in convective, nucleate, transition and film boiling. Real time computer simulations of reactor fuel elements have implemented on a Honeywell DDP516 Mini computer with 8K of memory.

(ii) Essential results None

5. Next steps

To fully test the system against an experiment with well known heat transfer properties.

6. Relation with other projects

It is intended to use the experimental technique in the following areas:

a) Transition to film boiling induced by a pressure reduction.

b) Rewetting of a hot surface with a liquid coolant.

7. Reference documents

None

8. Degree of availability

On application to the NII when available

PROJECT TITLE : Investigation of the transient flow response in a BWR core	CLASSIFICATION 1.1.2
SPONSORING COUNTRY : Italy	ORGANISATION : CNEN - CISE
DATE INITIATED : 1974 DATE COMPLETED : Dec. 1975	PROJECT LEADER : V. Marinelli (CNEN) G. P. Gaspari (CISE)

Description :

1. General aim

Experimental validation of a computer code for the calculation of the onset of the CHF under LOCA conditions.

2. Particular objectives

Validation of the thermal-hydraulic response of the code by means of mass hold-up measurements during transient simulating LOCA.

3. Experimental facilities and programme

Measurements of mass hold-up during transients of inlet flow stoppage at constant power and pressure and inlet flow stoppage and power decay at constant pressure will be taken on a 16 rod BWR test section 12 ft long: the plant used is IETI-III (CISE).

4. Project status

4.1 Progress to date

The computer code, named DOLCE, has been developed and satisfactory results have been so far obtained for the predictions of transient CHF in a BWR during LOCA and other transients. The mass hold-up measurements have been initiated.

4.2 Essential results

Both Eulerian and Lagrangian methods used in the DOLCE code give resonably good results in the evaluation of transient CHF.

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5. Next steps

See Items 3, 4.1

6. Relation with other projects

1.1.2 (other programs)

7. Reference documents

DOLCE Computer Code, CNEN Internal Report - Under publication

8. Degree of availability

To a limited extent.

9. Budget

60 M Lit. Personnel 50 MM.

PROJECT TITLE : Investigation of flow blockage effects in a subchannel array	CLASSIFICATION 1.1.2
SPONSORING COUNTRY : Italy	ORGANISATION : CNEN A.B. Atomenergy (Sweden)
DATE INITIATED : June 1973 DATE COMPLETED : May 1976	PROJECT LEADER : V. Marinelli (CNEN) B. Kjellen (Sweden)

Description :

1. General aim

Study the flow redistribution of the coolant in blocked subchannels.

2. Particular objectives

Validate the theoretical methods as LEUCIPPO, COBRA, in their ability to predict the flow distribution in blocked subchannels by extensive comparison with experimental data.

3. Experimental facilities and programme

Single phase flow is completed in a test section of 4x4 rods 2,5 m long where the blockages are installed. Measurements of the three components of the velocity are taken, by means of a special 5 Pito-tubes probe, as well as radial and axial pressure drops. The blockages consist in 100% and 70% obstructions within the center subchannel, in a large blockage of one half test section and also in ballooning type blockage of four rods.

4. Project status

4.1 Progress to date

The measurements have been completed.

4.2 Essential results

Preliminary indications show the existence of long relaxation (50 D) length before the normal flow redistribution is obtained downstream a blockage.

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5. Next steps

Analytical work will be done to reduce the data and to compare them with the subchannel code predictions.

6. Relation with other projects

1.1.2 (other programs)

7. Reference documents

None

8. Degree of availability

To a limited extent

9. Budget

40 M lit. Personnel : 36 MM

PROJECT TITLE : Thermohydraulic transients in pressure tube reactors during blowdown. Transitori termoidraulici in reattori a tubi in pressione durante lo svuotamento.	CLASSICATION 1.1.2.
SPONSORING COUNTRY : Italy (mainly) and Canada	ORGANISATION : CISE sponsored by CNEN (mainly) and AECL.
DATE INITIATED : 1968 DATE COMPLETED : 1977	PROJECT LEADER : At present under responsibility of UTM (CISE)

Status (latest updating): May 1975

Description :

1. General aim: to set up a reliable and verified calculation procedure to predict thermohydraulic transients in pressure tube reactors during blowdown.

2. Particular objective: understanding of basis thermohydraulic phenomena involved in blowdown conditions in water reactors.

3. Experimental facilities and programme

3.1. Experimental facilities

3.1.1. IETI-1: multi-purpose facility for scaled-down experiments; open circuit; flowrate: 0,8 kg/s; pressure: 100 bar; preheating power 700 kW (AC); test section power 300 kW (DC).

3.1.2. CIRCE : large-scale facility simulating in a closed circuit 2 full-scale power channels; water flowrate: 22 kg/s; steam flowrate (from circulator or boiler) 3 kg/s; test section power 12,5 MW (DC).

3.2. Programme

3.2.1. scaled-down blowdown tests with simple geometries simulating breaks both upstream and downstream of the power channel;

3.2.2. researches concerning single thermohydraulic phenomena involved in blowdown;

3.2.3. integrated blowdown tests simulating breaks in different circuit locations with a full-scale geometry relevant to a single power channel;

3.2.4. blowdown code development for thermohydraulic transient predictions.

4. Project status

4.1. Progress to date (with reference to the above programme)

- (3.2.1.): almost completed
- (3.2.2.): tests regarding heat transfer crisis in transient conditions completed and fully analyzed; preliminary results relevant to post dryout heat transfer and hot surface rewetting obtained.
- (3.2.3.): 17 blowdown tests simulating inlet header failure carried out.
- (3.2.4.): a prediction code (TILT) developed; a more sophisticated version (RATT) in progress.

4.2. Essential results

- Set up of suitable experimental procedures and techniques for transient conditions.
- Availibility of substantial amount of experimental information relevant to blowdown transients, both in scaled-down and full-scale conditions, in terms of mass, pressure and temperature.
- A satisfactory understanding and prediction of heat transfer crisis in transient conditions; reliable predictions of steam-water density and pressure drops; limited understanding of post dryout heat transfer and rewetting phenomena.
- Availibility of a sufficient calculation model for transient conditions.

5. Next steps

- Further experiments and analysis of post dryout heat transfer and rewetting phenomena; starting of research programmes about critical two-phase flow, heat transfer crisis in stagnation and reverse flow, flow distribution in parallel channels during blowdown.
- Full-scale integrated tests under a wider range of parameters and conditions.
- Implementation of up-to-date physical models in prediction codes.

6. Reference documents (Main titles)

1) A. Magni "TILT - a digital simulation programme for the study of hydrodynamic processes and core heat-up of boiling water pressure tube reactor during transient conditions" Proceeding of the CREST Specialist Meeting on ECC for high water reactor. Munich (October 1972).

2) A. Premoli, D. Di Francesco, A. Prina "Una correlazione adimensionale per la determinazione della densità di miscele bifasiche" LA TERMOTECNICA, n. 1, January 71.

3) G.P. Gaspari, R. Granzini, A. Premoli, C. Sandri "Mass holdup, pressure and time to dryout predictions under LOCA conditions.Comparisons with scaled down experimental results" Paper presented at the European Two-Phase Flow Group Meeting Harwell 3-5 June 1974 and ASME pubblication 74.WA/HT-43 presented at the Winter Annual Meeting, New York, November 17-22, 1974.

4) Baldassarre R., Gaspari G.P., Granzini R., Pagliari V. "Predictions of transient dryout by using TILT code and steady state CHF correlation" CISE Report to be published in 1975.

5) A. Azzalin, A. Premoli, R. Ravetta, V. Tarzia, T.S. Thompson "An experimental investigation on blowdown in pressure tube reactor conditions" CISE R-342 (Nov. 1973).

6) A. Azzalin, M. Dubbini, A. Premoli, B. Prevedini, V. Tarzia, R. Ravetta "Experimental tests on CIRCE loop under typical conditions concerning DBA (Design Basic Accident) of CIRENE Reactor" paper presented at the "Juice Meeting on Reactor Safety" Sheridan Park, Nov. 5-6, 1974.

7) Agostini V., Premoli A., "Valvola di intercettazione rapida per impiego in acqua-vapore" Energia Nucleare Vol. 10, 1, January 1971.

8) Pierini G., Sandri C. "The RATT code under development at CISE in support of the pressure tube reactor LOCA analysis" Meeting of European Two-Phase Flow Group, Haifa, June 1-5, 1975.

9) A. Premoli "An experimental investigation on voiding of power channels cooled by steam-water mixtures" Energia Nucleare, 16, (1969).

7. Degree of availability: to a limited extent.

PROJECT TITLE : Analysis of pressures in a containment	CLASSIFICATION 1. 1. 2.
SPONSORING COUNTRY : ITALY	ORGANISATION : NIRA-Cirene
DATE INITIATED : 1974 DATE COMPLETED : end of 1975	PROJECT LEADER : NIRA

Description :

Scope of the work is the analysis of the pressure, and the pressure differential in a containment. Jet forces internal missiles are also investigated.

A code, PACO, has been developed for the pressure transient and pressure differential evaluation.

The work is performed in the framework of the contract awarded from CNEN to NIRA for the design of the CIRENE prototype Nuclear Island.

Reference documents :

NIRA Report

"PACO-N - Codice per il calcolo di transitori di pressione e tempe - ratura" 01300 - NT - 30

Classification 1.1.2.	
Title 1 Blowdown code assessment	Country : JRC Sponsor : CEC Organization : JRC ISPRA Establishment
	Project leader : G. Forti
Initiated : January 1974 Completed : 1980 Status : progressing Last updating : March 1975	

1) General aim

To acquire a working knowledge of the scope and limitations of the major accessible blowdown/ECC codes

- To compare the main codes with well defined experimental results to demonstrate their abilities to predict real situations

- To implant fundamental improvements in the theory and numerical methods used by the more promising of the codes, or develop a completely new code with the required capabilities

2) Particular objectives

Theoretical back-up of the Ispra blowdown programme

3) Experimental facilities and programme : -

4) Project status

1. Progress to date : General study of theory and codes. Extensive tests of RELAP 3. Tests with DANAIDES. Analysis and sensitivity study of THETA 1-B. Development of the NICKY equilibrium blowdown code in progress.

7) Reference documents :

JRC Safety programme progress report 1974.
NICKY - A computer programme for the analysis of blowdown in nuclear power reactors in an equilibrium approximation by G. Forti, NEA meeting on LOCA computer programmes Ispra Oct. 1974 (to be published).

8) Degree of availability : Freely available

9) Budget : No investments, only computer time.

10) Personnel : 3 men/year.

11) Additional Information

Classification: 1.1.4	
Title 1 (Original Language): Bestimmung der Nachzerfallswärme von U-235 (PNS 4234 - I.1.3., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Decay Heat Measurement of 235 U	Project Leader: K. Baumung
Initiated (Date): Sept. 1974 Completed (Date): 1977	
Status: continuing Last Updating (Date): December 1975	

1. General aim

As present uncertainties of the 235 U decay heat data contribute one of the main uncertainties to the LOCA analysis, more accurate data shall be provided.

2. Particular objectives

Calorimetric measurements of the decay heat will be performed in the time period of 10 to 1000 seconds after irradiation with LWR-pellet-like samples of UO_2 and uranium metal of corresponding size.

3. Experimental facilities and program

The fuel samples will be irradiated in the thermal column of the FR2 reactor and then pneumatically transported into a computer-controlled adiabatic microcalorimeter. The time dependent temperature rise of the samples and energy lost due to γ-ray-escape will be recorded and directly yield the total delayed power release.

4. Project status

4.1. Progress to date

After a study of the literature on decay heat determination the calorimetric method was assumed to be the most promising because it is the simplest one and directly yields the quantity of interest

Therefore a computerized adiabatic microcalorimeter with a short time constant was drawn up by which it should be possible to extend the range of calorimetric measurements of the decay heat to cooling times as short as 10 seconds.

The γ-energy-flux escaping the samples will be measured by a Moxon-Rae-type detector.

4.2. Essential results

Fuel samples:
Unlike the metallic samples the UO_2 devices show bad thermal conductivity. In order to avoid temperature gradients through the sample after the initial temperature profile due to the reactor irradiation had flattened, a suitable 235U-enrichment could be determined. With this enrichment the slope of the β- and γ-energy absorbtion is just compensated by the activation profile due to selfshielding, thus providing a flat heat source distribution.

γ-escape measurement:
Carrying away up to 40 % of the total power released, the γ-energy-flux from the samples must carefully be measured. As this flux is non-isotropic, a timedependet correction has to be provided. This correction which is based on measured delayed γ-spectra and was computed with a Monte-Carlo-code, will be used to calculate the total energy loss by γ-escape from the measured counting rates.

5. Next steps

In parallel to the construction of the equipment, the control programs will be provided. Then, after cold tests, the experiment will be installed at the reactor and measurements started.

6. Relation with other projects

This experiment yields basic data for the LOCA analysis.

7. Reference documents

/1/ 1st PNS-Semiannual Report 1975, KFK 2195 (German with English abstracts)
/2/ 2nd PNS-Semiannual Report 1975, KFK 2262 (German with English abstracts)

8. Degree of availability

Unrestricted distribution.

PROJECT TITLE : BURNOUT IN REVERSE FLOW	CLASSIFICATION I.I.4
SPONSORING COUNTRY : ITALY	ORGANISATION : CNEN
DATE INITIATED : JANUARY 75 DATE COMPLETED : DECEMBER 76	PROJECT LEADER : G.B. FARELLO

Description :

Critical heat flux for tubes with very low flow rates is a very important problem for predicting Burn-out during blow down after the LOCA. The test section for that experiment allows measurement of the critical heat flux in two tubes, four meters long, 11.5 mm inside diameter, with an upward or downward water flow and a specific mass velocity between 100 and 1000 kg/m^2 sec.

The experimental set up is a S.S. loop whose maximum power is 4 MW and whose maximum flowrate is 27 tons/h.

CLASSIFICATION

1.2

TITLE 1	MELANGE DE L'ECOULEMENT PRINCIPAL ET DU DEBIT D'INJECTION.	COUNTRY FRANCE SPONSOR E.D.F./SEPTEN ORGANIZATION C.E.A.
TITLE 2	STEAM WATER MIXING.	Project Leader C.E.A./DTCE/SE/EN
Initiated 1975	Completed 1977	Scientists M. HOUDAYER E.D.F.
Status Avant Projet d'Installation	Last updating : 20.01.75	M. POCHARD CEA

I - GENERAL AIM

Injection de secours PWR.

II - PARTICULAR OBJECTIVES

Etude de l'interaction mécanique et thermodynamique du débit d'eau froide provenant de l'injection d'eau de réfrigération de secours (lors d'un accident de perte de réfrigérant primaire) avec le débit principal, en diphasique.

III - EXPERIMENTAL FACILITIES AND PROGRAMME

Trois types d'expériences sont envisagés :

- EPIS I : expérience préliminaire eau-air à l'échelle 1/11 destinée à mettre en évidence les effets dynamiques à l'exclusion des effets thermique (condensation de vapeur).
- EPIS II : expérience préliminaire eau-vapeur à l'échelle 1/25 destinée à mettre en évidence l'importance des phénomènes de condensation.
- EPIS III : (éventuellement) expérience air-eau à l'échelle 1/3 pour étudier l'effet d'échelle du point de vue dynamique.
- ELISE : expérience globale eau-vapeur sur maquette.

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IV - PROJECT STATUS

4.1 - Progress to date

Avant projet de EPIS I et EPIS II.

4.2 - Essential Results

V - NEXT STEPS

VI - RELATION WITH OTHER PROJECTS

Code CLYSTERE.

VII - REFERENCE DOCUMENTS

Néant.

VIII - DEGREE OF AVAILABILITY

Propriété E.D.F - C.E.A.

Title 1 (original language)	COUNTRY: FRANCE
Programmes de calcul pour l'étude du renoyage	SPONSOR: CEA
	ORGANIZATION
	C.E.A.
Title 2 (english) Reflooding computer codes	Project leader DSN/SETS N.TELLIER Scientists :
Initiated (date) 1974 Status : progressing Programmes en cours de tests et d'amélioration Completed : (date) Last updating (date) Janvier 75	

1. But général

Mise au point de programmes de calcul pour l'étude de la phase de renoyage de l'accident de perte du caloporteur d'un P.W.R.

2. Objectifs particuliers

Mise au point du code CERES pour l'étude de la thermohydraulique du circuit primaire pendant la phase de renoyage.
Mise au point du code FLIRA pour l'étude du renoyage d'un canal.
Ajustement des codes sur les essais ERSEC
Application aux calculs relatifs aux essais PHEBUS et aux calculs de réacteurs de puissance.

3. Installations expérimentales et programmes

———

4. Etat du projet

a) code CERES

En cours de tests (les coefficients d'échange dans le coeur sont calculés à l'aide des corrélations FLECHT).

b) code FLIRA

1ère version en cours de test: la vitesse de montée du front de trempe est donnée par une corrélation déduite des expériences ERSEC.

5. Prochaines étapes

Mise au point de FLIRA 2 où la vitesse de montée du front de trempe est calculée 1975

Interprétation des essais ERSEC

Couplage avec CERES

.../...

6. Relation avec d'autres projets

Essais sur la boucle ERSEC
Essàis sur la boucle PHEBUS

7. Documents de références

- FLIRA : Un modèle de calcul de remouillage après un accident de perte du fluide primaire

  par M.CHABRILLAC et J.P.L'HERITEAU

  NEA-CPL Thermal Reactor Safety Seminaire ISPRA 23-25 octobre 74.

- Programme de calcul de renoyage de P.W.R. : Code CERES

  par LANGE et MEGNIN

  note G.A.A.A. - ETG-NT-73 195

<table>
<tr><td colspan="2">Title 1 (original language)

Etude expérimentale du refroidissement de secours des réacteurs à eau.</td><td>COUNTRY :FRANCE
SPONSOR :C.E.A.
ORGANIZATION
C.E.A.</td></tr>
<tr><td colspan="2">Title 2 (english)

Expérimental study of the water reactor safety cooling</td><td rowspan="2">Project leader
CEA/DTCE/STT
M.DERUAZ
Scientists :</td></tr>
<tr><td>Initiated (date)

Status : progressing</td><td>Completed : (date)

Last updating (date)
Janvier 1975</td></tr>
</table>

1. But général

Etude du transfert de chaleur lors de la phase refroidissement de secours de l'accident de perte de caloporteur.

2. Objectif particulier

Essais sur un tube de refroidi intérieurement sur grappes dans le but de mettre au point des corrélations de coefficient d'échange fonctions des paramètres thermohydrauliques locaux.

3. Installations expérimentales

Boucle ERSEC II - principales caractéristiques

Pression	1 à 6 bars
température initiale de paroi	400 à 900°C
vitesse de renoyage	0,5 à 10 cm/s
nombre de barreaux	1 à 64
hauteur chauffante	3,65m (3,20m pour les premiers essais sur 1 tube)

4. Etat du projet

Section tubulaire, hauteur 3,20 m, flux axial uniforme terminé (nov.73-mars 74)
Groupe de 25 barreaux, hauteur 3,60m, géométrie 15x15 flux cosinus en cours depuis mars 1974.

Interprétation:

Cas du tube (coefficient d'échange en aval du front de trempe, modèle équilibré sans rayonnement) terminé.

.../...

5. Prochaines étapes

Essais :

Grappe de 25 barreaux géométrie 15x15 flux axial.....fin prévue :juillet 75
en cosinus.

Grappe de 36 barreaux géométrie 17x17 flux axial.....Janvier 75 - Mars 76 et
en cosinus. sept. 75 - Novembre 75

Tube avec faible fuite thermique ....Mars 1975 - Juin 75

Début des essais sur B.W.R. 8 x 8 ....Janvier 76.

6. Interprétation

Essais sur tube, modèle de déséquilibre, rayonnement pris en compte de façon simplifiée ....Décembre 75

Application au cas des grappes ....Courant 76

Coefficient d'échange au niveau de la zone de transition ....Juin 76

6. Relation avec d'autres projets

Essais sur la boucle PHEBUS
Programmes de calcul pour l'étude du renoyage

7. Documents de référence

Heat transfer during the reflooding of a tubular section

by. D.ANDREONI, M.COURTAUD. R.DERUAZ

European two-phase flow meeting 3-7 juin 1974 - Harwell

	Classification : 1.2 1.1.1
Title 1 (original language) EVA PROGRAM	Country : FRANCE Sponsor : CEA FRAMATOME Organization CEA FRAMATOME WESTINGHOUSE
Title 2 (English) Two-phase flow pump test program. Joint R & D program between FRAMATOME and CEA with the WESTINGHOUSE Participation.	**Project leader:** Mr. DELAYRE CEA Mr. DUBOURG FRAMATOME
Initiated (date): JUNE 1974 Completed (date): DECEMBER 1976 Status: PROGRESSING Last updating (date): JULY 1975	**Scientists** : Mr. FAJEAU CEA Mr. MARINI FRAMATOME

1. OBJECTIVES

The dynamics of the reactor coolant pump play key role in determining the consequences of a hypothetical loss of coolant accident (LOCA).

For a more accurate and refined representation of the pump model, the pump performance will be measured under the different conditions of pressure, two-phase flow, and speed that might occur during the LOCA.

The "EVA" test loop is designed for testing a WESTINGHOUSE primary pump (1/3 scale model) in order to :

1/ Measure the pump characteristics during the conditions simulating the LOCA

2/ Develop a correlation of these two-phase flow results with pump performances as measured in simple phase.

The experiments will be performed with steady state steam water flow in homogenous and non homogenous conditions.

2. PROJECT STATUS

The EVA test facility is under construction at Cadarache. The test facility is using as a source of steam, the steam supplied by PAT reactor.

The main components of the loop such as the steam water mixer, the steam water separator, the circulation pumps and the measuring devices are near completion and the erection of the test loop is underway.

The qualification tests of the instrumentation of the loop will start in August.

3. PLANS FOR NEAR FUTURE

The loop is supposed to be completed in October and the shakedown tests of the loop will be performed in November.

The 1st test point will be run in December.

About one thousand of test points will be run representing all flow conditions and operating modes of the pump which may be anticipated during a loss of coolant accident.

The test program will spread out on the whole 1976 year.

4. RELATIONS WITH OTHER PROJECTS

EDF Programs on Pumps in both simple phase and two phase conditions.

5. AVAILABILITY OF "RESULTS"

Joint Property of CEA, FRAMATOME and WESTINGHOUSE.

Classification: 1.2		
Title 1 (Original Language): Durchführung theoretischer Arbeiten im Rahmen des Notkühlprogramms: Auswertung der Flutversuche am Einrohr und Stabbündel (RS 0036 A - I.1.2; A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Theoretical Studies within the Framework of the Emergency Core Cooling Program; Evaluation of the Flooding Experiments with Single Tubes and Rod Bundles		Project Leader: Dr. Riedle
Initiated (Date): April 1972	Completed (Date): March 1975	
Status: Completed	Last Updating (Date): December 1975	

General Aim and Particular Objectives

The experimental data recorded within the experimental project RS 36 on reflood heat transfer and hydraulic were interpreted to give information on core cooling, quenching times and reflood water level rise.

Experimental Facilities and Program

The experimental program RS 36 on reactor reflood included three experimental facilities, Monotube reflood, PWR bundle and BWR bundle. The tests were completed.

Project Status/Progress to Date

The data evaluation of the BWR blowdown tests under low pressure were continued (Type A, B, C). A recalculation of the swell levels and the local distribution of the steam was conducted with the correlation for the steam bubble ascent velocity of YEH and CUNNINGHAM under various pressures and powers, in order to test the consistence with the measured data.

Project Status/Essential Results

The calculated heat transfer coefficients were in good agreement both from the difference of the measured tube and saturation temperature of the fluid and the values from the steam convection, as long as the steam was not overheated, that means the axial measuring position must not be too far away from the swell level.

From the swell level at a certain time, when an axial measuring point was dried out, and the corresponding collapsed level some statements could be made about the upscuming of the water in the bundle containment. The swell level depended on water mass, pressure, steam input and the power. The results of the swell levels showed, that cooling and dry-out occur nearly at the same time over the whole bundle (in radial direction).

Next Steps

Work on this project has been completed.

Relation with Other Projects

RS 0037 C Emergency Core Cooling Program - PWR Post DNB Experiments with a bundle of 25 fuel rods

Reference Documents/Degree of Availability

H.P. Gaul, K. Riedle, K. Ruthrof, J. Sarkar, H. Amm, G. Blank
Notkühlprogramm ND-Versuche: DWR-Wiederauffüllversuche (2. Serie)
Techn. Bericht zum Förderungsvorhaben BMFT RS 36
Kraftwerk Union(August 1973)

Dr. Riedle, H. Gaul, H. Sarkar, H. Amm
Notkühlprogramm - ND Versuche
Ergebnis der 3. Serie der DWR-Flutversuche
Fachbericht zum Förderungsvorhaben BMFT RS 36, RS 36/1
Kraftwerk Union(August 1973)

Classification: 1.2	
Title 1 (Original Language): Notkühlprogramm - Niederdruckversuche Wiederauffüllversuche mit Berücksichtigung der Primärkreisläufe (RS 0036 B - I.1.2, Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** KWU, Erlangen
Title 2 (english): Emergency Core Cooling Program-Refilling Experiments with Simulation of the Circulation Loop	**Project Leader:** Ruthrof
Initiated (Date): January 1973 **Completed (Date):** August 31, 1976 **Status:** Continuing **Last Updating (Date):** December 1975	

General Aim

Experimental investigation of the feed back of the primary loops of a PWR on the reflood of the core.

Particular Objectives

Measuring of the thermohydraulic quantities which influence the cooling of the core, in particular flow pattern, heat transfer coefficients, flooding rates, quenching times and pressure differentials.

Experimental Facilities

In order to investigate the reflood phase in a PWR including the feedback of the complete primary system, a test facility is being build. Beside a testbundle it includes three scaled down primary loops with full height steam generator simulation.

Using a test bundle with 340 electrically heated rods of 3,9 m heated length results in a scaling factor of 1 : 134 between experiment and the reference power plant Biblis B. While all heights are simulated full size, the loop system is designed to have the same pressure drop as in the reactor.

The following tube portions have been welded and assembled:

- 6 feed lines between ball valve and primary circuit line
- U-tube (downcomer) and pre-separator
- steam outlet of the lower plenum leading to the U-tube
- separator vessel - containment line
- exhaust line for pressure control into the containment
- steam generator heating lines
- junction line between pre-separator and separator vessel
- feed pump up to the terminal box
- terminal box via orifices, control valves and ball valves
- ball valves via collector and reflood to the storage tank

Fabrication and assembly of the rod bundle container components has been continued:

- power supply sockets were provided with an Al_2O_3 layer
- internal and insulation resistance of the heating rods was measured
- assembly of the bundles completed
- welding of insulated ducts for two liquid level transducers
- double check of all thermo-couples in the bundle
- installation of pressure balance lines for protective channel/core container and core vessel/rod bundle container
- thermo-couples and current leads were hydrostatically tested
- manufacture of cooling tube containing pressure lines for density measurement in the bundle

The following work pertaining to the instrumentation of the test section was performed:

- obtain flow meter bids
- examination of thermo-couples and design of ducts
- fabrication of supporting structure to house the pressure meters for determination of the feed mass flow rates
- designing of the insulation pieces for the liquid level transducer
- calculation, construction and fabrication of the orifices to determine steam mass flow rates in the steam outlet out of the containment

The following plant components have been instrumented:

- U-tube, pump seals and primary circuit lines
- containment and separator vessel

The following measuring devices have been developed and/or tested:

- liquid level transducer
- flow meter for low flow rates
- device for density measurement in the rod bundle

Several minor tests have been conducted:

- Tests performed on a plexi-glass model led to an optimization of the hot injection
- Suitability test of a compressive pump for density measurement in the rod bundle. The pump works at low backpressures and has to be tested at higher pressures (6 bar)
- Examination of a thermo-couple radiation shield for the fluid temperature measurement in the bundle. Noticeable improvement, however, could not be achieved. Determination of power loss in the contact
- Plate area depending on the conductivity of the demineralized water. Results show that increase of the conductivity up to 10 µS/cm is permissible, i. e. an inductive flow meter can be used.

The instrumentation will provide information on heat transfer and water level rise in the bundle, temperatures and heat transfer in the steam generator and flow conditions in the loops and at the break. The data acquisition system is capable of handling up to 300 data channels at 1 Hz scanning rate.

Research Program

The test program includes the following parametric variations:

- Max. initial clad temperature: 500 to 800 °C
- System pressure: 1 - 6 (40) bars
- Max. decay heat flux: 4 to 8 W/cm^2
- Time function of decay heat: const, ANS standard
- Reflood rates: 6 - 60 cm/sec
- Split of reflood rates top/bottom: 0/1, 1/1, 2/1, 1/2
- Time function of injection rates: Const, accumulator characteristic
- Break size: 0,25 to 2 F (double ended guillotine)
- Break location: hot leg, cold leg, between SG- and pump
- Simulated pump resistance: locked rotor, free rotor
- Residual water: 0 to lower grid plate
- Loop seals: 0 to full
- Loop wall temperatures: 150 to 300 °C

Project Status / Progress to Date

The high frequency generator for soldering the heating rods was put into operation.

100 heating rods were seal soldered on the upper end, leak tested with helium; the ceramic end caps were soldered into the lower rod ends and the internal resistance measured.

Project Status / Essential Results

Installation of the tube injection lines is nearly completed. Injection rate control which is very much susceptible to oscillations was improved by interposing an evolution device and installing a new profilometer disc in the valve positioner.

The elevation regulating scan can principally be used in the rod bundle.

The flow meter for low rates (transit time principle) provides signals which can be analysed.

For flow measurement at low rates (from 10 mm/sec upwards) the ultrasonic flow meter is suitable.

Density measurement in the rod bundle container using nitrogen as flash gas has proved to be workable; the injected nitrogen rate ($3{,}6 \times 10^{-1}$ m^3/h) is unallowably high for the refilling tests. This method will be pursued using water as a transmitting agent.

Next Steps

Assembly of the bundle in the container.

Assembly of the current supply unit for the 340 heating rods.

Ducts for the thermo-couples and power lines in the pressure vessel.

Testing and calibration of plant components (U-tube, separator vessel, steam generator etc.).

Preparation of auxiliary programs in order to achieve a more effective shake down of various reactor operating sections.

Completion of the main program "PKL 01" covering data storage, control and plant control.

Relation with Other Projects

see RS 36/2 - A 74

Reference Documents / Degree of Availability

H. Kremin, R. Mandl, Dr. K. Riedle, K. Ruthrof, J. Sarkar, H. Schmidt "Wiederauffüllversuche mit Berücksichtigung der Primärkreisläufe PKL". 1. Technischer Fachbericht, Förderungsvorhaben BMFT RS 36/2 Kraftwerk Union Aktiengesellschaft (Feb. 1975)

Company Confidential.

R. Mandl
"Wiederauffüllversuche mir Berücksichtigung der Primärkreisläufe PKL, Instrumentierung der Versuchsanlage". 2. Technischer Bericht, Förderungsvorhaben BMFT RS 36 B, Kraftwerk Union Aktiengesellschaft (Jan. 1976)

Company Confidential.

Classification: 1.2	
Title 1 (Original Language): Notkühlprogramm - Niederdruckversuche SWR - 2. Doppelbündel (RS 0036 C - I.1.2, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Emergency Core Cooling Program - Low Pressure Experiments. BWR Second Cluster	Project Leader: Dr. Riedle
Initiated (Date): August 1974 Status: Continuing Completed (Date): September 1976 Last Updating (Date): December 1975	

General Aim

The project belongs to the tests, which were carried out under number RS 36. Investigations shall be executed on the behaviour of the BWR-core during emergency cooling after MCA.

As results informations are expected on heat-transfer coefficients at flooding, spraying and coupled flooding and spraying as a function of the initial temperature, pressure, power and injected flow rates.

Particular Objectives

Informations about rewetting velocity of shroud and fuel rod.
Temperature behaviour at different water levels with and without steam injection.
Long term cooling of the rods by spraying.

Experimental Facilities and Program

The research program RS 36 includes the 3 steps:

- Monotube reflood
- PWR bundle reflood
- BWR bundle reflood, bundle I

Preliminary tests on reflood were carried out on a vertical internally cooled tube (Monotube) which simulates a single reactor subchannel. Besides having the same hydraulic diameter and height of PWR subchannel, a parallel unheated tube was added for downcomer-simulation.

Preliminary reflooding tests were carried out on a single internally cooled tube. In order to include the effects of multichannel core geometry and radial distribution of power and flow a test facility for a 340-rod bundle was designed identical to those of KWU PWR's. Further tests were conducted on two parallel 7 x 7 rod bundles with BWR geometry.

Spacer geometry, heated length (3500 mm), pitch (18,75 mm), rod diameter (14,3 mm) inlet or outlet grid plate are designed identical to those of KWU BWR's.
The axial power distribution is approximated by a chopped cosine.

Project Status/Progress to Date

Experimental investigations of the series "A" were carried out, encircling the following parameters:

Quantity of water spray/bundle:	0,48; 0,68; 1,02 m^3/h
System pressure:	1, 5, 10 bar
Initial Temperature:	$T_{saturated}$

In contrary to the normal test series the container, which simulates the RPV, was not flooded initially. The heated rods were cooled by spraying, when the power was increased by steps of 5 kW/sec from 0 to 100 %. The test was cancelled when the temperature T approached $T_{saturated}$ + 50 ^oC.

The test series "B" were carried out with the following parameters:

System pressure:	1; 5; 10 bar
Initial temperature:	$T_{saturated}$ an 500 ^oC
Vapour quantity	0,05; 0,1; 0,2 kg/sec
Pressure drop:	0,01; 0,02; 0,03 bar/sec
Power:	110/150 or 220/300 kW
Cooling level:	2450; 1985; 1370; 755 mm

Some tests were carried out to determine correction factors for the flooding height, when vapour was fed in or pressure was lowered, in order to determine $c_p \cdot \varrho$ of the heater rods. The flooding height was controlled by enforced inlet or outlet of water, the desired pressure was controlled by valves.

The test series "C" were carried out with the following parameters:

Initial Temperature:	600 ° and 800 °C
System pressure:	1; 5 and 10 bar
Power:	300/220 kW
Flooding velocity:	1; 1,5; 2; 3; 3,5; 5 cm/sec
Spraying of one bundle:	0,68 and 1,03 m^3/h

About 20 % of the sprayed water wetted the containment and not the bundle. The water level was kept constant about 300 mm beneath the heater rods. Two tests were carried out with joint spraying and flooding.

Project Status/Essential Results

The test series "A" and "B" are completed. The results are plotted and stored.
The result of the tests on $c_p \cdot \varrho$ was lower than expected theoretically (~ 5 %).

The test series "C" are completed and evaluated. During spraying the power was decreased about 40 % when a temperature of 925 °C was reached. The result was, that the heat transfer during spraying is better at higher pressure load compared with 1 bar.

Next Steps

Additional tests as proposed by the SK "Notkühlung" : comparative tests, evaluation of heat transfer coefficients from the test results, investigation of parameter influences.

Relation with Other Projects

see RS 36.

Reference Documents/Degree of Availability

E. Hicken, K. Riedle
"Bubble rise velocity and heat transfer in a vertical rod bundle"
Paper presented at the Meeting of the "European Two-Phase Flow Group"
at Haifa, June 2 - 6, 1975

H.-P. Gaul, E. Hicken
H. Loser, K. Riedle
K. Ruthrof, J. Sarkar
"Wärmeübergang an stagnierendes Fluid in zwei SWR-Bündeln unterschiedlicher Leistung"
Paper presented at Gemeinsame Fachtagung der Fachgruppen "Reaktorsicherheit" und "Thermo- und Fluiddynamik" der Kerntechnischen Gesellschaft. 28. - 30. Januar 1975

K. Riedle
"German reflood heat transfer program"
Presented at the 3rd Water Reactor Safety Research Meeting
Gaithersburg, Md. Sept. 29, 1975

Classification: 1.2	
Title 1 (Original Language): Theoretische Untersuchungen zur Niederdruck- und Wiederauffüllphase der Kernnotkühlung (ATT 085 A - I.1.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (english): Low Pressure and Refilling Phase of Emergency Core Cooling	Project Leader: Dr. H. Karwat, Dr. A.B. Wahba
Initiated (Date): — Completed (Date): Status: continuing — Last Updating (Date): December 1975	

1. General Aim

The general aim of this project is the development of computer codes to predict the thermal and hydraulic response of water cooled reactors during the refilling phase of a loss-of-coolant accident (LOCA).

2. Particular Objectives

One main objective is to develop a code which is able to simulate efficiently the flooding of the core and the various physical phenomena involved in this process.

Besides this development the application of existing codes is continued in order to indicate which particular problem will need special consideration in the new code.

From the analytical verification of related experiments additional information is expected to support the analytical work.

3. Experimental Facilities and Program

Not relevant.

4. Project Status

The activities during this year were concentrated on four points:

- Application of the US-Code RELAP-4 /1/, to study the behaviour of a light water reactor during the final phase of blowdown.
- Application of the KWU-Code WAK-1, to study the influence of different parameters on the refilling process.
- Drafting the main features of a basic version of LRA-Code for the flooding of the core.
- Evaluation of the rewetting experiments on a highly heated tube (RS 62).

4.1 Progress to Date

In order to study the physical phenomena during the final phase of blowdown of a pressurized water reactor, a nodalization of the whole primary loop had to be done. For an improved simulation of important components like steam generators, pressurizer, containment and emergency core cooling injection devices a nodalization with 16 volumes and 21 junctions was chosen. Several computer runs were performed to test the performance of the actual RELAP-4 version with respect to:

- Amount of water left in the loop after blowdown, comparing the separation model for the two phases with homogeneous assumptions.
- Efficiency of core cooling injection into both hot and cold leg.
- Response of the fluiddynamics in the primary loop on pressure increase in the containment near the end of blowdown.

Using the refill program WAK-1, the influence of the following input parameters on the refilling time was studied:

- The interaction between steam from the core and cold injected water in the upper plenum is accounted for by a condensation efficiency parameter.
- The motion of the upper quench front is essentially dependent on the input table for the axial distribution of the average surface temperature and on the input value of the Leidenfrost temperature.

For the evaluation of the rewetting experiments of a heated tube, done by the KWU (RS 62), program development was carried on. For example DATEBG was developed to determine and plot the experimental variables from the measured signals. From the measured fluid temperature at a certain axial position the history of the heat transfer coefficient was determined using the evaluation model DIFARZ /2/. Other variables like the velocity of the quench front were also determined from the measured quantities.

4.2 Essential Results

Computation results of the 16 volume PWR example using RELAP-4/002 Vers. 8 has shown:

- The use of the separation model in the core region leads to strong oscillation in the flow rates. These oscillations caused a considerable increase in computation time.
- The simulation of hot injection into the volume representing the upper plenum resulted in an unreasonable increase in the water mass in this volume (Sandwich formation).

The parameter study using the refilling program WAK-1 has shown that the flooding time of the core increases rapidly when the condensation efficiency is decreased.

Most of the rewetting experiments done by KWU on a highly heated tube were evaluated using the data reduction code DATABG and the evaluation model DIFARZ. The results of this evaluation are given in /4/. An acceleration of the quench front due to an increase in the flooding velocity as well as due to an increase in the subcooling of the inlet water was observed.

5. Next Steps

- The development of a refill program is now concentrated on the simulation of the flooding process in the core. Detailed calculation of the transient thermal behaviour of an average pin should accompany the flooding process. Accurate information of the transient cladding temperature together with the Leidenfrost criterion for the re-wetting of the clad will lead to the separation of the quench front from the water front. Work is going on to simulate the different heat transfer regimes during the flooding process /3/.

- Improvement of the evaluation model DIFARZ to generalize its application to related experimental programs.

6. Relation with Other Projects

The work has strong relations with refilling and emergency core cooling experiments of light water reactors carried out within the projects RS 36 and RS 62.

7. Reference Documents

/1/
K.V. Moore, W.H. Rettig
RELAP-4 - A Computer Program for Transient Thermal Hydraulic Analysis
ANCR-1127, Rev. 1

/2/
J. Simon-Weidner
Beschreibung eines eindimensionalen Auswertungsprogramms DIFARZ zur Bestimmung von Wärmeübergangszahlen aus gemessenen Temperaturen. Interner Bericht
MRR-I-31, Oktober 1974

/3/
P. Schally, V. Teschendorff
BESAM-2 - Ein Modell zur Berechnung der Temperaturverteilung in einem Reaktorbrennstab. Interner Bericht
MRR-I-55, Dezember 1975

/4/
A. Berning, A.B. Wahba
Versuche zur Wiederbenetzung hochaufgeheizter Rohre - Auswertung von

Versuchsergebnissen. Interner Bericht
MRR-I-58, Dezember 1975

/5/
Quarterly Reports in the Series IRS-Forschungsberichte

8. Degree of Availability

Documents are available through
Laboratorium für Reaktorregelung und Anlagensicherung
D-8046 Garching
Federal Republic of Germany

Reports MRR-I are confidential and therefore normally not available.

Classification: 1.2	
Title 1 (Original Language): Untersuchungen zur Hydraulik des Flutvorgangs und zu bisher noch unberücksichtigten Einflußgrößen beim Wiederbenetzen (RS 184 - I.1.2; A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Investigations on the Influence of the Hydraulic during Reflooding	Project Leader: H. Hein
Initiated (Date): . 10. 75 — Completed (Date): 30. 9. 77	
Status: Continuing — Last Updating (Date): 31. 12. 1975	

General Aim

In order to improve the reflooding model, the hydraulic effects during reflooding will be studied in detail.

Particular Objectives

A detailed knowledge of the hydraulics in a channel during the reflooding and the resulting flow pattern shall improve the calculation of heat transfer in the unwetted area. With the help of this experiment criteria will be worked out for the transition of vapour - to fog flow and from fog flow - to film boiling in order to get more information of the extension of different heat transfer regions.

The coupling of the rewetting model based on heat conduction in the wall with the hydraulics of the channel is a presupposition for a general applicability of theoretical calculations.

Experimental Facilities

For these experiments the testrig used for the program RS 62 will be modified for getting more detailed information on the hydraulics during the reflood period. Also for these experiments constant inlet conditions will be focused.

For special tests an annular test section with a quarz-glass tube for the outer wall will be used.

Research Program

For three different hydraulic diameters rewetting experiments will be carried out variing the parameters initial wall temperature, inlet subcooling, supply velocity and system pressure. Also the ratio stored heat to the water content within the channel will be varied.

With the annular test section a comparison will be made for the advancing of the rewetting front for Zry- and stainless steel channings.

To improve the rewetting model information is needed on the rewetting temperature and on the effects near the rewetting front.

Project Status/Progress to Date

Some discussions have been arranged in order to improve the instrumentation for measuring of local density and flow conditions for an inside flooded tube.

For comparision of the rewetting of stainless steel and Zry tubes the existing rig with an annular test section was extended and the test tubes were installed and instrumented.

Project Status/Essential Results

First results for rewetting experiments with the Zry- and the stainless steel tube showed that the rewetting front with the Zry canning is about twice faster than with the stainless steel tube.

Next Steps

A literature study will be conducted concerning the hydraulics during reflooding, especially on the subject in what kind the flow velocity, phase and density distribution and other parameters are considered in the existing codes.

For the experimental investigations the most appropriate instrumentation has to be selected. After this some reflooding tests will be run with tubes of different internal diameters. In the annular test section the tests with Zry-tubes will be conducted. Finally the detailed investigations of the surroundings of the rewetting front will be prepared.

Relation with Other Projects

RS 62 Experiments for the Establishment of a Theory of the Rewetting of Highly Heated-up Rods by Pipe Experiments

RS 36 Emergency Core Cooling Program - Low Pressure Experiments

Reference Documents/Degree of Availability

No reports available.

Classification 1.2

Title 1 Performance of Spray Cooling	**Country** UK
Title 2	**Sponsor** UKAEA **Organisation** Winfrith Laboratory
Initiated 1968 **Completed** **Status** Continuing **Last updating**	**Project leaders**

1 General Aim

To optimise spray cooling and determine safe fuel ratings.

2 Particular Objectives

To measure heat transfer coefficients and quenching times in a way suitable for use in calculating reactor blow-down transients.

3 Experimental Facilities

The High Pressure and the Low Pressure Emergency Spray Cooling rigs at Winfrith.

4 Project Status

From thermocouple results of blowdowns, heat transfer coefficients have been correlated with spray cooling flow rate; radiation characteristics (emissivity etc.) pressure; spray sub-cooling etc.

Next Steps

Work to date has been with deliberately contrived flow stagnation in the channels; some flow will be super-posed. Further attempts will be made to optimise (speed up) quenching.

Reference Documents.

Internal documents.

CLASSIFICATION 1.2

Title 1: L.O.C.A. Computations Title 2: -

Initiated: 1st January 1975

Status: Progressing Last Updating: 1st January 1976

Country: United Kingdom Sponsor: UK - NII

Organisation: Strathclyde University

Project Leaders: H C Simpson, D H Rooney

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1. General Aims:

To become familiar with RELAP computer codes and to check the sensitivity of the procedures and correlations contained therein.

2. Particular Objectives:

To check the codes against depressurisation and transient two-phase flow experiments carried out at Strathclyde.

3. Experimental Facilities and Programme:

Three circulating test loops available

(i) steam water, vertical tubes, pressures up to 70 bar

(ii) Freon 113, vertical and horizontal, transparent pipework, pressures just above atmospheric.

Tube sizes 50 to 100 mm diameter. Data recorded include depressurisation rate, pressure differences, mass velocity and void fractions. Non equilibrium effects also being studied.

4. Project Status:

(1) Data available from steam-water rig. Relap 4 code programmed for test rig and preliminary results obtained.

(2) Work started on modifying Relap 4 to operate with Freon liquid-vapour mixtures.

5. Next Steps:

(i) Comparison of total data from steam water rig with Relap 4 predictions

(ii) Comparison of data from Freon 113 test rigs with Relap 4 predictions

Classification 1.2 cont.

6. Relation with Other Projects

Working in conjunction with projects at National Engineering Laboratory and Manchester University through N.I.I.

7. Reference Documents:

Reports pending

7. Degree of Availability:

By application to NII

1. Budget:

Around £10,000 per annum

2. Personnel:

Professor H C Simpson	- Academic Staff, Part-time on project
Dr D H Rooney	- Academic Staff, Part-time on project
Mr T M S Callander	- Academic Staff, Part-time on project
Mr R O'Mahoney	- Research Fellow, Full-time on project

Several Postgraduate Students

CLASSIFICATION 1.2

Title 1: P.W.R. Refill Studies

Title 2: -

Initiated: 1st November 1975

Completed: -

Status: Progressing

Last Updated: -

Country: United Kingdom

Sponsor: UK - NII

Organisation: Strathclyde University

Project Leaders : H C Simpson, D H Rooney

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1. General Aims:

To simulate the refill process in a P.W.R. downcomer and study its effectiveness.

2. Particular Objectives:

To produce a theoretical model, or correlation, defining the refill process enabling the limiting conditions to be predicted.

3. Experimental Facilities and Programme:

Work to be carried out in three phases. Phase 1 is developed annulus with tangential water injection. Phase 2 is developed annulus with normal water injection. Phase 3 is 1/10 scale model of P.W.R. downcomer. All test sections transparent, fluids steam and water, pressures just above atmospheric. Measurements to be taken include steam and water flowrates, pressures, temperature distributions. Cine photography to capture liquid bridging effects.

4. Project Status:

Phase 1 data being collected.

5. Next Steps:

Production of theoretical model

6. Relation with Other Project:

Similar in some respects to the Wallis work at Creare.

Classification 1.2
PWR Refill Studies

7. Reference Documents:

Reports pending

8. Degree of Availability:

By application to NII

1. Budget:

£8700

2. Personnel:

Professor H C Simpson	- Academic Staff, Part-time on project
Dr D H Rooney	- Academic Staff, Part-time on project
Mr T Campbell (Ph.D. Student)	- Full-time on project

PROJECT TITLE : Bottom flooding ECCS Sistema di raffreddamento di emergenza per allagamento dal basso.	CLASSIFICATION 1.2.
SPONSORING COUNTRY : Italy	ORGANISATION : CISE sponsored by CNEN
DATE INITIATED : 1971 DATE COMPLETED : 1977	PROJECT LEADER : At present under responsibility of UTM (CISE).

Status (latest updating): May 1975

Description :

1. General aim: to predict the performance of bottom flooding ECCS in pressure tube reactors.

2. Particular objective: understanding of basic phenomena involved in bottom flooding ECCS in water reactors.

3. Experimental facilities and programme

3.1. Experimental facilities

- IETI-1: (see N1.1.2)for scaled-down experiments
- REM : for full-scale experiments; flowrate 2,8 kg/s; pressure 10 bars; heating power 300 kW.

3.2. Programme

3.2.1. Preliminary scaled-down tests relevant to tubular and annular geometry.

3.2.2. Full-scale experiments adopting an indirectly heated 19-rod bundle.

3.2.3. Code development for fuel rod temperature predictions.

4. Project status

4.1. Progress to date (with reference to the above programme)

- (3.2.1.): First set of tests completed;
- (3.2.2.): Construction stage completed;
- (3.2.3.): Preliminary code developed; a more sophisticated version (2nd version) initiated.

4.2. Essential results

- basic understanding of the physical phenomena involved;
- set up of the experimental procedures and techniques.

5. Next steps

- Full-scale experiments
- Further scaled-down experiments
- Completion of the 2nd version code.

6. Reference documents

1) Martini R., Premoli A. "A simple model for predicting E.C.C. transients in bottom flooding conditions" CREST Meeting - Munich, October 18-20, 1972.

7. Degree of availability: to a limited extent

Classification
1.2

Title 1 Transient boiling heat transfer in emergency core cooling conditions	Country : JRC Sponsor : CEC Organization: ~~JRC ISPRA~~ Establishment
Initiated : 1974 Completed : December 1976 Status : progressing Last updating : March 1975	Project leader: E. Burck

1.) General aim

Investigation and visualisation of transient boiling conditions

2.) Particular objectives

To study the transient boiling conditions in the pressure range 1-20 bars for several quenching body shapes, inlet subcooling conditions and initial temperatures between 200 and 800°C (which covers the whole interesting range for fuel rod and pressure vessel flooding).

3.) Experimental facilities and programme

Quenching facility with flooding and expansion vessel. The characteristics of this facility are :

- flooding velocities : 1-37 cm/s
- system pressure : 1-20 bar
- cooling water temperature : 20-210°C
- initial surface temperature : 200-800°C

4.) <u>Project status</u>

1.) <u>Progress to date</u> : The construction of the Quenching Facility has been completed in 1974. The final instrumentation and calibration of the facility if foreseen for January - March 1975.

2.) <u>Essential results</u> : The latest theoretical and experimental literature in this field has been investigated in preparation for the interpretation of the experimental results and the choice of parameters to be investigated. The problem of the determination of the transient surface temperatures and heat fluxes was overcome by applying inverse heat conduction analysis with temperature dependent physical properties.

5.) <u>Next steps</u> : Experimental investigation of the different flooding conditions.

6.) <u>Relation with other projects</u> : The programme has been planned so as to be complimentary to other work in the quenching field.

7.) <u>Reference documents</u> :

1.) JRC safety programme progress report 1974.
2.) H. Lauer, Numerical solutions of the inverse one-dimensional transient heat condution equation and their application to transient boiling problems. Atke <u>24</u>,(3),p.215, 1974
3.) E. Burck, W. Hufschmidt, E.De Clercq, Instationäre Wärmeübertragung beim Sieden von Wasser an der senkrechten Wand eines Reaktordruckbehälters. Atke <u>21</u>, (2), pp 127-135, 1973.

8.) <u>Degree of availability</u> : Freely available

9.) <u>Budget</u> : The expected total investment from the CEC is 65 000 UA which includes the cost of the facility and the running costs.

10.) <u>Personnel</u> : 5 men/year

11.) <u>Additional information</u> : -

Classification: 1.2

Title 1 Level Swell Reflood Heat Transfer	**Country** BRD (U.S.A) **Sponsor** Babcock and Wilcox Proprietary
Title 2	**Organisation** BBR Mannheim
Initiated: Jan. 1975 **Completed:** **Status:** **Last updating:** Dec.1975	**Project leaders** Dr. B. E. Bingham R. T. Bailey

1. General aim

To determine reflood characteristics of a B & W "vent valve" plant and to evaluate the effectiveness of level swell cooling when the water inventory is insufficient to cover the core entirely with a two-phase froth.

2. Particular Objectives

3. Experimental Facility and Programme

In 1975, reflood and level swell tests were conducted with an electrically heated 56 tube full length bundle with an axial power profile peaked at the ten foot elevation.
Level swell experiments were conducted in two modes:

1) a constant water inventory is maintained by establishing a make-up flow equal to the steaming rate;
2) the water is allowed to boil off, progressively uncovering the bundle.

The ranges of system parameters covered by the investigation are:

Average Linear Power, kw/ft	0.1-0.3
Pressure, psia	20 - 180
Inlet Subcooling, $(h_f-h)/\frac{Btu}{lb}$	0 - 175

Thirty forced flooding experiments were conducted using this facility covering the following ranges of system parameters:

Peak Linear Power, kw/ft	0.5-1.0
Flooding Rate, in/sec	1.0-3.0
Pressure, psia	25-60
Inlet Subcooling, ^{o}F	0-140
Maximum Initial Temperature, ^{o}F	800-1400

4. Project Status

4.1 Progress to date

The experimental phase has been completed.
Evaluation of the data is in progress.

4.2 Essential results

Level Swell: For a given core water inventory the swell level increases with power density and inlet enthalpy and decreases with pressure. Within the range of experimental variables, the Wilson bubble rise model predicts reasonably accurate void distributions. The computer code FOAM2, used to calculate the swell level and steaming rates during the quiscent period of small break LOCA's, has been shown to be an accurate formulation of the phenomenon.

Reflood: still under review.

5. Next Steps

During 1976, the level swell/reflood program will be continued using a single tube. System parameters affecting the reflood and level swell phenomenon that will be investigated are:

1) Hydraulic diameter of flow channel
2) Power profile (uniform, symmetric, inlet and exit peaks)
3) Chemical additives

6. Relation With Other Projects

7. Reference Documents

8. Degree of Availability

Proprietary
(It is intended to publicly release the level swell results during the third quarter of 1976).

Classification 1.2	
Title 1 FLECHT - Low Flooding Rate Test Program (Full Length Emergency Cooling Heat Transfer)	COUNTRY Belgium (USA) SPONSOR ORGANIZATION : Westinghouse Nuclear Europe
Title 2	**PROJECT LEADER**
Initiated May 1974 — **Completed** Nov, 1975 **Status** start testing Dec, 1974 — **Last updating** Dec 12,1974	**SCIENTISTS**

1. GENERAL AIM

The general objective of the FLECHT test program is to obtain experimental data for use in evaluating the heat transfer capabilities of a PWR Emergency Core Cooling System during a postulated loss-of-coolant accident.

2. PARTICULAR OBJECTIVES

The objectives of the tests to be conducted in the modified FLECHT test configuration are to supplement the parametric effects studied in the original FLECHT program, and to provide heat transfer coefficient and entrainment data at flooding rates of 1 in/sec and below. The forced flooding tests will be conducted with rod bundles having a cosine and a skewed axial power profile.

3. EXPERIMENTAL FACILITY

The FLECHT-SET test facility will be modified to conduct forced flooding tests as shown in Figure 1.

The modified facility consists of :

a) The original FLECHT test section housing with baffle installed in the upper plenum exhaust to improve liquid carryover separation.

b) The 10 x 10 rod bundle and related existing instrumentation including the ANC liquid level transmitter.

c) The existing pressurized water supply accumulator and injection line with three rotameters injection rates from 0.5 to 12 in/sec.

d) A close coupled carryover tank connected to the test section upper plenum.

e) A commercially available steam separator with a capacity of 2500 lbs/hr, and a liquid collection tank to collect liquid entrained in the exhaust steam.

f) Exhaust piping with a system pressure control valve and an orifice plate flow meter to mesure steam flow rate.

4. PROJECT STATUS

a) Progress to-date :

Modifications to the test facility have been completed, and shakedown testing has been started with the cosine axial power profile rod bundle.

b) Results : None

5. NEXT STEPS

Complete testing with a rod having a cosine axial power profile in April, 1975.

Complete testing with a rod bundle having a skewed axial power profile in November 1975.

6. RELATION WITH OTHER PROJECTS

This program is related to all other Emergency Core Cooling System Test Programs such as :

Delayed DNB
UHI

Blowdown, Refill and Reflood
FLECHT-SET

7. REFERENCE DOCUMENTS

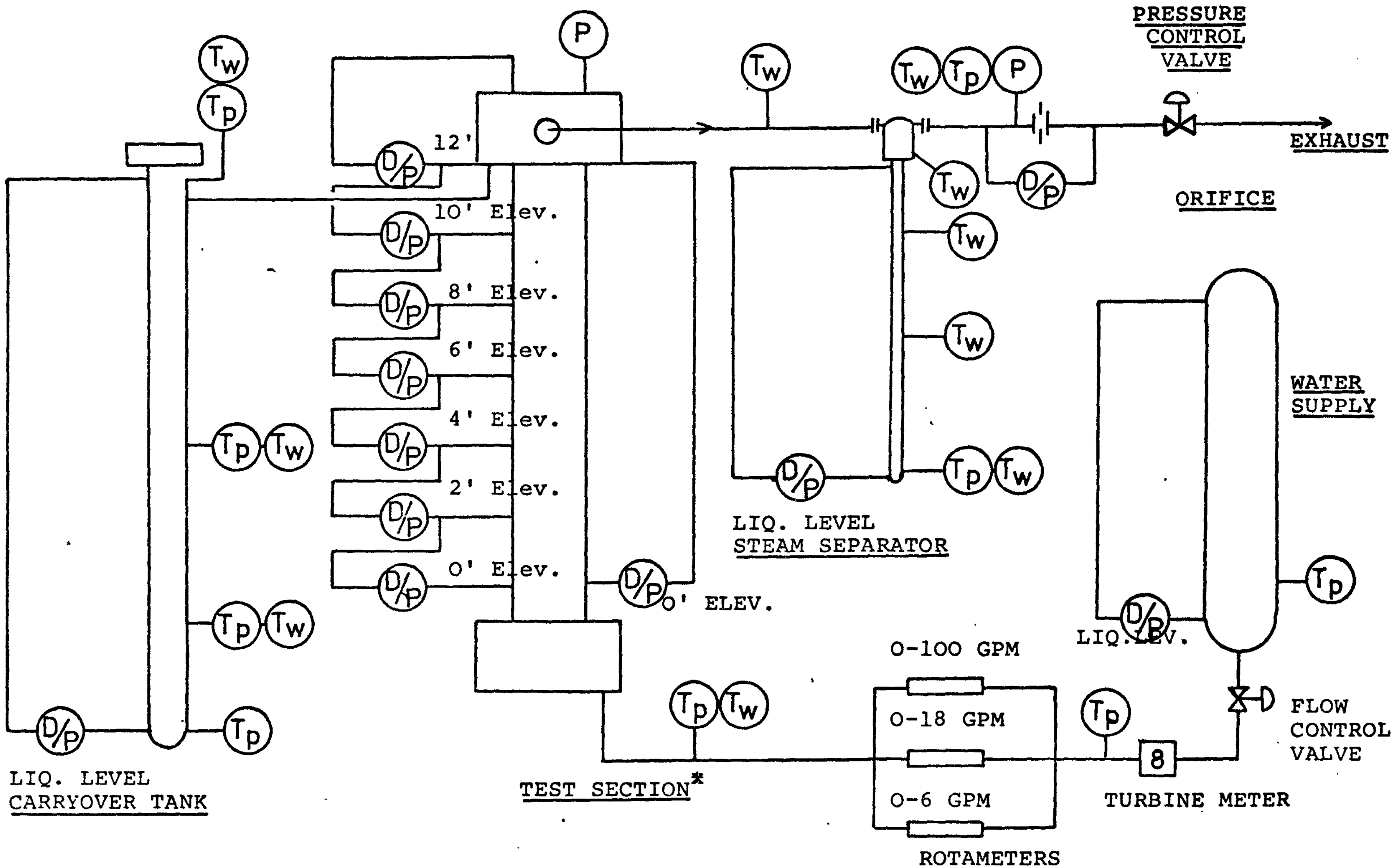
a) WCAP-7665 - PWR FLECHT Final Report, April, 1971.

b) WCAP-7931 - PWR FLECHT Final Report Supplement, October 1972.

8. DEGREE OF AVAILABILITY

Available upon request.

FLECHT LOW FLOODING RATE TEST CONFIGURATION



* ALL INSTRUMENTATION IS NOT SHOWN

Classification 1.2	
Title 1 FLECHT SET Full Length Emercency Cooling Heat Transfer Systems Effect Tests.	COUNTRY Belgium (USA) SPONSOR ORGANIZATION : Westinghouse Nuclear Europe
Title 2	PROJECT LEADER SCIENTISTS :
Initiated (date) 7/30/74 Completed : Status : Last updating	

FLECHT : FLECHT-SET

(Full Length Emergency Cooling Heat Transfer
System Effects Tests)

1. GENERAL AIM

Following a primary system loss-of-coolant accident, the system would rapidly depressurize. The loss of coolant may partially or wholly uncover the reactor core. The Emergency Core Cooling System is provided to rapidly reflood the reactor vessel under such conditions, and ensures that any damage to the core does not lead to any unacceptable consequences either in the plant or off-site.

The original FLECHT series of tests were designed as separate effects type tests to investigate the reflood heat transfer history of hot fuel rods in the core during the reflood phase of a LOCA. The reports of this series of tests are given in References 1-4.

2. PARTICULAR OBJECTIVES

FLECHT-SET is a continuation of the FLECHT bottom flooding test except that the effects of the system volumes, resistances, elevations and other heat inputs are modeled to obtain the system feedback on the flooding rate and heat transfer. The program will consist of two phases. Each phase is intended to simulate a 4 loop PWR with various degrees of sophistication. Details on each are included in subsequent sections.

3. EXPERIMENTAL FACILITIES AND PROGRAM

Experimental facility is illustrated by figure 1 and is described in references 5 and 6.

The program is divided in 2 steps :

- PHASE A consisting of scoping tests (1 loop no steam generator)

- PHASE B including a more complete systems effect simulation (2 loop steam generator simulation).

4. PROJECT STATUS

4.1. Programs to-date

Phase A consisted of a set of early scoping tests employing a simplified 1 loop system simulation without a steam generator (long lead item). The simplification (1 loop representing 4 loops) is considered necessary in order to measure flood rate and particularly the test section effluent two phase flow rate. Without the steam generator producing single phase flow at its exit, this is not measurable with standard orifice measuring techniques. Hence a simple system devised which separates, collects, and measures test section liquid effluent, then heats the remaining steam to saturation or above, thereby allowing a meaningful single phase orifice flow measurement. The liquid carryover is separated and collected at a measured rate (at the steam generator location) prior to passing through the largest flow resistance of the loop. A high quality mixture ($x > .95$) then enters a 24 ft. length of heated pipe where any remaining liquid is vaporized prior to passing through the loop orifice. Since the flow through the calibrated orifice is single phase, the flow rate can be determined by measuring the pressure drop and upstream temperature and pressure. A total effluent flow rate and quality can be calculated from the collection rate of liquid and the flow rate through the orifice.

The test in this configuration are complete and a data/analysis report has been issued (reference 5). The general result found from these tests was that the variable flow into the test assembly, caused by the system response during reflooding, yielded higher heat transfer than that which would be calculated using the FLECHT heat transfer correlation and the calculated flooding rate.

Phase B is intended to be a more complete systems effect simulation of a PWR 4 loop plant and 1 broken loop and 3 unbroken loops, including steam generator heat addition and elevation effects. Since the steam generators superheat the test section effluent, meaningful.orifice flow measurements can be made downstream of the steam generators using the loop orifice. The FLECHT-SET phase B loop drawing is given in Figure 1. The system is described in detail in reference 6.

A total of 35 phase B tests have been completed including facility shakedown tests and repeat tests. Of these tests, 20 will be reported in a data report and will be separately analyzed in a data evaluation report.

4.2. Essential Results

The same general trends observed in Phase A were also observed in Phase B ; the variable bundle flooding rate resulted in higher heat transfer than that calculated by the FLECHT correlation.

Several questions have been raised on the scaling logic used to design the FLECHT-SET facility. The AEC critically reviewed the facility and has issued a task force report on the facility. In general, they either agreed with the design or suggested modifications which would make the scaling logic more exact. The AEC was particularly concerned about the observed large oscillations which occurred at the beginning of reflood. The Phase A data indicated that the large oscillations were caused by the rapid heat release from the test section housing. Since the rate of heat release could not be controlled from the housing, (although the time integral of the heat release could be controlled), the majority of the Phase B tests were conducted with the housing heated to the fluid saturation temperature such that the housing heat release was minimized.

5. NEXT STEPS

With the issuance of the new ECCS criteria, the AEC has re-evaluated its reflooding heat transfer requirements and has requested that the systems effects tests stop and that the FLECHT-SET facility be converted into a forced flooding heat transfer facility such that specific reflood heat transfer questions identified in the new criteria could be examined. The FLECHT-SET testing has stopped and the facility is being converted to a forced flooding mode of operation and tests in this configuration are scheduled to begin in December 1974.

6. RELATION WITH OTHER PROJECTS

This program was in the line of other ECCS programs on the post blowdown phenomena like FLECHT, STEAM WATER MIXING...

7. REFERENCE DOCUMENTS

1. J.O. Cermak, A.S. Kitzes, F.F. Cadek, R.H. Leyse, and D.P. Dominicis, "PWR Full Length Emergency Core Heat Transfer (FLECHT) Group 1 Test Report", WCAP-7435, January 1970.

2. F.F. Cadek, D.P. Dominicis, and R.H. Leyse, "PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group II Test Report", WCAP-7544, September 1970

3. F.F. Cadek, D.P. Dominicis, and R.H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report", WCAP-7665, May 1971.

4. F.F. Cadek, D.P. Dominicis, H.C. Yeh and R.H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report Supplement", WCAP-7931, September 1972.

5. J.A. Blaisdell, L.E. Hochreiter, J.P. Waring, "PWR FLECHT-SET Phase A Report", WCAP-8238, December 1973.

6. W.F. Cleary, et, al., "FLECHT-SET Phase B System Design Description", WCAP 8410, 1974.

8. Degree of availability

Available upon request.

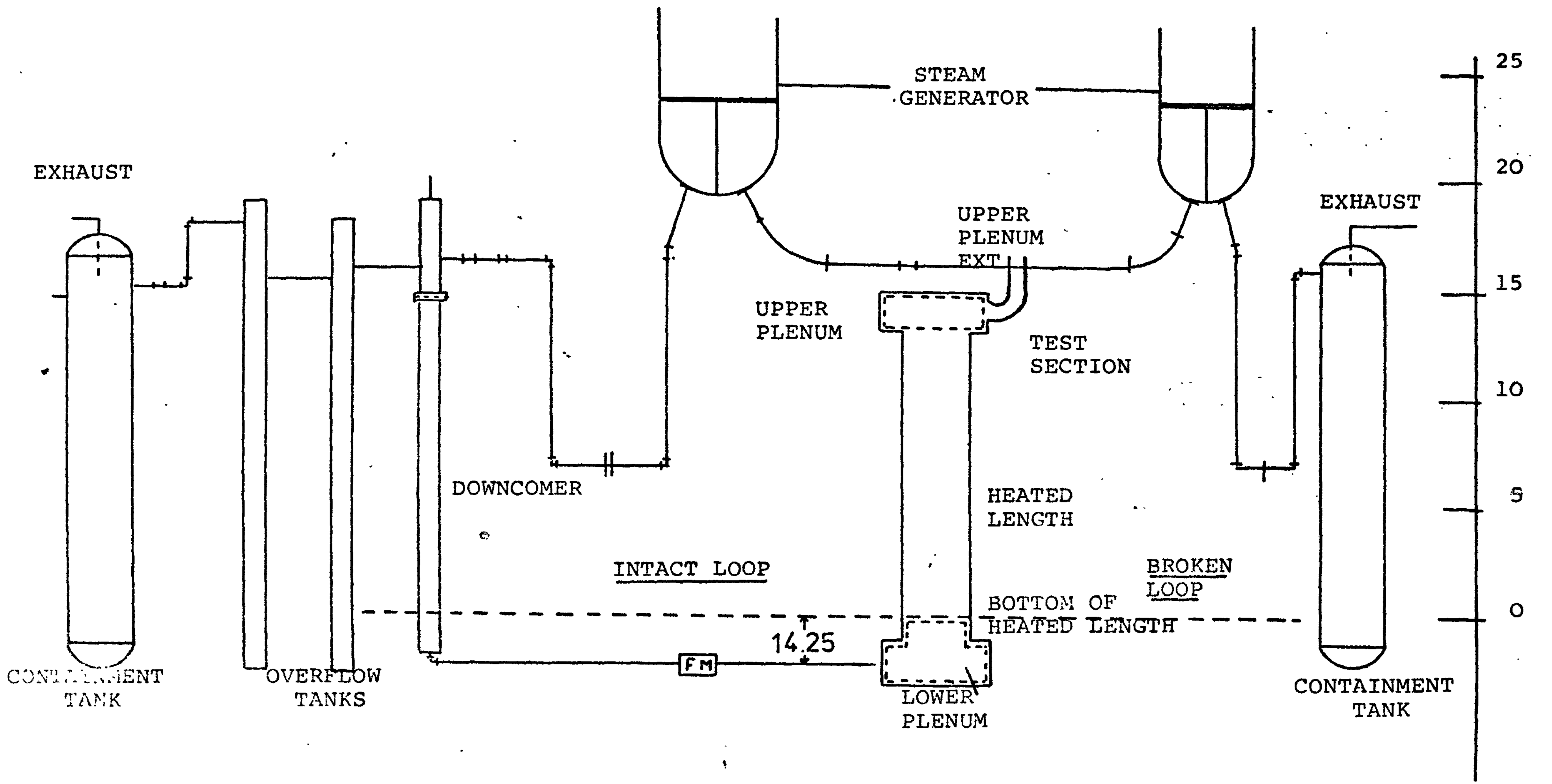


FIGURE 1

FLECHT-SET PHASE B

Classification 1.2	
Title 1 Steam Water Mixing Tests.	COUNTRY Belgium (USA)
	SPONSOR
	ORGANIZATION : Westinghouse Nuclear Europe.
Title 2	PROJECT LEADER SCIENTISTS
Initiated Completed Status Last Updating	

1. GENERAL AIM

During a LOCA, the effects of venting steam with cold water accumulator and safety pump injection are not quantified. In order to calculate the steam flows vented through the cold leg, the effective resistances must be determined experimentally.

2. PARTICULAR OBJECTIVES

The AEC interim criteria states in part :

1. "No steam flow shall be permitted in intact loops during the time period that accumulators are injecting".

2. "All effects of cold injection water, in either a hot or cold leg, in steam flow (and ΔP) should be included in the calculation".

The intent of the steam/water mixing program is to relax these overly conservative design criteria by obtaining pressure drop data during cold water injection for use in blowdown and reflood codes.

3. EXPERIMENTAL FACILITIES AND PROGRAM

Tests were conducted at approximate conditions expected to exist during and after blowdown. Table 1 presents a list of the important parameters and their ranges.

The test sections represent scaled segments (length to diameter ratio is constant) of the piping between the reactor coolant pump and the reactor vessel. The full PWR primary coolant loop resistance is also simulated.

Surge tanks at either end allow a constant pressure drop to be set across the loop, representing a fixed downcomer head. The steam flow resulting from this fixed driving force was measured. A typical test setup is pictured in Figure 1.

The effect of scale was studied to extend the test results to a full scale PWR. Tests have been run at 1/14 and 1/3 scale. Tests were also run with and without the full length cold leg extension pictured in Figure 1 for the 1/3 scale test section.

Instrumentation included density measurement by a low energy X-ray attenuation technique, as well as temperatures, pressures and pressure drops.

This work was performed by Westinghouse at the Canadian Westinghouse Laboratories in Hamilton, Ontario, Canada. This program has been submitted to EPRI (Electrical Power Research Institute) for cooperative funding.

4. PROJECT STATUS

Progress to-date and essential results.

A series of tests have been completed at 1/14 scale with injection angles of 90°, 60° and 45° in both the accumulator and SIS phase of reflood. Test section pressure drops in the accumulator range can be predicted reasonably well with a simple model based on one-dimensional momentum considerations. For 90°, the effect of accumulator injection is to decrease test loop steam venting capability by 5 to 30% from the no-injection case. For 45° injection, the steam venting capability is increased due to the pumping action of the angled injection. For the SIS range of flow rates, cold leg injection has a very minor effect on overall loop resistance.

The 1/14 scale simulated blowdown tests have been performed and the pressure drop data was found to agree reasonably well with the one-dimensional momentum prediction. Density measurements indicated that the two-phase flow was nearly homogeneous during the higher pressure blowdown tests.

Density measurements have also been obtained for both the high ECC flow (accumulator) and low ECC flow (pumped injection) portion of the reflood transient. The pressure oscillations which were observed on the pressure transducers was found to be caused by oscillating flow. The oscillating flow behaviour was observed on both the density and thermocouple readings. The 1/14 scale report shall be issued shortly.

The 1/3 scale tests and data analysis is complete and the report is presently being published. The 1/3 scale tests showed similar behavior but more scatter as compared with the 1/14 scale data. The same model which was used to represent the steady cold leg pressure drop data for the 1/14 scale tests will also represent the 1/3 scale data if the upper bound limit is increased to 1 psia. Scale effects were observed in the 1/3 scale tests, however, they can be included in the 1 psia upper bound on the data.

5. NEXT STEPS

EPRI has indicated that they would require additional testing, these requirements are now being determined.

6. RELATION WITH OTHER PROJECTS

This program is related to all other ECCS programs that aim to a better understanding of the post blowdown transient such as FLECHT, FLECHT-SET ...

TABLE 1

COLD LEG STEAM/WATER MIXING TESTS

Parameter	Range
System Pressure	45 to 20 psia
Cold Leg Steam Velocity	50 to 400 ft/sec
Cold Leg Steam Quality	60% - 300 psia to 550°F - 20 psia
Water Injection Velocity	1 to 90 ft/sec
Accumulator Water Temp.	80 to 150°F
Water Injection Angle	45° 60° 90°

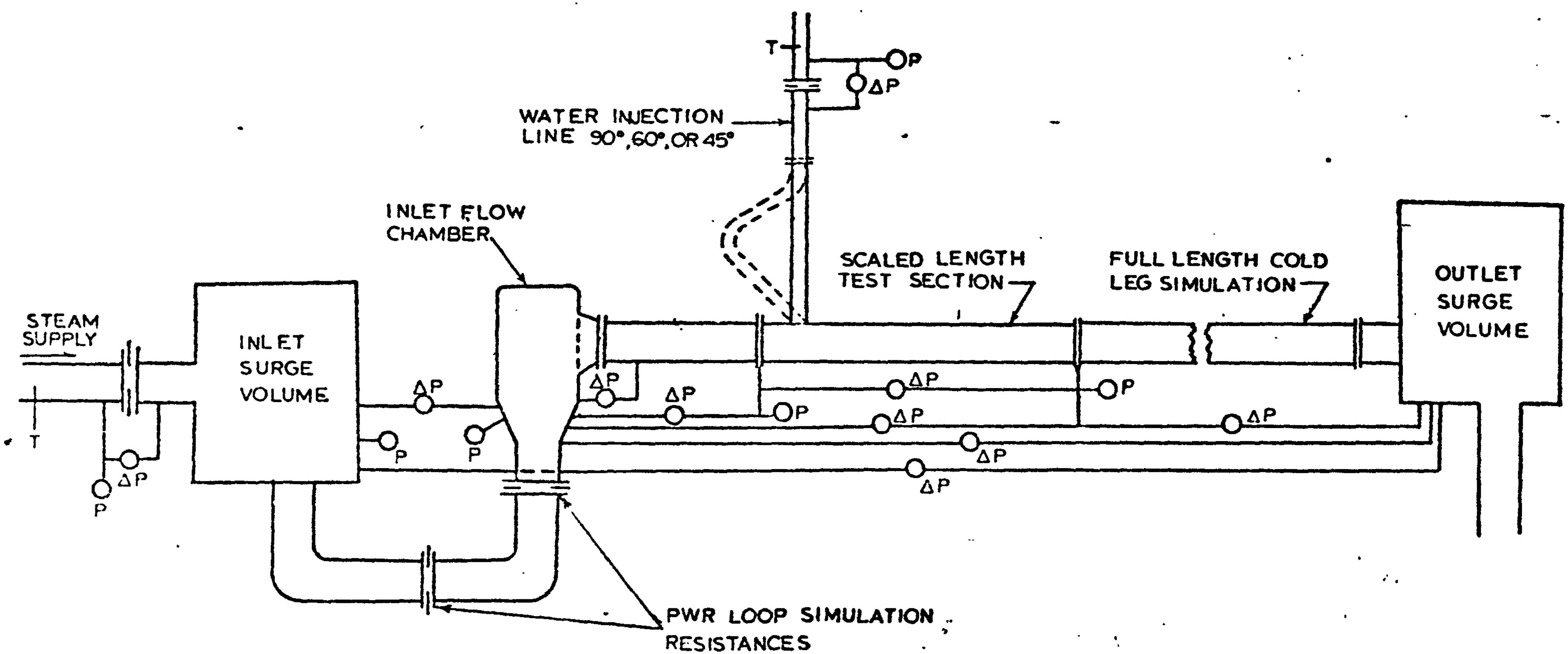


FIGURE 1
STEAM-WATER MIXING TEST CONFIGURATION
SHOWING PRESSURE AND FLOW INSTRUMENTATION

Classification: 1.3	
Title 1 (Original Language): Verhalten von Zry-4-Hüllrohren unter den bei Kühlmittel-Verlust-Störfällen auftretenden Beanspruchungen (RS 107-I.1.3, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Behaviour of Zry-4 Canning Tubes under Loss-of-Coolant-Accident Conditions	Project Leader: H.-J. Romeiser
Initiated (Date): August 1973 **Status:** Continuing **Completed (Date):** June 1976 **Last Updating (Date):** December 1975	

General Aim and Particular Objectives

Objective of this task is the investigation of the behaviour of fuel rod canning tubes with respect to loss-of-coolant-accident (LOCA) conditions. The investigations concern the inside burst test of fuel rod specimen at elevated transient temperatures to determine the diameter increase, the burst-pressure, and the burst-temperatur.

Experimental Facilities and Program

The program is divided into three parts:

a) The influence of different parameters on diameter increase and burst-rupture will be investigated by a parameter study.
The following parameters will be regarded:

- Heating rate
- maximum temperatur T_{max}
- Internal pressure
- material condition (oxidated, H_2 content)
- test-atmosphere (air, inertgas, steam)

Tests will be run with direct resistivity-heating of fuel rod specimen filled with alumina-pellets and a distinct helium prepressure. Temperature-time-correlation will be simplified.

b) The influence of the increasing gap between the fuel and the cladding during the heat-up phase on the diameter increase will be studied. Therefore special specimen must be developed with internal heaters and high heat capacity.

c) To find the correlation between the conservative tests above and a realistic excursion of a LOCA, tests will be run with approximated temperature-time-correlations.

Project Status/Progress to Date

The burst rupture tests on different Zry-fuel rod specimen were continued. The test conditions and parameters were:

Zry-fuel rods with PWR-dimensions
length of the specimen: 480 mm
Medium internally/externally: air/air
Direct heating
Filling: Al_2O_3-pellets
Filling rate: 75 %

Material condition:

a) hydrided samples with a hydrogen content of about 300 ppm (gaseous hydrided)

b) artificially aged samples with oxide layers of 10 - 20 µm and hydrogen contents of about 300 ppm (corrosion in a LiOH-solution at 360 °C and about 200 bar)

Burst rupture tests were performed in order to determine the oxygen content in the residual gas after the test with a quadrupole mass spectrometer. The test conditions and parameters were similar to the burst-tests.

Burst rupture tests with <u>indirectly (internal) heated specimen</u> of Zry-4 fuel rods were conducted. The internal pressure was constant and varied from 10 - 110 bar, the heating velocity was about 100 °C/s.

Transient tests on internally gas filled tubes were performed in order to investigate the temperature effect on the wall when ballooning begins.

The test for the <u>simulation of the GaU</u> with Biblis A data were continued.

<u>Project Status/Essential Results</u>

The Zry-fuel rods with different material conditions showed following behaviour compared with as received samples:

a) gaseous hydrided samples had nearly the same circumferencial elongation

b) artificially aged samples had lower circumferencial elongations.

The nitrogen/oxygen analysis of the internal pressure tests showed the following results:

Temperature (^{o}C)		750	800	860	890	1000
Internal pressure (bar)		80	50	20	20	10
Preussure time (sec.)		60	80	280	60	30
Circumferencial elongation (%)		70	12	34	14	25
Conditions on the end of the tests	P bar	54,0	45,6	13,5	17,8	8,8
	P_{N_2} bar	49,1	43,7	11,8	16,5	8,6
	P_{O_2} bar	4,9	1,9	1,7	1,3	0,2
	P_{N_2}/P_{O_2}	10	23	7	13	37

The burst rupture tests with internal heating gave lower circumferencial elongations compared with direct heated samples.

The GaU-burst tests showed that it was necessary to adapt the internal pressure, because otherwise in the begin of the test the full differential pressure will deform the samples during the first peak, resulting in an unrealistic behaviour during the second peak of the temperature.

Next Steps

Internal pressure tests will be conducted in a steam atmosphere with Helium and UO_2-filling.

Relation with Other Projects

RS 177: Preliminary empirical description of the fuel rod behaviour during LOCA

Reference Documents/Degree of Availability

No reports available.

Classification: 1.3	
Title 1 (Original Language): Vorläufige empirische Beschreibung des Verhaltens von Brennstäben bei hypothetischen Kühlmittelverluststörfällen (RS 177 - I.1.3; Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Preliminary Empirical Description of the Fuel Rod Behaviour during LOCA	Project Leader: Dr. Wunderlich
Initiated (Date): ^ 9. 1975 Completed (Date): 31. 8. 1976 Status: Continuing Last Updating (Date): 31. 12. 1975	

General Aim

Deformations of fuel rod tubes during LOCA are to be described empirically. Therefore the experimental data of the ballooning and burst tests gained from RS 107 will be used for the calibration of a material law, which describes analytically the process of fuel rod ballooning during LOCA.

Particular Objectives

For ballooning the creep law derived by Norton has to be improved, because constant Norton parameters are not applicable in the whole stress and temperature range during LOCA.

Experimental Facilities

No experimental facilities are necessary.

Research Program

- development of the empirical material law
- adjustment of the parameters to the data of the RS 107 burst tests
- modification of the parameters due to influences of indirect heating, oxidation and hydriding.

Project Status/Progress to Date

From the experimental data of the directly heated ballooning and burst tests RS 107 the temperature dependence of the burst stress of KWU fuel rod tubes was determined and compared with data from the literature.

Using the investigated burst stress data, the ballooning of two specimen from RS 107 was calculated and compared with the experimental data.

Project Status/Essential Results

The literature study on the temperature dependence of burst stress data showed a rather broad spectrum, resulting from different test conditions and evaluations.

The strain velocity during ballooning can be described by a modified Norton law, using a plastic correction factor, which containes a limiting stress value. The limiting stress value is assumed to be between the yield and the burst stress and to correspond during transient processes to the burst stress.

By suitable selection of the burst stress value good agreement was obtained between experimental and calculated data for the cases of constant stress and strain- dependent stress. The burst stresses, as used in the calculations, were within the scatter range of the experimental data. Another result was, that good agreement between calculated and measured values was obtained by suitable adjustment of the temperature dependence of the limiting stress value. Up to 30 strain the influence of the limiting stress on the calculated strain value is greater than the influences of the cladding stress assumptions, because the calculated strain is the same for strain dependent or constant cladding stress.

Next Steps

Adjustment of the parameters of the flow-creep law to the data of the RS 107 test.

Relation with Other Projects

RS 107: Behaviour of Zry-4 Canning Tubes under Loss-of-Coolant-Accident Conditions

Reference Documents/Degree of Availability

No reports available.

Classification: 1.3	
Title 1 (Original Language): Parameteruntersuchungen über die Beeinflussung der Hüllrohre durch Nachbarstäbe beim Kühlmittelverluststörfall (RS 185 - I.1.3, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Investigations on the Influence of Neighbouring Fuel Rods During LOCA	Project Leader: Romeiser
Initiated (Date): 1. 10. 75 — Completed (Date): 31. 7. 77	
Status: Continuing — Last Updating (Date): 31. 12. 1975	

General Aim

The influence of neighbouring fuel rods on the ballooning and burst behaviour of a single fuel rod during LOCA and the mechanical and thermal forces are to be investigated.

Particular Objectives

The experimental tests shall give information, whether the results, obtained for a single fuel rod (parameter: differential pressure, temperature, holding time) are changed by neighbouring fuel rods. Expecially it will be investigated, what happens when the neighbouring rods are touched: change of diameter increase, time until burst and influence of burst on neighbouring rods.

Experimental Facilities

The test apparatus consists of the following equipments: five-zone-oven with control equipment for the surrounding temperature of the samples.
Inside heating (heating transformer with control equipment) to adjust the higher temperature of the samples I or II.
Pressure apparatus to adjust the inside pressure.
Measuring and recording equipment to control and record the data.
Sample arrangement consisting of 2 active and 6 passive samples.

Research Program

Two fuel rod specimen are surrounded by six dummies, made of compact rod material, arranged in a 3 x 4 - 4 configuration. The influence of the ballooning and bursting on the neighbouring rods will be investigated with cooling (gas) and without cooling for the following cases:

a) tests with equal temperatures:
   The arrangement shall guarantee, that besides ballooning the lateral displacement of the rods can be investigated. Of special interest are the ballooning of one specimen in a non-disturbed surrounding, the ballooning of a specimen towards another deformed specimen in a non-disturbed surrounding and tests with two ballooning rods.

b) tests with different thermal load:
   The specimen are heated internally, resulting in higher temperatures compared with the surrounding dummies. Planned are tests with one internally heated specimen, two specimen with equal temperatures, and two specimen with different temperatures

The tests will be run under argon-atmosphere, the fuel rods will contain helium. This concept will be improved when the first results are available.

Project Status/Progress to Date/Essential Results

The test arrangement for isothermal investigations was designed and constructed. Some components of the equipment are built and assembled. The material for the specimen and dummies have been procured. The tests have not yet started, therefore no results are available.

Next Steps

The test apparatus will be completed, the specimen and other components will be installed. Some preliminary isothermal tests with and without argon cooling will be performed.

Relation with Other Projects

RS 107 Behaviour of Zry-4 Canning Tubes under Loss-of-Coolant-Accident Conditions

Reference Documents/Degree of Availability

No reports available.

<table>
<tr><td colspan="3">Classification: 1.3</td></tr>
<tr><td colspan="2">Title 1 (Original Language): Notkühlung von LWR: Theoretische und experimentelle Untersuchungen zum Brennstabverhalten beim Kühlmittelverlustunfall und ATWS und zur Auswirkung von Brennstabschäden auf die Wirksamkeit der Kernnotkühlung
(PNS 4230 - I.1.3., Jahresbericht A 75)</td><td>COUNTRY:
BRD

SPONSOR:
BMFT

ORGANIZATION:
GfK, Karlsruhe</td></tr>
<tr><td colspan="2">Title 2 (english):
Theoretical and Experimental Investigations of LWR-Fuel Rod Behaviour during LOCA and ATWS</td><td rowspan="2">Project Leader:
A. Fiege
(coordination)</td></tr>
<tr><td>Initiated (Date):
1973
Status:
continuing</td><td>Completed (Date):
1979
Last Updating (Date):
December 1975</td></tr>
</table>

General Aim

The aim of this project is the development of verified analytical models for the response of LWR fuel rods to LOCA and ATWS conditions and the reliable description of failure mechanisms and their feedback to the effectivness of ECCS in a fuel behavior code system SSYST (PNS 4231).

The detailed quantitative understanding incorporated in the fuel behavior code must be verified by representative experiments.

The basic philosophy of the experimental Program of the Projekt Nukleare Sicherheit (PNS) is to investigate the failure mechanisms of zircaloy-clad fuel rods as a function of the main parameters (as differential pressure, temperature and material properties) systematically with a broad spectrum of out-of-pile experiments and to confirm these out-of-pile experiments with special in-pile investigations.

The main tasks of this program are:
- Investigations of the material properties of zircaloy at high temperatures (PNS 4235.1, 4235.2, 4235.3)
- Out-of-pile loop experiments under reactor typical coolant conditions in different accident phases (PNS 4236, 4238, 4239)
- In-pile experiments in the steam contamination (DK)-loop of the FR2 reactor (PNS 4237.1, 4237.2)

In addition, a smaller experiment is performed in the FR2 reactor to provide better data of the 235 U decay heat in the first 10 to 1000 sec of a LOCA (PNS 4234)

These different projects mentioned before are described in seperate reports.

Classification: 1.3	
Title 1 (Original Language): Theoretische Untersuchungen zum Brennstabverhalten beim Kühlmittelverlustunfall und ATWS (PNS 4231 - I.1.3., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe Univ. Stuttg. (IKE)
Title 2 (english): Theoretical Investigation of LWR Fuel Rod Behavior during LOCA and ATWS	**Project Leader:** R. Meyder H. Unger
Initiated (Date): 1973 / **Completed (Date):** 1979	
Status: continuing / **Last Updating (Date):** December 1975	

1. General aim

The aim of this project is the development of a verified analytical model to describe the behavior of zircaloy clad fuel rods in different reactor incidents as LOCA and ATWS. Especially the effect of ballooning and its consequences are studied and described quantitatively. The theoretical studies are performed in close cooperation between GfK/IRE and the "Institut für Kernenergetik" (IKE), Stuttgart.

2. Particular objectives in 1975

Further development of the modular code system SSYST which allows to simulate the interaction between heat conduction in a fuel rod, heat transfer in the gap, swelling and ballooning of fuel and clad, pressure in coolant and fuel rod as well as the thermo- and fluid dynamic conditions in the coolant channel and the primary coolant system of a LWR. Improvement of some moduls which already exist and the integration and improvement of a reflood code.

Documentation of the first version of the code system, SSYST-Mod-1.

Test of the interaction of SSYST moduls by means of various special problems and investigations, including complete primary coolant system analysis, employing all of the moduls. Performance of parametric studies on the influence of expansion, resp. ballooning behavior of zircaloy-fuel rods, burnup, of the gap conductivity, on the time behavior of heat transfer and temperature as well as geometrical data. Investigation of parameters which are important for the reflood phase of a LOCA, such as heat inventory in the fuel rod, coolant temperature, vapor production etc.

3. Research program

The development of the fuel rod behavior model will be performed in three steps:

1) Development of a single-rod model describing the fuel rod behavior during LOCA and ATWS, especially time, position and extent of the ballooning.

2) Development of a bundle model describing the behavior of fuel rod clusters during LOCA including rod-to-rod interaction and failure propagation.

3) Calculation of the effects of coolant channel blockages upon the effectiveness of emergency core cooling.

4. Project status

4.1 Progress to date

The progress achieved in 1975 can be summarized as follows:

- The modul STADEF simulating the two-dimensional clad deformation has been completed.

- The modul ZIRKOX, taking into account the clad oxidation has been established.

- The program WAK of KWU, performing thermo- and fluiddynamic calculations of the coolant, e.g. the water-level during the low-pressure-LOCA-phase, for a coolant channel has been made available, improved and integrated as a modul.

- A procedure describing the pressure-dependent solid-solid heat transfer between clad and fuel has been integrated into the modul WUEZ.

- The first version (SSYST-Mod-1) of the program system has been completed and documented /4/. It is available for the CDC 6600 and the IBM 370/168.

- The IBM-version of the SSYST-kernel has been improved which resulted in a significant reduction of computer costs.

- Work has been done in order to describe the statistics of the ballooning.

- The work on improved description of zircaloy properties was continued with the verification of tensile and burst-tests of PNS 4235.1 resp. PNS 4238 and in determining optimized constants in equations for diffusion coefficients in ZrO_2, α and β zircon according to the results of oxidation experiments of PNS 4235.2.

4.2 Essential results

All moduls for the first version of the program system, SSYST-Mod-1 are completed, tests of the system have been performed successfully. The system as well as its components have been improved and applied to various problems. In particular, the following results have been achieved:

Within the framework of the German participation in the calculation of "standard-problems" of the USNRC (CASP-program), standard problem 2 has been carried out using RELAP3, resp. RELAP4 (emergency core cooling experiment 1011 at the "1 - 1/2 loop semiscale test facility of the ANC, Idaho). Using the modul STADEF and Norton's creep law, different ballooning experiments have been calculated. The calculations showed that the results depend strongly upon the constants used in the creep law and confirm the importance of further material investigations, such as performed in PNS 4235.1.

Data for a 1200 MW_{el}-PWR assuming a cold leg double-ended break have been worked out and used for SSYST calculations in order to analyse the LOCA behavior. The results are reported in /2/. Various sensitivity studies employing especially different moduls lead to an improved understanding of the applicability of the code system as well as the interaction and importance of different parameters during a LOCA, such as gap conductance, mechanical and metallurgical properties of the clad, fission product inventory (burnup) and decay heat, heat transfer problems during fast transients and the events during the reflood resp. rewetting phase. The influence of a temperature peak of 5 to 10 K on the ballooning e.g. corresponds to the influence caused by changes in clad thickness within the limits of the manufacturing tolerances. More results are given in /2/.

5. Next steps

The program system SSYST will be improved continuously with respect to the development of moduls as well as to the data-handling and numerical methods used in the code. In general, the analytical work and the code development will be extended as follows:

- Investigation and improved simulation of the reflood phase of a LOCA.
- Extension of the single rod models in order to investigate failure propagation problems on the basis of simplified simulation of neighbouring subchannels.
- To improve information on whole core behavior during LOCA sequential single rod analysis will be done for a number of rods. These rods represent classes which differ in burnup and power.
- Supporting calculations for the experimental program (PNS 4236-39).
- Verification of experiments.
- Calculation of standard problems.

6. Relation with other projects

This project (PNS 4231) is part of the major project PNS 4230 and strongly connected to the experimental activities PNS 4235 - 4239.

7. Reference documents

/1/ 1st PNS-Semi-Annual Report 1975, KFK 2195 (German with English abstracts)

/2/ 2nd PNS-Semi-Annual Report 1975, KFK 2262 (German wigh English abstracts)

/3/ Gulden, W. et al.
Dokumentation von SSYST-Mod-1. Ein Programmsystem zur Berechnung des Brennstabverhaltens bei einem Kühlmittelverlustunfall.
PNS-Arbeitsbericht Nr.58/75, IKE-Bericht Nr.4-51 (German)

/4/ R.Meyder, W.Gulden
Die analytische Beschreibung des Brennstabverhaltens bei Störfällen. KFK-Nachrichten 3/1975 (German)

8. Degree of availability

- KFK-reports: unrestricted distribution
- PNS-Arbeitsberichte: restricted distribution.

835

Classification: 1.3	
Title 1 (Original Language): Untersuchungen zum mechanischen Verhalten von Zircaloy-Hüllrohrmaterial beim Kühlmittelverlustunfall (PNS 4235.1 - I.1.3., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Investigation on the Mechanical Behavior of Zircaloy-Cladding Material at High Temperatures	**Project Leader:** M. Boček
Initiated (Date): April 1972 **Completed (Date):** 1978/79 **Status:** continuing **Last Updating (Date):** December 1975	

1. General Aim

Investigation of the plastic behaviour of Zircaloy-4 during different reactor incidents, especially LOCA-typical temperatures and stress transients and in a LOCA-typical environment..

2. Particular Objectives

The main result desired is the determination of a mechanical equation of state, containing all the parameters which influence the plastic strain.

3.1 Experimental facilities

To determine the particular dependencies extensive series of examinations are needed. For this purpose uniaxial strain tests seem more appropriate than multiaxial deformation tests on tubes. A recalculation of the multiaxial stress conditions existing in the cladding to the uniaxial case corresponding to a simple strain test is necessary due to the strong anisotropy of Zircaloy. Thus a detailed knowledge about the influence of texture upon the plastic behaviour is indispensable.

It is expected that a lot of uncertainities will be inherent to the equation of state derived in this way. Therefore parallel burst tests on Zircaloy-tubes should be performed. These attendend investigations may allow for the determination of the circumferential strain rate.

3.2 Research Program

- Investigation on texture at high temperatures
- Investigation on the temperature- and strain-rate dependence of strain, stress and strain rate exponent.
- Investigations on the influence of oxidation products on plastic behavior
- Investigations on phase transformation
- Postexamination on irradiated material
- Investigations on the behavior of the cladding under internal pressure
- Determination of a constitutive equation for plastic deformation

4. Project status

4.1 Progress to date

Investigations were performed on:

- The temperature dependence of the total elongation, the yield stress and the strain rate exponent for different sample orientations in air atmosphere.
- The strain rate and temperature dependence of the strain rate exponent in the temperature range of 400 - 1100°C in air atmosphere.
- The deformation dependence of the strain rate exponent in air atmosphere.
- The influence of heat treatments on the deformation behaviour of the material.
- The Metallography of the phase transformation $\alpha \rightarrow (\alpha+\beta)$.
- Metallographic observations of the grain structure after deformation.
- The macroscopic fracture behaviour
- Isothermal burst tests at temperatures of 800°C and 900°C in vacuum.

4.2 Essential Results

- The total elongation shows a maximum at 850°C and decreases above 1050°C to very low values and is dependent on texture. The yield stress decreases with temperature and is dependent up to 650°C on texture. The strain rate exponent has shown to be very sensitive to sample orientation (angle γ between the sample axis and the rolling direction). Thus the superplastic behaviour is not only determined by the temperature- and stress- (strain rate-) range, respectively but depends also on the texture. Samples with $\gamma = 45^{\circ}$ show normalplastic behaviour.

- For normalplastic deformation the strain rate exponent is essentially independent upon strain rate and temperature. The apparent activation energy and the constant of Norton's creep equation in this range were determined for temperatures between 400°C and 840°C.
- When the material behaves superplastically a strong dependence of the strain rate exponent $\bar{m}$ upon strain is observed if the deformation is carried out in air atmosphere.
- Heat treated samples (1000°C/1H/vacuum) revealed a considerable grain growth and subpressing superplasticity.
- The phase change $\alpha \rightarrow (\alpha+\beta)$ was investigated metallographically. Occasionally small β-phase precipitates are observed inside the α-grains, however intensive precipitation occours in the grain boundaries.
- No stretched grains were observed on samples for which $m \geq 0,3$.
- Samples deformed to fracture below 850°C failed by necking. No necking occured on specimens for which the strain rate exponent $m \geq 0,3$.
- Isothermal burst tests on Zircaloy-4-cladding were initiated at temperatures of 800°C and 900°C. Even at times of exposure of 30 sec, i.e. high deformation rates at 900°C and an internal pressure of 35 [kp/cm^2], circumferential strains of 80 % were obtained.

5. Next steps

The data obtained from tensile tests in air atmosphere and from creep tests in vacuum will be used for compiling a deformation code.

Burst tests will continue, the apparatus will be improved to allow for continuous measurement of circumferential strain.

6. Relations to other projects

PNS 4235.2
PNS 4235.3
RS 107

7. Reference documents

1.st PNS-Sem.-Annual Report 1975, KFK 2195 (German with English abstracts)
2 nd PNS-Sem.Annual Report 1975, KFK 2262 (German with English abstracts)

M. Boček, P. Hofmann, S. Leistikow, C. Petersen.
Zum Materialverhalten von Zircaloy-Hüllrohren beim Kühlmittelverlustunfall.
KFK-Nachrichten (34), 3 75 (German).

M. Boček, C. Petersen
Das plastische Verhalten von Zircaloy-4 oberhalb 400°C
Reaktortagung Nürnberg 8. - 11. April 1975 (German).

8. Degree of availability

Unrestricted distribution.

Classification: 1.3	
Title 1 (Original Language): Untersuchungen zur Hochtemperatur-Wasserdampf-Oxidation an Zircaloy-Hüllmaterial (PNS 4235.2 - I.1.3., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Investigation on the High Temperature Steam Oxidation of Zircaloy Cladding Tubes	**Project Leader:** S. Leistikow
Initiated (Date): 1973 **Completed (Date):** 1978/79	
Status: continuing **Last Updating (Date):** December 1975	

1.+2. General aim and particular objectives

Studies of Zircaloy/Steam High Temperature Oxidation Kinetics.

Behavior of Zircaloy 4 tubing under different temperature/pressure-transient conditions in steam.

3.1 Experimental facilities

An induction heating system for program-controlled Loca-similar time-at-temperature steam exposure of tube sections and an experimental set-up for stress-rupture testing of internally pressurized tube capsules under outer steam oxidation were installed and successfully operated.

3.2 Research program

To start the induction heating system as tool for routine experiments, the former experimental oxidations kinetic results at 1 - 15 min, 900 - 1300°C had to be reproduced and confirmed. Then Loca-similar runs were performed under program controlled time-at-temperature conditions to obtain results of the oxygen take-up and its sensitivity to changing Loca conditions. Furthermore stress-rupture experiments to check in the influence of stress upon oxidation rate and of oxidation upon the mechanical properties were started.

4. Project status

4.1 Progress to date

The induction heated high-temperature oxidation experiments confirmed quantitatively the already known parabolic time/temperature relationships of total oxygen consumption and penetration. Loca-similar experiments showed - compared to isothermal runs at maximum holding time - a reduction of oxygen consumption of about 25 - 30 %.

4.2 Essential results

Preliminary experiments to test the mechanical (creep) properties of unoxidized tube capsules in argon provided the necessary hoop stress/time-to-rupture relationships and elongations at rupture at 1000^{o} and $1100^{o}C$ for internal pressures between 2 and 10 at, 20 min and 20 sec. These data will be compared with those being presently measured under oxidizing steam conditions. After rupture circumferential elongations of the capsules of 75 - 101 % were measured.

After other runs the morphological evaluation of combined burst-creep experiments under steam oxidation showed that fast creep tends to open-up cracks in the ZrO_2-scale, making the flanks to parts of the quickly oxidizing surface, combined with local wall-thinning, while slow creep favors only local increase of oxide scale thickness.

A comparison between different oxidizing gases (steam, oxygen, air) showed that air is the most effective oxidizing medium (especially at temperatures above $1100^{o}C$), while steam is the most moderate one. In consecutive runs a certain nitrogen content of the tube specimens stimulated, while hydrogen moderated oxygen consumption at $1000^{o}C$.

5. Next steps

Exposure of tube sections in metallic and preoxidized state to different Loca-similar conditions.

Exposure of tube capsules to steam under superimposed creep and comparison with mechanical data evaluated by creep testing in argon.

Metallographic examination of reaction products to measure the extent of oxidation under superimposed creep.

6. Relation with other projects

These experiments belong to the Material Research on LWR-Fuel Behavior under Loca conditions as part of the Project Nuclear Safety in the Nuclear Research Center of Karlsruhe.

7. Reference documents

/1/ M. Boček, P. Hofmann, S. Leistikow, C. Petersen
"Zum Materialverhalten von Zircaloy-Hüllrohren beim Kühlmittelverlustunfall"
KFK-Nachrichten 7, Heft 3 1975 (34-44) (in German).

/2/ 1st PNS-Semi-Annual-Report 1975, KFK 2195 (German with English abstracts)

/3/ 2nd PNS-Semi-Annual-Report 1975, KFK 2262 (German with English abstracts)

8. Degree of availability

Unrestricted distribution

Classification: 1.3	
Title 1 (Original Language): Untersuchungen zum Brennstabverhalten in der Blowdown-phase eines Kühlmittelverlustunfalls (PNS 4236 - I.1.3., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Investigations of the Fuel Rod Behavior during the Blowdown-Phase of a Loss-of-Coolant-Accident	Project Leader: G. Class K. Hain
Initiated (Date): 1972 — Completed (Date): 1977/78 Status: continuing — Last Updating (Date): December 1975	

1. General Aim

The aim of experiments is to provide information about the failure limits of fuel rods under incident conditions. The improved knowledge of the fuel element behavior in the blowdown phase of a loss-of-coolant accident as a result of the experiments is to be used in setting up a theoretical model.

2. Particular Objectives

Initially, the experiments will be carried out under blowdown conditions typical of PWR's, later on under those typical of BWR's. The fuel rod behavior will be determined by measurement under the transient load including the heat removal from and the internal pressure of the fuel rod.

3. Experimental Facilities and Research Program

A loop facility is being built for the experiments in which the initial steady state conditions can be set with respect to rod power, coolant condition and coolant flow. Blowdown transients can be initiated from this initial (quasi) steady state phase. The experimental program so far includes the simulation of hot and cold leg breaks with break sizes of 1F and 2F. In each case experiments will be carried out at different rod powers and internal pressures.

4. Project Status and Essential Results

Construction of the test facility has progressed far enough to allow a first water pressure test of the high pressure system to be carried out. The first types of fuel rod simulator (750 W/cm max., design incorporating annular Al_2O_3 pellets) and test rod to be used in measuring heat transfer coefficients either have been completed or are about to be completed in the very near future.
With the exception of a few units, the measurement and control components are being erected in situ. This includes special measuring techniques such as the true-mass-flow-meter and the optical measurement of the cladding temperature.

5. Next Steps

All efforts are now being concentrated, on the one hand, on the completion and startup of the test facility together with the special measuring techniques and, on the other hand, on the completion of the evaluation computer programs.

The startup of preliminary experiments is expected for the end of the 1st quarter of 1976.

While work is going on to complete the experimental facility, possibilities are being studied to use annular ThO_2 pellets to improve the quality of the simulation of the fuel rod simulator.

6. Relations with Other Projects

PNS 4231, 4237, 4238, 4239

7. Reference Documents

1st PNS Semi-Annual-Report 1975, KFK 2195 (German with English abstracts)

2nd PNS Semi-Annual Report 1975, KFK 2262 (German with English abstracts)

8. Degree of Availability

Unrestricted distribution

Classification: 1.3	
Title 1 (Original Language): Untersuchungen zum Brennstabversagen in der 2. Aufheizphase eines Kühlmittelverlustunfalles. In-pile-Versuche mit Einzelstäben im Dampf-Kontaminations-Loop (DK-Loop) des FR 2. (PNS 4237.1 - I.1.3., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Investigations of Fuel Rod Failure in the 2nd Heatup Phase of a LOCA. In-pile-Experiments with Single Rods in the DK-Loop of the FR-2-Reactor	Project Leader: E. Karb, L. Sepold
Initiated (Date): April 1972 Status: continuing	Completed (Date): 1979 Last Updating (Date): December 1979

1. General Aim

The aim of this project is to investigate the failure mechanisms of Zircaloy clad fuel rods in the 2nd heatup phase of a loss-of-coolant accident.

2. Particular Objectives

The aim of the in-pile experiments is to investigate the influence exerting the mechanisms of fuel rod failure by nuclear parameters (thermal and mechanical fuel behaviour, influence of fission products, of irradiation and fission gas release from the fuel by temperature transients, generation of true decay heat) which cannot be simulated in out-of-pile tests.

3.1 Experimental Facilities

The DK-loop (Dampf-Kontamination = steam contamination) was designed for the in-pile operation of fuel rod specimens cooled by superheated steam.

It is capable of transporting a coolant mass flow of 140 kg/h through the test section at a pressure of 160 bar and a temperature of 550° C (superheated steam).

A detailed description of the loop is given in SAR, Ref. /1/. The loop has been in operation for several years.

3.2 Research Program

According to present planning 42 in-pile tests will be performed with non-irradiated specimens and later on with fuel rods pre-irradiated between 2500 and 3500 MWd/t of burnup.

4. Project Status

All test fuel rods are available. The loop is ready for operation. The hardware data transfer and test rod instrumentation are assembled. The first two in-pile tests were already performed. Pre-irradiation has begun.

4.1 Progress to Date

The extensive preparations and the implementation of the first main tests with nuclear rods were the dominating activities in 1975.

The specification for the nuclear specimens was terminated at the beginning of 1975. The last work required in this context was the design of the plenum. Then, the specimens were ordered in spring. They were delivered in early August.

The test loop has been changed above all to improve the accuracy of the measurements. Just in time before startup of the tests the nnection of the loop and specimen instrumentation with the CALAS aata acquisition system was completed.

A thermocouple specification was set up to measure the cladding temperature by means of thermocouples and a small series (for preliminary tests) was ordered, inspected and accepted which complied with this specification.

The two tests were successfully performed in October 1975. Pre-irradiation rigs were installed in FR-2. With the introduction of these rigs the pre-irradiation began with the fuel specimens to attain the maximum target burnup.

Within the framework of activities relating to the safety report on in-pile experiments with pre-irradiated pins the radiological environment burden caused by the experiments is presently calculated for various boundary conditions.

4.2 Essential Results

The design of the plenum volume required calculation of the pressure as a function of time, which develops at the point of ballooning in the test rod as compared to the full length PWR rod.

Satisfactory agreement was reached for the most important period of time of failure sequence, i.e. the time approximately 1 second after the onset of ballooning. The result is that the plenum of the test rod has the same volume as both PWR rod plena together. The calculated results were supported by gas flow experiments with helium and argon performed under the anticipated pressure conditions in the rod.

The differences between the test rod and the regular PWR rod are a shorter pellet stack length (test rod to PWR rod length $\approx 1/7$), and the fact that the test rod has only one (upper) plenum whereas the PWR rod has the free gas volume divided into an upper and a lower plenum.

In both tests the fuel specimens were exposed to internal pressures of 50 and 100 bar helium (at steady-state temperature) and subjected to the specific temperature transient in the DK-loop of FR-2. The first rod bursts at 50 bar and about 800^{o} C of cladding temperature. This pair of values lies within the scattering band of out-of-pile results.

5. Next Steps

Measurement and further post examinations of the two first rods are to be performed in the Hot Cells at the beginning of 1976.

Particular attention will be paid in 1976 to the problem of thermocouple attachment to the irradiated fuel rod specimens.

Errors mus still be corrected in data acquisition. Also some minor modifications at the test loop will be required in 1976.

The safety report will have to be established for the experiments with pre-irradiated specimens.

6. Relation with Other Projects

This project is linked to the PNS 4230 group (emergency core cooling of LWRs). Close Connections to 4237.2.

7. Reference Documents

/1/ RB Report No. 4/71 (July 71), safety report for the steam contamination loop (Project FR 2/55a).

/2/ 1st PNS Semi-Annual Report 1975 KFK 2195 (German with English abstracts)

/3/ 2nd PNS Semi-Annual Report 1975 KFK 2262 (German with English abstracts)

8. Degree of Availability

- KFK reports: Unrestricted distribution
- RB reports: Restricted distribution

Classification: 1.3	
Title 1 (Original Language): Referenzversuche zu den in-pile-Experimenten PNS 4237.1 mit elektrisch beheizten Brennstabsimulatoren (Einzelstäbe) (PNS 4237.2 - I.1.3., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Reference Tests for PNS 4237.1. In-pile Experiments with Electrically Heated Fuel Rods (Single Rods)	**Project Leader:** B. Räpple
Initiated (Date): 1.1.1973 **Status:** continuing	**Completed (Date):** 1979 **Last Updating (Date):** December 1975

1. General Aim

Investigations of the mechanisms and the extent of failure of Zircaloy clad fuel rods during the second heatup phase of an LWR loss-of-coolant accident.

2. Particular Objectives

Provision of experimental data by means of electrically heated fuel rod simulators as a basis of comparison with results of experiments obtained with fuel rods exposed to nuclear heating. (PNS 4237.1)

3.1 Experimental Facilities

The experiments are being performed in the in-pile loop of the FR2 Research Reactor.

3.2 Research Program

Some 25 rods are exposed to transient operating conditions under the test program. The pressure differences, the temperature gradients and the power profiles at the specimen are assimilated to the conditions applicable to nuclear tests and the resulting points of failure are examined and evaluated.

4. Project Status

Present activities relate to testing of the heater rod concept, the test rig and the measuring system.

4.1 Progress to Date

Manufacturing was completed of a first series of 5 heater rods to be used in preliminary in-pile tests. One of the rods was installed in a test rig together with Al_2O_3 ring pellets and subjected to about 30 steady-state and 3 transient tests in a first in-pile program.

4.2 Essential Results

The tests had been essentially a success. Disturbances only occurred in the measurement of wall temperatures, which had a negative effect on axial temperature profile recording at the cladding tube. All the other test targets have been reached. They allowed to gather experience relative to

- the gamma dose rate of the irradiated specimen,
- the necessary setting times (electric heating, reactor, loop),
- possible sources of error at the specimen, the instrumentation, the electric and the measurement systems inclusive of data processing (CALAS),
- the application of the methods allowing to determine the thermal and electric power, respectively,
- processing of measured values in steady-state and non-steady-state operation.

5. Next Steps

It is planned to perform additional preliminary in-pile tests during the first half of 1976 aiming at the following targets:

- Continued testing of the heater rod, the electric and measurement systems; error correction.
- Determination and comparison of the power generated in the heater zone (electric, thermal, nuclear).
- Determination of the gamma heat fraction in the aggregate thermal power.

t is planned to install another three test rigs for this purpose. The prerequisite of these preliminary in-pile tests is the installation in the test rig of a new calibrated orifice.

Work on specimen design and testing will be continued and evaluation will start of the preliminary in-pile tests.

6. Relation with Other Projects

The tests will alternate with the nuclear tests to be performed at the same test facility and in the same test position of FR-2. The results obtained in the preliminary in-pile tests will also be used in the nuclear power determination method under the PNS 4237.1 project.

The main tests planned for a later date are intended to yield reference data on the nuclear test parameters of the same project.

7. Reference documents

/1/ 1st. PNS-Semi-Annual Report 1975, KFK 2195, (German with English abstracts)
/2/ 2nd. PNS-Semi-Annual Report 1975, KFK 2262, (German with English abstracts)

8. Degree of availability

Unrestricted distribution

Classification: 1.3	
Title 1 (Original Language): Untersuchung zur Wechselwirkung zwischen aufblähenden Zircaloy-Hüllen und einsetzender Kernnotkühlung (PNS 4238 - I.1.3., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Investigations of the Interaction between Ballooning Zircaloy Claddings and the Reflooding Emergency Core Cooling Water	**Project Leader:** K. Wiehr
Initiated (Date): January 1973 **Status:** continuing	**Completed (Date):** 1979 **Last Updating (Date):** December 1975

1. General aim

The aim of this project is to obtain experimental information for development and verification of the SSYST fuel rod behavior code with regard to the heat-up and reflooding phases of a LOCA.

2. Particular objectives

Important features of the experiments are the interaction between the ballooning mechanism and the emergency core cooling and the recording of the time-dependent ballooning process of the zircaloy cladding.

Particular objectives are the investigations of:

- time-dependent ballooning mechanism of single rods
- interaction between ballooning and cooling
- thermal and mechanical effects of the rod-to-rod interaction on failure behavior in rod bundles
- information on failure propagation
- extent and distribution of flow blockage.

3.1 Experimental facilities

The experiments will be carried out with special fuel rod simulators with 3.90 m heated length, axial power profile and a total length of 5 m. The fuel rod simulators are assembled to a 5x5 array and installed in the test section of the test rig.
The test rig is able to simulate the processes during the refill and reflooding phases of a LOCA. The test rig is a closed loop system with steam and water circuits. The water circuit delivers the cooling water into the test section at different flooding rates and inlet - temperatures. The electrical power control of the fuel rod simulators will be done by an automatic system which adjusts the power according to the decay heat.

X-ray penetration with a camera will be used for observation of the ballooning process. For temperature measurement of the ballooning cladding a radiamatic device with a two-colour pyrometer has been developed. The internal gas pressure in the ballooning zone will be measured by means of a capillary with a micro-pressure gauge.

The 130 experimental data are recorded by a fast recording system at a scanning frequency of 10 kHz and ten times per second. A fast data line with terminals and video displays is used to process the measured data by the CALAS-system.

3.2 Research program

The experiments begin with separate effect tests related to heat transport inside the fuel rod simulator, to inner gas flow in the gap between cladding and pellets and to deformation mechanism of the cladding. With these tests performed on shortened fuel rod simulators and under simplified conditions the corresponding moduli of the SSYST-code will be developed and verified.

The integral tests with a 5x5 rod array are concentrated, for the time being, on single rod tests in a steam atmosphere and under flooding conditions in order to investigate the detailed ballooning

mechanism. After sufficient understanding of the single rod failure mechanism has been generated the tests will be extended to bundle tests for investigations of rod-to-rod interaction and possible failure propagation.
The test parameters will be varied in the following range:

- rod power	24 to 80 W/cm
- axial power profile	stepped profile and cosine-shaped profile
- cladding temperature at beginning of flooding	600 to 900 °C
- pressure difference across cladding	70 to 130 bar
- system pressure	1 to 4.5 bar
- flooding rate (cold)	1 to 9 cm/s
- water inlet-temperature	25 to 65 °C

4. Project status

The technical development of the experiments has been finished. Some separate effect tests have been performed with the fuel rod simulator. The status of the project allows the beginning of the integral experiments on the ballooning mechanism with full-length fuel rod simulators to be scheduled for the first quarter of 1976.

4.1 Progress to date

The development of the fuel rod simulators has been finished. Their thermo- and fluiddynamic behavior is in good agreement with the nuclear fuel rod. Methods of fabrication have been established. A special explosive welding technique has been developed for welding Inconel 600 and Zircaloy-4. A prototype fuel rod simulator was successfully tested at the full design power of 30 kW and a cladding temperature of 1000 °C. The instrumentation of the fuel rod simulators has been determined.

Construction of the test rig and the controllable electrical power supply has been completed.

For recording the ballooning process special measuring devices have been developed. The temperature measurement at the point of maximum cladding deformation will be made by a contactless technique. A special two-colour pyrometer has been developed for this purpose. The lower limit of the measuring range could be lowered to about 500 ^{o} C. Transmission of the local ballooning pressure to a micro-pressure gauge of 15 mm^3 volume is made by a capillary of 0.36 mm diameter. Tests with this measuring system have shown a fast reponse time. The ballooning pressure can be calculated from the signal with adequate accuracy. For recording the time dependent ballooning of the zircaloy cladding a radiographic system with a camera was installed. First pictures taken through 5 mm steel have furnished good results.

One and two dimensional heat conduction computer programs for thermal design and experimental verification of the fuel rod simulators have been finished. Work has been started on the adaptation and verification of some moduli of the SSYST fuel rod behavior code.

First separate effect tests have been conducted on the gas flow in the gap and the deformation mechanism of zircaloy claddings.

4.2 Essential results

The development of the fuel rod simulators has been completed successfully. Calculations with one-dimensional transient heat conduction codes on the thermal behavior have shown good agreement between fuel rod simulator and nuclear fuel rod. It has been established by means of two-dimensional calculations that the influence of the axial heat conduction is small and of importance only very close to stepwise variations of boundary conditions.

Special measuring devices for recording the transient deformation of the cladding have been tested (a two-colour pyrometer for contactless temperature measurement, a micro-pressure gauge with a capillary for the local ballooning pressure, and a radiographic system for the deformation of the zircaloy cladding), the results were very satisfactory.

First experiments on the ballooning mechanism have resulted in data of the temperature and strain rate which are characteristic of normal plastic behavior.

5. Next steps

Important steps to be taken in the next few months include the following aspects:

- commissioning of the test rig
- separate effect tests
- full-length single rod tests in a steam atmosphere and under flooding conditions
- adaptation and improvement of SSYST moduli

6. Relation with other projects

Other theoretical and experimental investigations on fuel rod behavior during a LOCA performed by GfK (PNS 4231, 4235, 4236, 4237, 4239) and KWU (RS 1o7) and on transient events in the reactor core and primary circuit in the low pressure phase of a LOCA, performed by KWU (RS 36).

7. Reference documents

/1/ 1st PNS-Semi-Annual Report 1975,
KFK 2195 (German with English abstracts)

/2/ 2nd PNS-Semi-Annual Report 1975,
KFK 2262 (German with English abstracts)

8. Degree of availability

Unrestricted distribution

2. CORE MELTDOWN

Classification: 2.1	
Title 1 (Original Language): Auswertung von WASH 1400 bzgl. Energiebilanzen im deutschen Kernschmelzprogramm (RS 72 C - I.1.5, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Evaluation of the WASH 1400 Report with Respect to Coremelting (Energy Balances)	Project Leader: Goetzmann
Initiated (Date): September 1975 Completed (Date): March 1976 Status: Continuing Last Updating (Date): December 1975	

General Aim

Study of the chapters of WASH-1400 dealing with coremelting problems (Appendix VIII).

Particular Objectives

Comparison and assessment of the results; the main differences will be discussed. In particular, the accident will be analysed which was the basis of RS 72 a/b.

Project Status / Progress to Date and Essential Results

Study has started with Appendix V, VII and VIII of the Rasmussen-report.

Next Steps

Work will be finished during the first quarter 1976.

Relation with Other Projects

RS 72 a and RS 72 b. Theoretical Determination of Energy Balances for Core Melting

Reference Documents / Degree of Availibility

No reports available.

<table>
<tr><td colspan="3">Classification: 2.1</td></tr>
<tr><td colspan="2">Title 1 (Original Language):
Kernschmelzen: Theoretische Untersuchung der Abschmelzphase
(RS 73 - I.1.5, Jahresbericht A 75)</td><td>COUNTRY:
BRD
SPONSOR:
BMFT
ORGANIZATION:
KWU, Erlangen</td></tr>
<tr><td colspan="2">Title 2 (english):
Core Meltdown Program
Theoretical Investigation of the Different Phases of the Meltdown Process</td><td rowspan="2">Project Leader:
Goetzmann</td></tr>
<tr><td>Initiated (Date):
October 1972
Status:
Continuing</td><td>Completed (Date):
March 1976
Last Updating (Date):
December 1975</td></tr>
</table>

General Aim and Particular Objectives

The aim of this investigation is to develop a computer code which describes the different phases of the meltdown process including the blowdown and heat-up phase following a hypothetical loss-of-coolant accident in a LWR.

Experimental Facilities and Research Program

For a hypothetical loss-of-coolant accident with melting down of the core several characteristic phases are defined:

blowdown-phase: characterized by the dropping of the water level with uncovering of the core and the corresponding temperature rise of the fuel rods.

heat-up-phase: heating up of the rods due to the fission products decay heat and the chemical reaction of steam with the fuel cladding, characterized by ballooning and perforation effects.

meltdown-phase: temperature rise to the melting point of the different components of the core (fuel, cladding, spacers) leading to a slumping of the affected regions of the core; part of the molten material will move further down the core to solidify again, part will fall to the bottom of the core, part will fall out of the core to the lower plenum of the reactor vessel. The mode in which the core is slumping and losing continuously its integrity is depending on many parameters as e. g. the thermal-physical properties of the molten material, the interaction between fuel and cladding and the amount of heat generated. Thus, the model describing the core meltdown has to include all reasonable possibilities of geometric changes of the core and to evaluate the heat and mass transfer connected herewith. Important points in these analyses are the shifting and redistribution of the inherent heat sources, the interaction with the surrounding of the core, the generation of steam potentially increasing the metal-water-reaction, and the conditions at which the supporting grid of the core loses its integrity causing a greater part of the melting core to fall into the lower regions of the vessel.

The above mentioned physical processes are investigated in detail - first to find the important parameters and to establish the governing equations, second to combine the different investigations in a single computer code.

Special attention to the slumping of the core, where in the first phase of investigation a "tree" of the different possibilities of slumping shall be formulated. In a second step a reduction in the variety of the slumping models shall be performed due to adequate experimental and theoretical informations. The parameters with the highest probability of occuring during the melt-down-process shall then be substituted by a more detailed contemplation leading to suitable correlations and evaluation formula respectively.

Project Status / Progress to Date and Essential Results

Documentation and Integration of the individual modules have been continued.

KOCH-2

describes the heat up of the residual water. A one-dimensional heat transfer model has been integrated for the reactor pressure vessel. The analysis is finished and the module was tested (replaces BOIL-2).

CHEBAS-2

physics and input have been documented. The output was adjusted to user's requirements.

SLUMP-2

documented by an intermediate report.

SLUMP-3

completed and integrated into MELSIM.

NACHZRF

GKSS transmitted final version of the NACHZRF module including documentation and input-date covering the test example. Preparation of the modifications required for the integration into MELSIM, completed thus the NACHZRF results can be integrated into the test example.

KUMI

GKSS transmitted a modified version of the KUMI module together with a preliminary documentation. IKE provides the connecting link for MELSIM.

In addition, data were collected for the test sample reactor. These data were used afterwards to perform comparative calculations between KUMI and RELAP.

UMGEB
The UMGEB module was extended, thus providing a detailed model for the radial core environment. The model consists of several concentric cylinders, coupled by radiation.

Intermediate report
An intermediate report was prepared and sent to the IRS.

Next Steps
Completion of work will be delayed for about three months, since rearrangements of the computer center at the Stuttgart University will result in a several weeks' outage of the computational facilities. Evaluation and documentation of the already calculated test example is scheduled for the first quarter 1976.

Relation with Other Projects
see RS 73 A 74

Reference Documents / Degree of Availability

K. Hassmann, Dr. M. Peehs
"Kernschmelzen - Theoretische Untersuchungen der Abschmelzphase".
1. Technischer Bericht Förderungsvorhaben BMFT RS 73
Kraftwerk Union Aktiengesellschaft (Nov. 1975)
Company confidential.

Classification: 2.1		
Title 1 (Original Language): Untersuchungen thermohydraulischer Vorgänge sowie Wärme- und Stoffaustausch in der Coreschmelze (RS 48/1 - I.1.5., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: TU-Hannover
Title 2 (english): Theoretical and Experimental Investigation of the Thermohydraulic Behaviour of a Molten Core at the Bottom of the Reactor Pressure Vessel		Project Leader: Prof. Dr. Mayinger
Initiated (Date): May 1971	Completed (Date): April 1975	
Status: finished	Last Updating (Date): December 1975	

1. General aim

In this research project the thermohydraulic behaviour of molten reactor core at the bottom of the reactor pressure vessel both theoretically and experimentally had been studied. In the program different geometries and boundary conditions were assumed and calculated. The experiments which had been made for verification the computer codes, had affirmed the numerical treatments.

2. Particular objects

Of particular interest was the determination of the heat flux density from the core debris to the wall of the reactor pressure vessel and at the surface of the molten core. In addition the temperature increase in the debris and the convection behaviour were determined. This studies were not limited to the specific geometry of the reactor pressure vessel, but were also extended to horizontal fluid layers. This had to be done by experiments and numerical investigations.

3.1 Experimental facilities

The experiments were made up with the help of the holographic interferometry using water as modelling fluid. This optical method allows to make visible the whole temperature field in the fluid. This temperature field can be recorded and from it the local heat transfer coefficient can be determined. For these investigations two-dimensional models were used and the examined fluid had to be transparent. The volumetric heat sources were simulated by Joules' heat. More detailed information about this measuring technique can be found in /1/.

3.2 Research program

In the theoretical part of the project the computer code THEKAR was established, which describes the thermohydraulic behaviour and the heat transfer in the molten core in different geometries. This includes a horizontal layer, the rotationally symmetrical case of a cylinder and a semisphere, which corresponds to the real conditions within a LWR-reactor.

The experimental studies had the aim to verify the calculated situations in the rectangular and semicircular geometry, essentially the measurement of the heat transfer coefficient. In addition experimental investigations were performed using models in the shape of a segment of a circle, the height of the models was smaller than the radius of curvation. Using these models the situation should be simulated when the bottom of the reactor vessel is only partially filled up.

4. Projekt status

4.1 Progress to date

The research project run out at April 30th, 1975. The theoretical part had been finished some month before and the experimental work in the last period had been done essentially with the interpretation of the results, which we obtained before. In addition to our previous results we were able to establish some empirical correlations of heat transfer in the geometry of a segment of a circle.

With this experimental results, we could introduce a model for the turbulence into the program-code THEKAR. With this model we were able to compute free convection at very high Rayleigh-numbers, which are expected for the molten core in the case of accident.

The results of this research program are explained in the final report in all details (1).

In addition to the equations we gave in earlier reports, we set up correlations for the heat transfer in a semispherical geometry.

For the upper plane wall we obtained the equation

$$Nu = 0.40\ Ra'^{0.2} \qquad (1)$$

and for the semispherical bottom the equation

$$Nu = 0.55\ Ra'^{0.2}$$

These equations are valid for the averaged Nusselt-numbers. Particularly at the bottom local values of the Nusselt-numbers differ from the averaged values with increasing Rayleigh-numbers. The maximum value lies closely to the upper wall in this case.

5. Next steps

The continuation of the work is now done in the project RS 166.

6. Relation with other projects

RS 73: Theoretical Core Meltdown Studies
KWU Erlangen, IKE-Stuttgart 1.10.72 - 31.12.75

7. Reference documents

Quarterly reports in the series: IRS Forschungsberichte (German)
Jan. 1975 - March 1975 IRS-F-25
Apr. 1975 - Jan. 1976 IRS-F-26

/1/ Untersuchung der thermohydraulischen Vorgänge sowie Wärmeaustausch in der Kernschmelze
BMFT-Forschungsvorhaben RS 48/1, Abschlußbericht 1975 (German)

8. Degree of availability

The reports are available at the IRS.

Classification: 2.1		
Title 1 (Original Language): Weiterführende Untersuchung des thermohydraulischen Verhaltens der Kernschmelze (RS 166 - I.1.5., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: TU, Hannover
Title 2 (english): Theoretical and Experimental Investigation of the Thermohydraulic Behaviour of a Molten Core in the Reactor Vessel and on Concrete of the Basement		Project Leader: Prof. Dr. Mayinger
Initiated (Date): May 1975	Completed (Date): Dec. 1978	
Status: continuing	Last Updating (Date): December 1975	

1. General aim

The aim of this research project is to establish computer codes, which describe the thermal interactions of the molten core with various components of the reactor. In order to test the reliability of the codes some experiments with modelling fluids have to be done.

2. Particular objectives

Of particular interest is the heat transfer from the molten core material to the wall of the reactor pressure vessel, while the molten material is flowing into the bottom of the vessel. In addition we want to investigate, whether freezing or liquidation of the material at the wall occurs and how it effects the thermohydraulic behaviour and the heat flux distribution at the bottom of the reactor pressure vessel.

The next important step is the investigation of the spread out of the molten core material on the concrete of the reactor containment and how this material penetrates into the concrete.

In this case the heat transfer from the melt to the concrete has to be found out by considering the influence of the molten concrete on the thermohydraulic behaviour of the molten material.

For the future an investigation of bulk boiling in the molten core is planned. In this case an estimation of the growth and rise of the bubbles is necessary.

3.1 Experimental facilities

For the reliability of the analytical investigations experiments are provided. Velocity fields have been measured with the Laser-Doppler-Anemometer, when the additional fluid is flowing into a layer, in which free convection occurs. A number of these experiments will be carried out with flat, two-dimensional models. In comparison with the numerical calculations the experiments should help to establish a turbulence model in the computer codes.

For the analytical description of the melting process from the beginning of the melt-through of the RPV to the penetration of the concrete by the core melt experiments are planned with material of low melting points.

3.2 Research program

The first step of the program is to establish a computer code, which describes the thermohydraulic behaviour of the molten core during the process of melting down, that means, when the molten material is flowing down continuously on to the bottom of the pressure vessel. It is planned to consider analytically also the time and local dependence of physical properties and the heat source density. Additionally investigations of the behaviour of solid particles in the core melt and the freezing and liquefaction of material at the walls are to be made.

The second point of the research program is the investigation of the formation of caverns within the concrete which is initiated by the heat generation and convection of the molten core material. We are working at a computer code, which describes this process under consideration of chemical reduction of the destroyed concrete in the melt.

Further analyses will describe the bulk boiling in the molten core material. In this part the growth and rising velocity of the bubbles shall be estimated. Additionally some calculations about the outcast of material from the molten core due to the boiling process are to be performed. Finally studies about safety retention systems and the cooling of the molten debris is planned.

4. Progress status

4.1 Progress to date

The project was started by expanding the existent codes for describing the filling up of the vessel with molten material coming from the slumping core. This code is finished in its first version and describes the time dependent thermohydraulic behaviour of the core melt at the bottom of the RPV beginning with the melt process of the core until the filling up of the vessel to its estimated maximum with molten core debris. The analyses show good results, which are discussed in part 4.2. For the corresponding experiments the necessary models and the experimental set-up are constructed.

To investigate the formation of caverns in the concrete a computer code was written, which describes the free convection in a layer of two liquid phases with different physical properties namely the molten core and the molten or dissolved concrete. First it was assumed, that the two phases do not diffusate into one another, but free convection will occur in the whole area. To avoid numerical mixing by the approximation schemes, it was necessary to treat the separation plane between the two phases like a border which is moved due to the convection. In this way we want to calculate the transport of the molten concrete, having lower density than the core material through the layer of the molten core. The properties of this transport will affect the local heat flux density at the bottem which is the most important parameter for the describing of the propagation of the melting front into the concrete. This code is already finished in a version, in which the moving of the phase border can be calculated correctly, however, the local heat flux density to the bottom is not computed up to now.

4.2 Essential results

Figure 1 shows an example of the results taken from the computer code mentioned in part 4.1. It shows the computed local heat flux at the bottom of the reactor pressure vessel during the process of filling up. The molten material is assumed flowing down exactly in the center of the vessel.

The local heat flux density, shown in the figure 1, demonstrates that the heat flux is governed exclusively by the flowing down fluid stream. Only the increase of the local heat flux density near the surface of the molten material shows, that there is a small effect of the free convection on the heat transfer. With increasing height of the melt, the simulated core melt flow falling down into the melt already at the bottom, is decelerated more and more before reaching the bottom of the vessel. For this reason the local heat flux density decreases with increasing fluid height. This strong effect of the fluid flowing down was already confirmed by some experiments.

5. Next steps

In the next steps of this project the prepared experiments for measuring the velocity fields shall be accomplished. Furtheron is planned to expand the computer code for the computations of filling up the process in a way, that takes into account the changes of the physical properties and heat source densities during the melting process. To extend the code for the description of the destruction of the concrete, it is planned to combine the calculated local heat fluxes at the bottom with the melting process of the concrete.

6. Relations with other projects

RS 79: Theoretical core melt cown studies
IKE-Stuttgart 1.10.72 - 31.12.75

RS 154: Investigation of the interaction between molten core material and concrete
KWU Erlangen 1.2.75 - 30.9.76

7. Reference documents

Quarterly reports in the series: IRS Forschungsberichte (German)

Report period:	Apr. 1975 - June 1975	IRS-F-26
	July 1975 - Sept.1975	IRS-F-27
	Oct. 1975 - Dec. 1975	IRS-F-28

8. Degree of availability

The reports are available at the IRS.

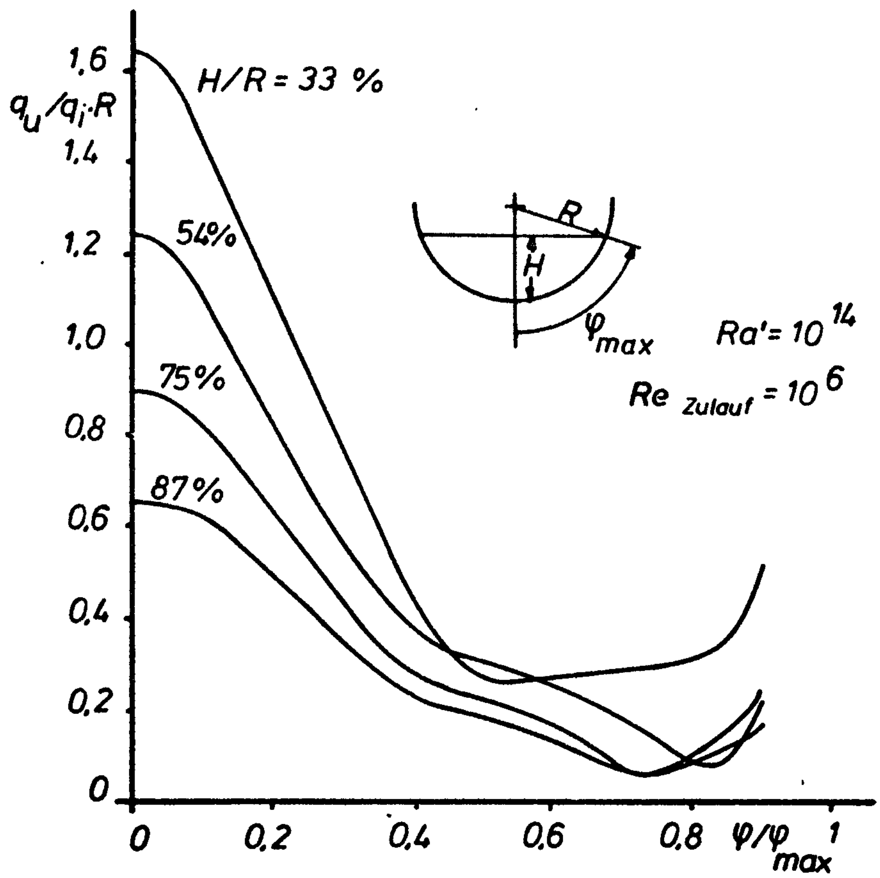


Figure 1: Local heat flux density at the bottom of the reactor pressure vessel

Classification: 2.1	
Title 1 (Original Language): Experimentelle Untersuchung der Abschmelzphase von UO_2-Zircaloy-Brennelementen bei versagender Notkühlung (PNS 4241 - I.1.5., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Experimental Investigation of the Meltdown Phase of UO_2-Zircaloy Fuel Rods under Condition of Failure of Emergency Core Cooling	Project Leader: Dr. S. Hagen
Initiated (Date): 1973 Status: continuing Completed (Date): 1978 Last Updating (Date): December 1975	

1. General aim

Investigation of the course of the melting process including the re-solidification of the melt on colder parts of single rods, single rods with spacers, bundles and bundles with ballooned cans. The influence on the melting process using different parameters like surrounding atmosphere and temperature gradients will be investigated.

2. Particular objectives

During the last year experiments were done with single rods and single rods with spacers in inert gas, air and steam-atmosphere.

3. Experimental facilities and research program

The decay-heat is simulated by electrically heating up a central tungsten rod of 6 mm. This rod is surrounded with UO_2-ring-pellets which replaces the normal pellets. The details of the experimental arrangement and the program are given in earlier reports.

4. Project status

4.1 Progress to date

In 1975 we have done experiments with single rods and single rods with spacers in inert gas, air and steam-atmosphere. To find out the sequence of the interaction between the UO_2-pellets, Zircaloy-can, Inconel-spacer and the oxiding atmosphere, single rods were systematically heated up to maximum temperatures beginning from 1600° C to 2200° C and from 1000° C to 1600° C for single rods with spacers.

4.2 Essential results

The main influence on the meltdown behaviour is arising from the competitive interactions between UO_2-pellets and Zircaloy-can on one side and the Zircaloy and the surrounding atmosphere on the other side. The spacers cause additional interaction between Inconel and Zircaloy and again this interaction is dependent on the oxiding behaviour of the surrounding atmosphere.
From the experiments in inert gas we find that at 1850° C, the melting point of the Zircaloy, the interaction between can and pellets is so strong, that in addition to the Zircaloy also the UO_2-pellets are molten down. The melt is running down and refreezes on the colder parts of the rod.

In air, during heat up, an oxidative skin is formed, which surrounds the inner parts of the rod, this is the residual Zircaloy and the UO_2, which is melting together again at 1850° C. The thickness of the oxid-layer is depending on the heating rate, becoming thicker if the heating rate is smaller. The surface is orange peel like and at temperatures nearly 1800° C little holes and crakes are formed and the melt comes out and runs down slowly along the skin. Experiments were done up to the surface temperature of the oxid-layer of 2200° C and still at this time the skin had a solid form.

In steam atmosphere an oxidative layer is formed again. But now the surface is very smooth. The layer seems to be very strong, for no holes and crakes are formed compared to the heat-up experiments in air. The skin remained completely undisturbed up to 2000° C, where it decomposed explosively in relative large pieces, presumably caused by the pressure of the inside formed melt.

The first experiments with spacers again show the influence of the oxidizing atmosphere on the meltdown behaviour. In inert gas already at ca. 1000° C the Inconel of the spacers starts to interact with the Zircaloy and perforates the can. With increasing temperatures larger regions of the can begin to melt, forming an alloy of Zircaloy with Inconel. Since the melt runs down the can, this process is not restricted to the region of the spacer.

Using air and steam atmosphere this interaction of the Inconel-spacer w th the Zry-can could not be perceived. A reason for this can be the formation of an oxide-layer between spacer and can. Until now experiments in air and steam atmosphere have been conducted only up to 1300^{o} C and heating rates of $0,5^{o}$ C/sec.

5. Next steps

The experiments with spacers in inert gas will be executed by using longer rods, to investigate how far the melt will run. In air and steam - atmosphere experiments with spacers will be done at higher temperature and higher heating rates.
The investigations on single rods in steam will be done with different heating rates and different quantities of steam.
Parallel to these experiments we are working on the development of a fuel rod simulator, to avoid the heating up by means of a tungsten rod.

6. Relation with other projects

PNS 4244, PNS 4245

7. Reference documents

Semi annual reports of PNS
IRS-F-reports

8. Degree of availability

Unrestricted distribution

Classification: 2.1	
Title 1 (Original Language): Kernschmelzen: Messung von Stoffwerten von flüssigen Reaktorcorematerialien (RS 80 - I.1.5., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** EURATOM, Ispra
Title 2 (english): Core Melting - Measurement of Physical Properties of Liquid Reactor Core Materials	**Project Leader:** R. Palinski
Initiated (Date): 1.12.1972 **Completed (Date):** 31.12.1975 **Status:** continuing **Last Updating (Date):** December 1975	

1. General aim

Core melting - Measurement of physical properties of liquid reactor materials

2. Particular objectives

Measurement of the viscosity and surface tension of core melts

3. Experimental facilities and program

- Measurement of viscosities
  - Method: Oscillating crucible
  - Materials: UO_2 and mixtures of UO_2, Zircaloy and steel ("Corium") (homogeneous phases only, to be identified by research project RS 74a).
- Measurement of surface tensions
  - Method: Maximum bubble pressure
  - Materials: Same as above

4. Project status

4.1 Progress to date

In the reference year the work was concerned with the measurement of the viscosity of the components of the core melt, namely UO_2, steel 1.4550 and zircaloy 4. Furthermore, a method for making samples of the core melt was developed and their viscosity was measured.

The modification of the facility for measuring surface tensions was terminated. As the depth of immersion of the capillary tip into the melt is required for the evaluation of the results, it is necessary, when immersing the capillary tip into the melt, to detect exactly the moment of the first contact with the surface. For this purpose a device was developed and tested which signalizes the contact both optically and accustically. For temperature testing experiments up to 2000°C have been made with the surface tension measurement facility. Further test measurements up to 3000°C are under preparation.

4.2 Essential results

Measurements of the viscosity of zircaloy 4. The average of three measurements in TaC-graphite crucibles was 4.7 cP at the melting point.

Steel 1.4550 and its components. The viscosity of Armco iron at the melting point (1510°C) is 7.15 cP according to the only hitherto known measurement /1/. This relatively high value, compared to the viscosity of pure iron (5.54 cP at 1532°C /1/ , 4.72 cP at 1536°C/4/), is attributed to the influence of the oxygen in the Armco iron. Our measurement of Armco iron resulted at the melting point a viscosity of 6.2 cP. The chemical analysis of the oxygen content of the utilized material is intended to give an indication whether this deviation may be due to a difference in oxygen concentration.

The viscosity of nickel at the melting point can be considered as established, since the values of /1/, /2/ and /3/, obtained with two different measurement methods, agree within 4%. According to /1/ the viscosity is 4.98 cP. Our control measurement resulted a value of 4.81 cP.

The measurements of the viscosity of steel 1.4550 gave at about 1460°C the values 6.0 cP and 6.22 cP.

Fabrication of core melt samples and measurement of their viscosity. The necessary samples were made as follows: in order to guarantee a simultaneous melting of the components and to reduce the reaction of the melt with the crucible, pellets were pressed from powder of UO_2, zircaloy and steel in corresponding weight distribution and sintered 10 minutes at 2300°C during the heating-up process for measuring the viscosity.

For determining the viscosity of the core melt in TaC the measurements were made in TaC-graphite crucibles (because of the small wall thickness of the TaC crucible of only 0.7 mm, necessary in order to obtain during the carburization of the Ta a homogeneous composition of the TaC, the TaC crucible was imbedded into a graphite crucible which guaranteed sufficient mechanical solidity).

For the determination of the viscosity of the core melt in ThO_2, crucibles of the company Desmarquest with high density and high purity were available. As in our experimental facility the outer diameter of the ThO_2 crucibles must not exceed 20 mm, only about 30 g of the core melt could be contained. For a sufficiently precise measurement, however, at least twice this quantity is necessary. Therefore two ThO_2 crucibles were inserted, one above the other, into a TaC-graphite crucible as described above. In the temperature range of the measurements the TaC crucible is well compatible with the ThO_2 crucible and prevents a reaction between ThO_2 and graphite. The lower ThO_2 crucible is closed by the bottom of the upper one. The upper one has a lid of ThO_2 onto which a further lid of TaC is placed. In order to solidify this system of super imposed crucibles by some stress in axial direction and to compensate the thermal expansions, a graphite felt was inserted between the TaC lid and a graphite lid which closes the graphite crucible. During a series of preliminary tests this arrangement of crucibles prooved to be the most suitable. Three viscosity measurements have been made in this way, determining the apparatus constant before each measurement.

For the calculation of the viscosity from (1) the density of the core melt in the liquid state must be known.

$$\eta = \frac{1}{\gamma}\left(\frac{\Delta\delta}{K}\right)^2 \quad (1)$$

η - Viscosity

γ - Density of the core melt

$\Delta\delta$ - Change of the logaritmic decrement of the amplitudes during transition from the solid to the liquid state

K - Apparatus constant

As the density of the core melt is not known so far, the result of four measurements is quoted as product $\eta\gamma$.

η_{WF} - Viscosity of the core melt during formation.

η_{NF} - Viscosity of the core melt after formation.

t - Measuring time.

Table 1

	$\eta_{WF}\,\delta$	t (min)	$\eta_{NF}\,\delta$	t (min)	core melt quantity (g)
I Measurement in TaC-crucible	0.437	-	-		62.4
II " " ThO_2 "	0.437	2	0.214	2.5	70.8
III " " " "	0.380	-	0.197	5	60.8
IV " " " "	0.420	4	0.221	2.5	71

Measurement II during the formation extended over 2 minutes and represents the average over 16 periods of the damped oscillations. The temperature increase during this time was of about 10°C. The investigated core melt was composed of the main components 35 w/o UO_2, 10 w/o zircaloy 4 and 55 w/o steel 1.4550 which according to /5/ at 2400°C are all liquid. At this temperature the formation of corium E is terminated after approximately 2 minutes. The values $\eta\delta$ were measured during this formation time. After terminated formation dropped by about 50% (measurements II, III and IV). During the measurement II after terminated formation the temperature increased by another 10°C. The strong decrease of $\eta\delta$ after terminated formation cannot be explained by this temperature increase, but is presumably the result of a change of the melt by the formation.

Assuming that the density of the liquid core melt lies about 15% below the density at 20°C, the following viscosity values are obtained:

Table 2

	η_{WF} (cP)	η_{NF} (cP)
I	5.7	-
II	5.7	2.8
III	5.0	2.6
IV	5.5	2.9

The arithmetic average is $\eta_{WF} = 5.5$ cP
respectively $\eta_{NF} = 2.8$ cP

References

/1/ Hiebler, H. and H. Trenkler: Berg- und Hütte. Monatshefte 12 (1967) P. 150/163.

/2/ Cavalier, G.: C. R. Acad. Sci. 256 (1961) P. 1308/1311.

/3/ Schenk, H., M. Frohberg and K. Hoffmann: Arch. Eisenhüttenwes. 34 (1963) P. 93/99.

/4/ Thiele, M. : Dissertation Techn. Univ. Berlin 1958, Inst. f. Metallkunde.

/5/ Peehs, M., "Investigations of Molten 'Corium' Phases", Thermodynamics (Proc. Symp. Vienna, 1974) 1, IAEA, Vienna (1975) 355.

5. Next steps

Measurement of viscosity of core melts and their components	January - December 1976
Measurements of surface tension of core melts and their components	January - December 1976

6. Relation with other projects

RS 71: "Research and development studies on the measurement of molten reactor core materials, compatibility studies between these materials and crucible materials", BATELLE-Institute e.V.

RS 74a: "Research project core melting: 5. investigation of the metallurgical interaction between melt and reactor pressure vessel wall", KRAFTWERK UNION AG.

7. Reference documents

Quarterly reports in the series IRS-FORSCHUNGSBERICHTE

Report period	Oct.	1973 - Dec.	1973	IRS - F - 19
" "	Jan.	1974 - March	1974	IRS - F - 20
" "	Apr.	1974 - June	1974	IRS - F - 21
" "	July	1974 - Sept.	1974	IRS - F - 22
" "	Oct.	1974 - Dec.	1974	IRS - F - 23
" "	Jan.	1975 - March	1975	IRS - F - 25
" "	Apr.	1975 - June	1975	IRS - F - 26
" "	July	1975 - Sept.	1975	IRS - F - 27
Annual report	Dec.	1972 - Oct.	1973	IRS - F - 18
Annual report	State:	End of	1974	IRS - F - 24

8. Degree of availability:

Restricted

Classification: 2.1	
Title 1 (Original Language): Eigenschaften von Coreschmelzen (PNS 4245 - I.1.5., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Properties of Core Melts	**Project Leader:** Dr. G. Ondracek
Initiated (Date): 1975 **Status:** continuing	**Completed (Date):** 1978/79 **Last Updating (Date):** December 1975

1. General aim

The general aim is to provide data on the properties of materials that enables the formulation of a code to describe a core meltdown accident and the behaviour of the molten Corium in the reactor pressure vessel.

2. Particular objectives

Preparation of samples of various types of Core melts (Corium) that arise under varying oxidation conditions. Determination of properties such as thermal conductivity, density, interfacial energies and thermal expansion.

3.1 Experimental facilities

Available equipment includes a dilatometer, an apparatus for measuring contact angles via sessile drop technique. Furthermore experimental facilities are available for the melting and powder metallurgical consolidation of samples as well as measurement of thermal conductivity.

4.1 Progress to date

Available experimental and theoretical results were critically reviewed in view of estimating properties of melts such as thermal conductivity, viscosity, and heat capacity. On the basis of these results an experimental programm was initiated to cover those areas where the theoretical approach needs to be verified by measurements.

4.2 Essential results

The structural characteristics of the interaction products between the materials involved (UO_2-Zry-Steel) are dependent on the concentration of the components as well as on the oxidation potential of the environment. The composition of Corium A was settled by assuming that the fuel pins, spacers and core support plate are molten, whereas Corium E involves additional structural materials. The indexes X_1, X_2, X_3 refer to non, partially, totally oxidizing conditions. Depending on the latter a single or two immiscible melts can occur. Their estimated properties are summarized below.

CORIUM MELT TYPE		MELTING POINT (K)	HEAT CAPACITY at the melting point (cal/g K)	VISCOSITY (cP)	THERMAL CONDUCTIVITY (at the melting point) ($\frac{cal}{cm\ K\ sec}$)
AX1	Metal-Phase	∿2275	0.085	3.4 (at ∿2675 K) 5.4 (at ∿2275 K)	
	Oxide-Phase	∿2675	0.071	5.7 (at ∿2675 K)	
EX1		∿2275	0.129	5.4 (at ∿2275 K)	0.049
EX2	Metal-Phase	∿1825	0.146	2.1 (at ∿2675 K) 5.4 (at ∿1875 K)	0.044
	Oxide-Phase	∿2675	0.080	5.7 (at ∿2675 K)	0.008
EX3		∿2075	0.202	4.3 (at ∿2075 K)	

5. Next steps

Preparation and characterization of Corium EX_1 and EX_3 and measurement of properties such as thermal conductivity and thermal expansion.

6. Relation with other projects

PNS 4244: Constitution and Reaction Behavior of LWR Materials at Core-Melting Conditions.

RS 80 : Measurement of Viscosity and Surface Tension of Core Componentes and Corium

7. Reference documents

KFK Report 2217 (1975) in German

8. Degree of availability

Unrestricted distribution

Classification
2.1.

Title 1	Country : JRC
Coremelting - Measurement of physical properties of molten reactor core materials	Sponsor : BMFT and CEC
	Organisation : JRC ISPRA Establishment
	Project leader R. Palinski
Initiated : 1.12.1972 Completed : 30.6.1975 Status : Progressing Last updating : March 1975	

1) General aim

Measurement of physical properties of molten reactor core materials to provide data for core meltdown analysis.

2) Particular objectives

Measurement of the viscosity and surface tension of core melts

3) Experimental facilities and programme

Measurement of viscosities
Method : Oscillating crucible
Materials: UO_2 and mixtures of UO_2, Zircaloy and steel ("Corium", homogeneous phases only, to be identified by research project RS 74a).

Measurement of surface tension
Method : Maximum bubble pressure
Materials: Same as above

4) Project status

1. Progress to date : The assembly of the apparatus for the viscosity measurements was completed. The first preliminary tests have shown that a number of modifications of the apparatus were necessary in order to improve the precision of the measurements, the maximum temperature and the safety of the installation. The following improvements have been accomplished : The period of oscillation has been increased up to about 10 sec in order to decrease the danger of appearance of turbulence in the core melt. The best solution proved to be an increase in the length of the torsion wire and the corresponding displacement of the point of attachment. Because of this modification, the charging of the apparatus also became simpler.

   - A cooling system has been installed in order to limit the heating of the torsion wire and the resulting variations of the period of oscillation and of the damping.
   - An absorption filter has been installed in the gas outlet to avoid the dispersion of Corium.
   - The laser beam shielding has been improved.
   - A new high temperature furnace has been installed.
   - The counting chains have been modernized.

   The construction of the apparatus for the measurement of the surface tension was completed. The first preliminary tests with liquid silver have given satisfactory results and modifications have essentially been completed.

2. Essential results : The modified measuring stand for the viscosity was tested. After completing calibration measurements for the determination of the constants of the apparatus, viscosity measurements were made with the following components of the core melt : UO_2, Zircaloy and Steel N. 14450.

Results obtained :

Material	Crucible Material	Viscosity at Melting Pt (cP)
$UO_{2\pm x}$	TaC	about 37
Steel No. 14450	Al_2O_3	" 3.8
Zircaloy	TaC and Al_2O_3	" 0.6

The measured value of the viscosity of UO_2 lies considerably above the value of 7 cP obtained by Tsai and Olander [1] and corresponds more closely to the value of 36-46 cP, measured by Bates et al., [2] and to the value of 25 cP assumed in the safety studies of Argonne [3].
The measured value of the steel is of the order of magnitude of iron (5 cP at the melting point) [4].
The measured value of Zircaloy of about 0.6 cP, obtained with an Al_2O_3 crucible, is unexpectedly small. Measurement with a TaC crucible however gave a similar result.
These data are to be considered as preliminary because each result has been derived from a single measurement only. The reproducibility of the results will be checked later.

[1] H.C. Tsai and D.R. Olander : "The viscosity of molten uranium dioxide", J. Nucl. Mater. 44 (1972) 83 - 86.

[2] J.L. Bates, C.E. McNeilly and J.J. Rasmussen, Material Science Research 5 (1970) 11.

[3] Argonne National Laboratory, Reactor Development Program Progress Report, ANL-7872, S. 8.1 (October 1971).

[4] K. Schäfer : "Eigenschaften der Materie in ihren Aggregatzuständen", 5. Teil, Bandteil a, Transportphänomene I (Viskosität und Diffusion), Springer (1969).

5) Next steps : Measurement of viscosity of core melts and their components. Continuation of preliminary testing of the surface tension measurement apparatus. Measurement of surface tension of core melts and their components.

6) Relation with other projects : There is a close relation with the following BMFT contracts (RS) :

RS 71 : "Research and development studies on the measurement of molten reactor core materials, compatibility studies between these materials and crucible materials", BATTELLE-Institut e.V.

RS 74a: "Research project core melting : 5. investigation of the metallurgical interaction between melt and reactor pressure vessel wall", KRAFTWERK UNION AG.

7) Reference documents :

Quarterly reports in the series IRS-FORSCHUNGSBERICHTE

Report period	Oct. 1972 - Dec. 1972	IRS - F - 14
"	"	Jan. 1973 - March1973	IRS - F - 15
"	"	Apr. 1973 - June 1973	IRS - F - 16
"	"	July 1973 - Sept.1973	IRS - F - 17
"	"	Oct. 1973 - Dec. 1973	IRS - F - 19
"	"	Jan. 1974 - March1974	IRS - F - 20
"	"	Apr. 1974 - June 1974	IRS - F - 21
"	"	July 1974 - Sept.1974	IRS - F - 22

JRC Safety Programme Progress Report 1974

8) Degree of availability : Freely available

9) Budget : The expected total investment including the cost of the facility and the running costs are :

BMFT : about 140.000 UA

CEC : " 53.000 UA

10) <u>Personnel</u> : 3.7 men/year

11) <u>Additional information</u> :

Classification: 2.2	
Title 1 (Original Language): Experimentelle Untersuchung der Dampfexplosion (RS 76 - I.1.5., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: EURATOM CCR Ispra
Title 2 (english): Experimental Investigation of the Vapour Explosion	Project Leader: H. Kottowski F. Tosselli K. Unger
Initiated (Date): 1.11.1972 Completed (Date): 1976 Status: continuing Last Updating (Date): December 1975	

1. GENERAL AIM

The aims of the theoretical and experimental program performed in the laboratories of the CCR Ispra and the T.U. Stuttgart are the investigation of the process of thermal interaction of molten reactor core materials and the coolant as well as the estimation of the consequences of vapour explosions in a core.

2. PARTICULAR OBJECTIVES

In particular, it is intended to study the thermal interaction between molten UO_2, molten structural materials (Zircaloy, Stainless Steel) and water. The subjects of special interest are:
- The theoretical investigation of the fuel/coolant interaction
- Basic oriented experimental studies of the factors influencing the process of interaction
- Experiments simulating reactor-like conditions with respect to the geometry and quantity of molten material.

The basic questions arising in a situation where molten materials come into contact with the coolant are to evaluate the generated mechanical work, in what manner it is released and what are the mechanisms determining the fragmentation of the molten materials.

3. EXPERIMENTAL FACILITIES AND PROGRAM

3.1. Experimental facilities

Because of the difficulty of modelling the fuel/coolant interaction, two test facilities have been built:

a) a monodimensional shock tube test rig for the study of the thermodynamic factors and

b) a tank test facility for the simulation of reactor - like conditions at least as far as the geometry conditions are concerned.

3.1.1. Shock tube test rig

The test rig consists of an interaction chamber where the melting crucible (diameter: 20 mm, length: 80 mm) is located, a channel (diameter: 9 mm, length:2000mm) which is separated from the crucible by a valve and an expansion vessel simulating the gas blanket.
The instrumentation equipment makes it possible to measure the pressure and production of vapour as a function of time and the velocity of the displaced liquid column.

3.1.2. Tank experiment

The tank facility consists of three main components:
- the resistance heated furnace
- the lock-tube system between the furnace and the interaction vessel with the crucible catcher
- the interaction vessel, which is coupled with a storage tank and a filter system for the extraction of the fuel debris from the water.

3.2. Program

It is planned to study the thermal interaction process theoretically and experimentally of:
- stainless steel and water
- Zircaloy and water
- UO_2 and water

3.2.1. Theoretical studies

The aim of the theoretical studies is the development of interaction models and calculation codes for the estimation of the conversion rate of thermal energy into mechanical work and pressure load of the reactor vessel.

3.2.2. Experimental studies

The detailed test program has been discussed and defined in the frame of the expert group "Dampfexplosion AG". In the following the matrix of the experimental program is listed:

Parameters determined by the test facilities

Melting material	stainless steel DIN 4550	Zircaloy 4	UO_2
Quantity melt channel exp.	~ 120 gr	~120 gr	~ 150 gr
Quantity melt tank exp.	~ 3 kg	~ 3 kg	~ 4 kg
Temperature of the melt	~1500°C	~ 1900°C	~ 2900°C
Gas content of the melt	unknown	unknown	unknown
Gas content of the H_2O	saturated at 100°C	saturated at 100°C	saturated at 100°C

Experimental test matrix

1) - temperature H_2O - system pressure with inert gas	20°C 1 bar	20°C 1 bar	20°C 1 bar
2) - temperature H_2O - system pressure with inert gas	80°C 1 bar	80°C 1 bar	80°C 1 bar
3) - temperature H_2O - system pressure with inert gas - partial pressure inert gas	220°C 25 bar 1,5 bar	220°C 25 bar 1,5 bar	220°C 25 bar 1,5 bar
4) - temperature H_2O - system pressure with inert gas	20°C 25 bar	20°C 25 bar	20°C 25 bar

3.2.2.1. Basic oriented shock tube experiments

These experiments are performed in order to provide data in a mono-dimensional test device to check and to adjust the physical interaction model and the calculation code. The factors measured directly during the experiments as a function of time are:

- pressure excursion
- velocity of the displacement of the liquid in the channel
- vapour production

Parameters which can be adjusted for the various tests are:

- coolant temperature
- mass of molten material (up to 150 gr)
- system pressure (up to 25 bar)

3.2.2.2. Experiments in the tank test facility

The goal of these experiments is the study of the thermal interaction of "large" quantities (up to 4 kg) of fuel and coolant in a confinement simulating the reactor vessel. The factors measured directly during the experiments are:

- the pressure excursions and strains as a function of time
- the temperature history in the tank during the interaction
- attempt to visualize the fragmentation history using a high speed camera.

4. PROJECT STATUS

4.1. Progress up to date

4.1.1. Theoretical work

The theoretical work was concentrated on two subjects which have been performed in collaboration with the University of Stuttgart:

a) Development of a twodimensional computation code with regard to the modelling of the channel experiments

b) Development of a theoretical or semi-empirical model of the fragmentation mechanism during the interaction process.

The fuel/coolant interaction code developed for UO_2 and sodium interactions was modified.

The setting up of the equation of state of water above the critical point (up to 2500°C and 10000 bar), which is absolutely necessary for the description of the first fuel-coolant contact and the vapour explosion process, was terminated. The testing of the modified code started.

As far as the fragmentation is concerned theoretical investigations are underway to determine its boundary conditions and propagation mechanisms. The development of a correlation of the "vapour-bubble-collaps" model is executed which most probably might be the dominant fragmentation process in a tank geometry.
A status report is in preparation.

4.1.2. Channel experiments

a) Experiments have been executed with stainless steel corresponding to the points 1 and 4 of the test matrix quoted above
b) Power characteristics tests for melting Zircaloy have been done.

4.1.3. Tank experiments

a) Mounting and adjusting of the experimental facility has been terminated
b) Experiments with molten stainless steel have been performed corresponding to the points 1, 2 and 4 of the test matrix
c) Melting tests with Zircaloy have been done in Zirconoxide crucibles. Difficulties arised because of their poor mechanical resistance. Actions have been initiated to substitute them by more suitable ones (Tantalcarbide)

4.2. Essential results

4.2.1. Channel experiments

The experimental conditions for the evaluated tests were the following: coolant temperature: 20°C; pressure in the blanket vessel: 1 bar (Argon); quantity melt (stainless steel DIN 4550): 130 gr, 120 gr, 110 gr. Fig. 1, 2 and 3 show the pressure history of the three evaluated tests. The figures display the pressure measured downstream the interaction crucible (curve p). Fig. 3 displays furthermore the reaction force of the crucible measured under the crucible with the help of a force transducer.

The quoted tests differ in the following points:

a) The melt in the test No. 1 corresponding to Fig. 1 was covered by a crust of about 1 mm thickness. During the interaction no fragmentation occurred.

b) The melt of the experiment corresponding to Fig. 2 was covered by solid lumps. Only a gradual increase of surface during the interaction was observed. Large droplets were formed which were squeezed out between the lumps.

c) The experiment displayed in Fig. 3 was executed with a melt of 50°C above the melting point. The melt showed after the interaction a sponge like surface of 5 mm depth. In the melt itself a cavity of about 1 to 1,5 cm diameter about 3 cm penetration depth was formed. The estimated increase of the surface was of about a factor 10. The surface structure is shown in Fig. 4. The pressure peaks measured did not exceed 10 bars. A definite analysis of these experiments will be given after termination of the tests.

4.2.2. Tank experiment

- Experiments:

The first series of experiments up to the end of 1975 dealed with the interaction of stainless-steel with water in the following conditions:

S.S. weight:	1500 g
Melt temperature:	1600°C
Water temperature:	20°C and 80°C
Fall-height (melt-H_2O):	30 and 10 cm
System pressure:	1 atm N_2
Gas content of water:	saturated and degassed at 100°C

Pressure, temperature and strain recorded as a function of time.

All experiments have shown only a slight fragmentation and consequently the pressure rises were small (less than 1 bar). Fig. 5 and 6 show the pressure rise during the interaction of molten stainless-steel with H_2O at 20° resp. 80°C. Fig. 7 shows the interaction debris of one experiment at 20°C.

5. NEXT STEPS

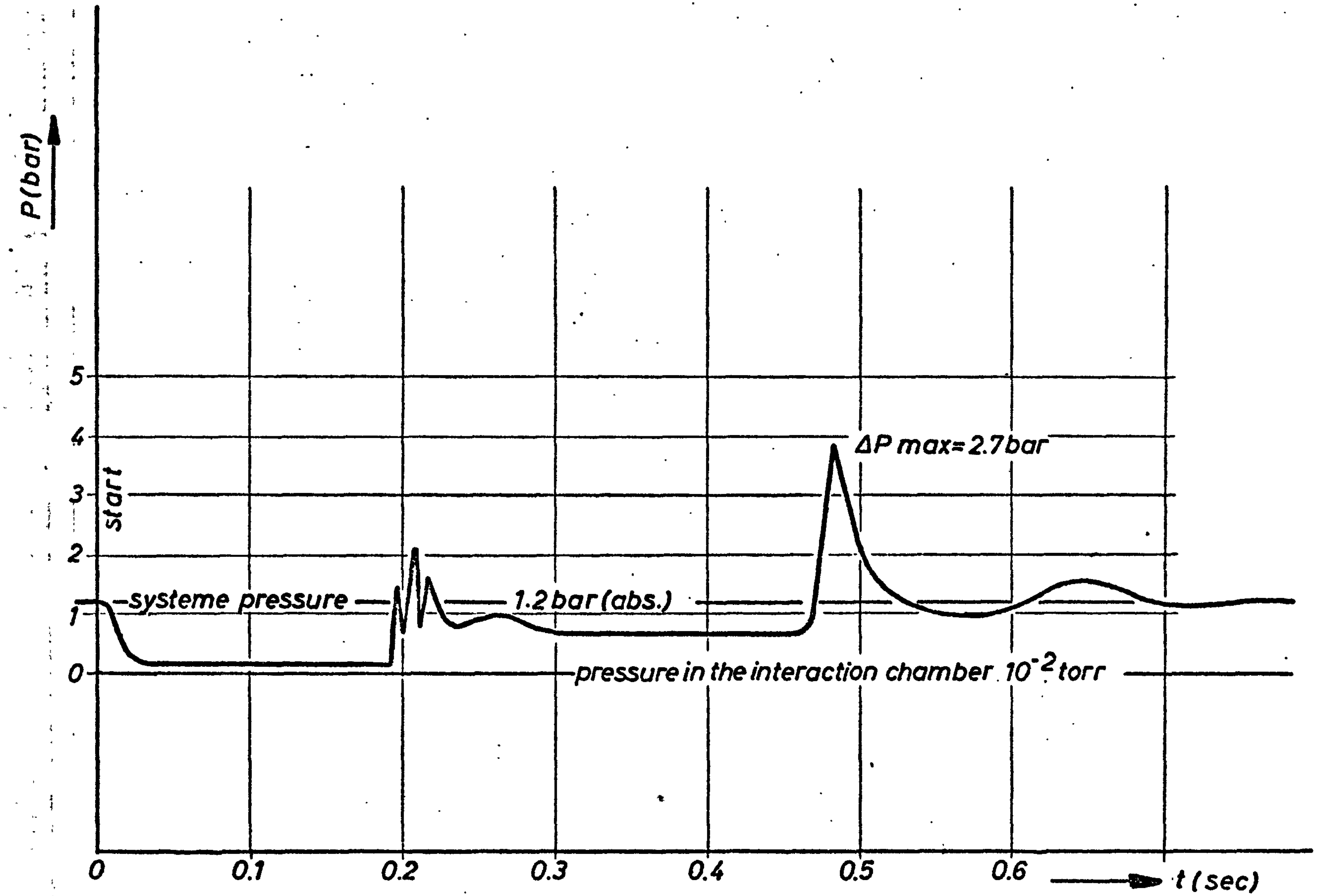
- Termination of the interaction tests with molten stainless steel and H_2O
- Initiation of the tests with Zircaloy and UO_2.

6. RELATIONS WITH OTHER PROJECTS

No new relations with respect to 1974 annual report.

7. REFERENCE DOCUMENTS (1975)

- Quarterly reports
- Calculation Model Code of Fuel/Coolant Interaction
  H. Goldammer, H.M. Kottowski; Working paper on the OECD Meeting on "Calculation Models", Paris 28/29 April 1975 (english) (limited available)
- Zwischenbericht des IKE und dem CCR Ispra über den Stand der begleitenden theoretischen Arbeiten zur Dampfexplosion
  R. Benz, G. Fröhlich, H. Goldammer, H. M. Kottowski
  (in preparation)



PRESSURE P2 f(t)

Fig. 1

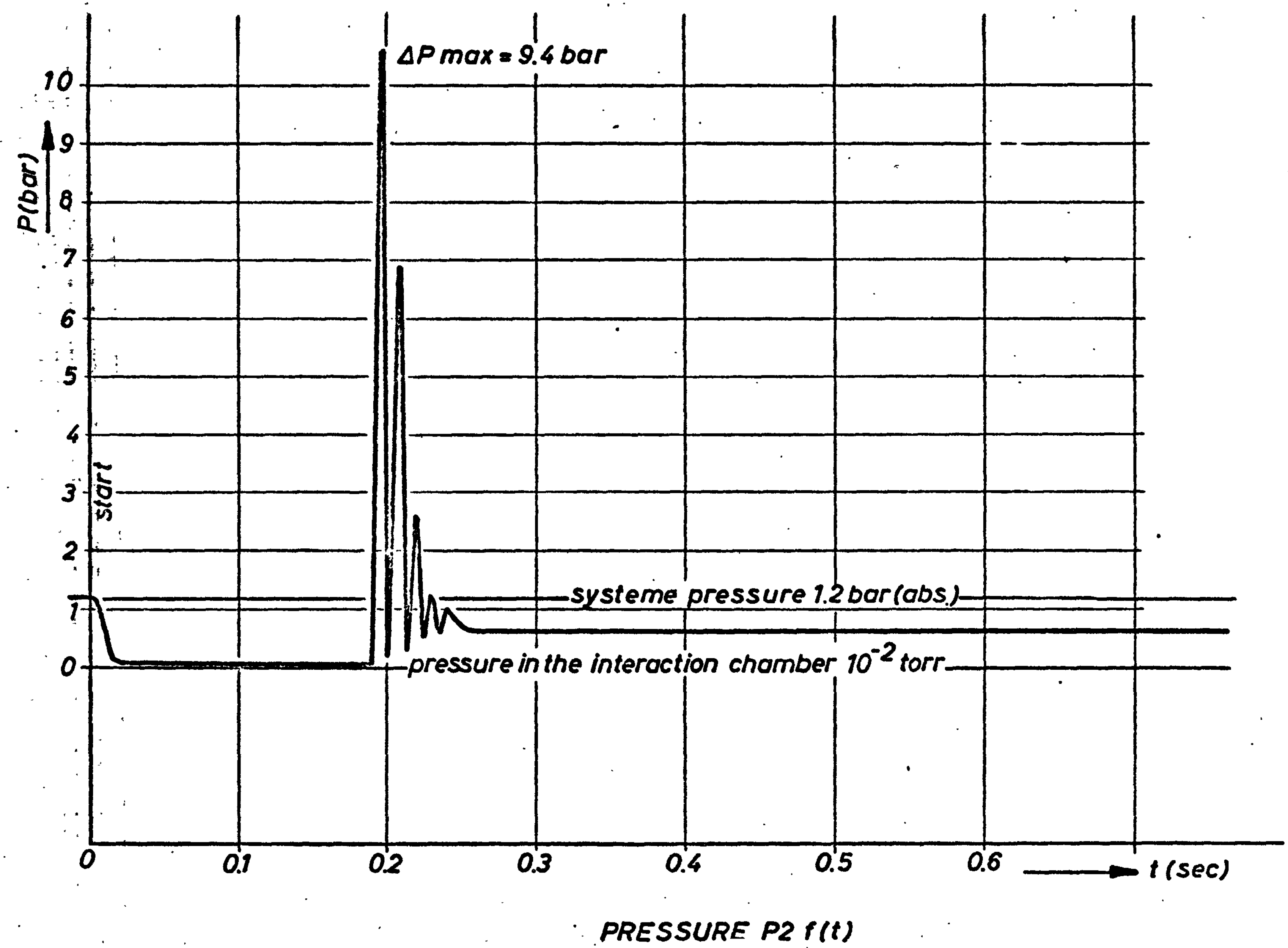
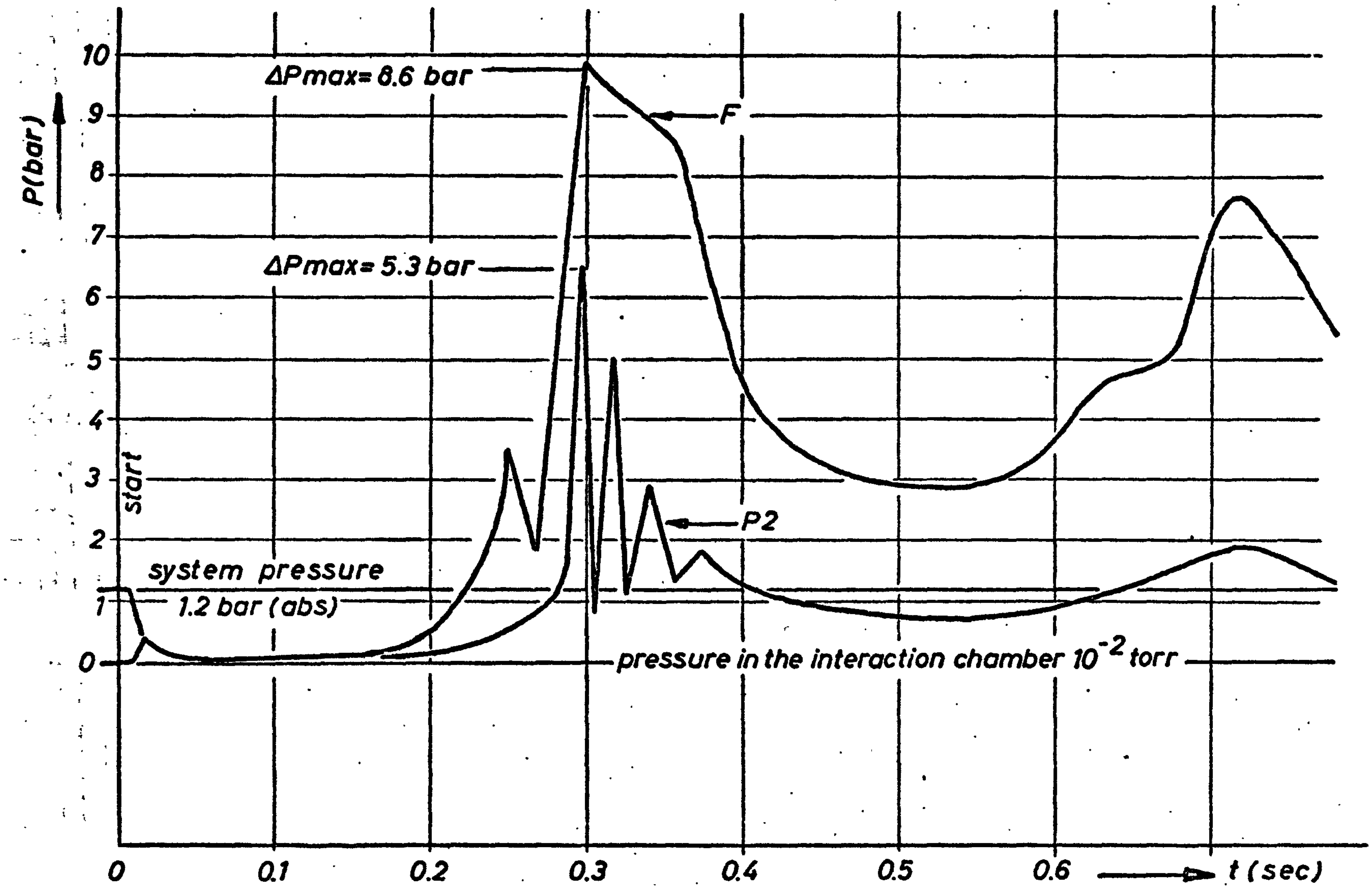

ΔP max = 9.4 bar
P(bar)
10
9
8
7
6
5
4
3
2
1
0
start
systeme pressure 1.2 bar (abs.)
pressure in the interaction chamber 10^-2 torr
0
0.1
0.2
0.3
0.4
0.5
0.6
t (sec)
PRESSURE P2 f(t)

Fig. 2



PRESSURE P2 AND F , f(t)

Fig. 3

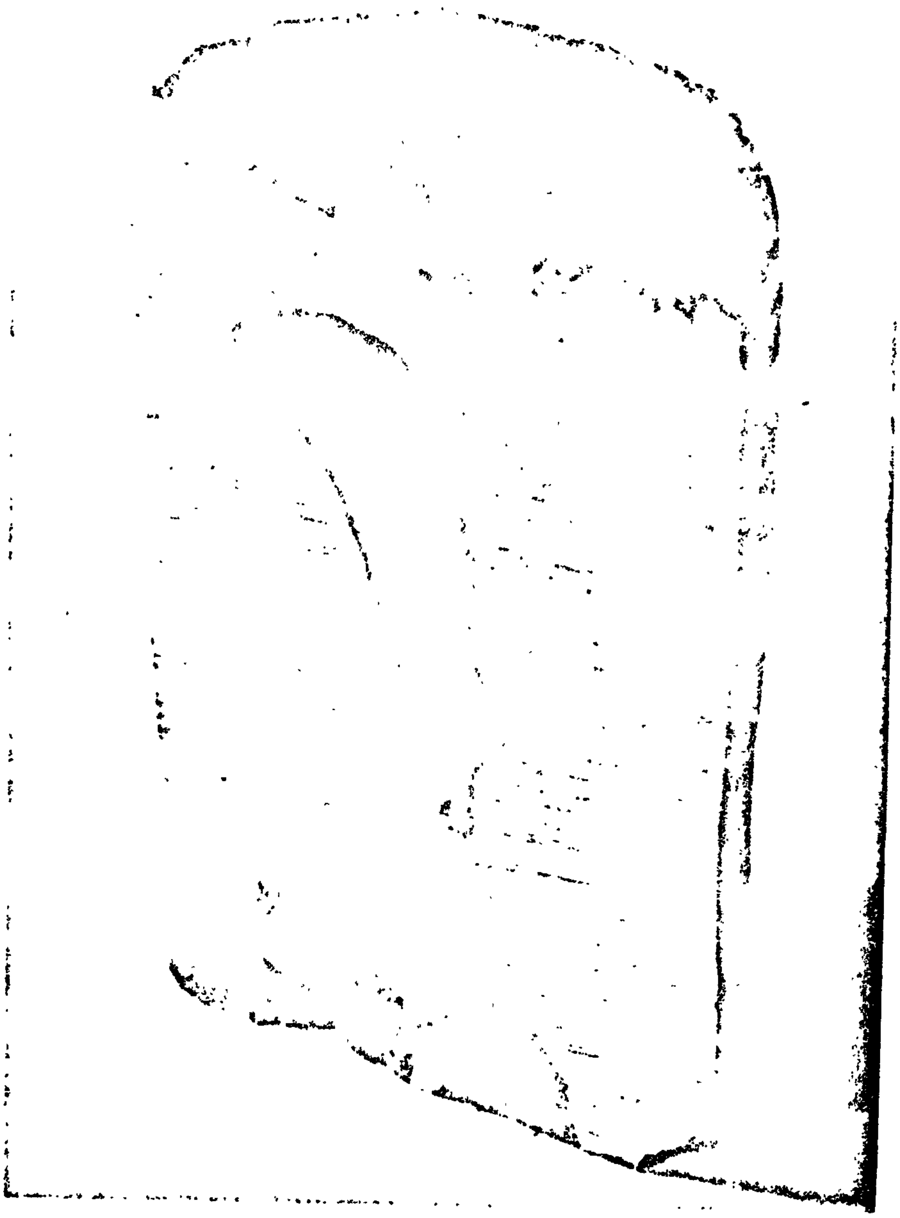
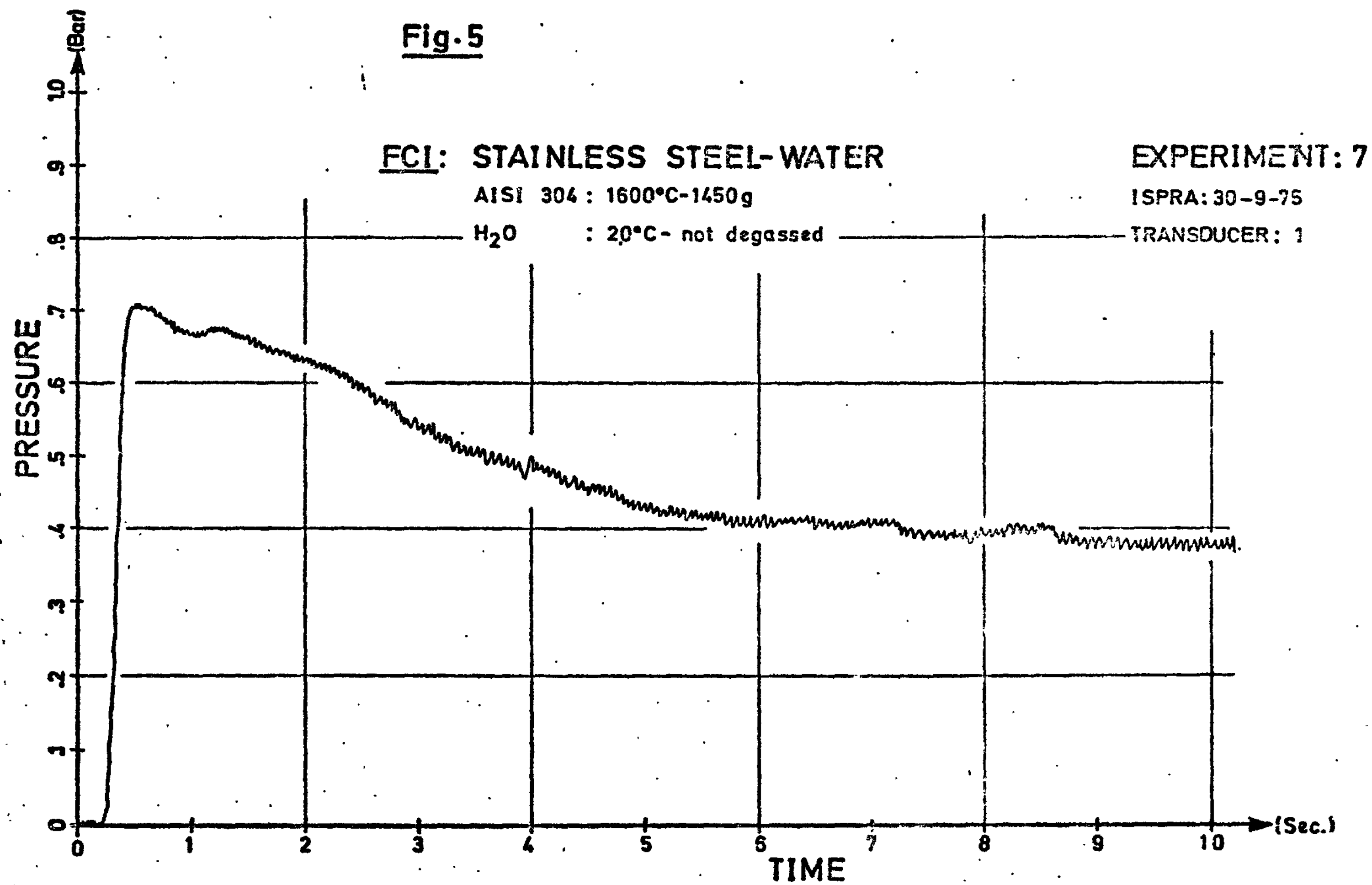


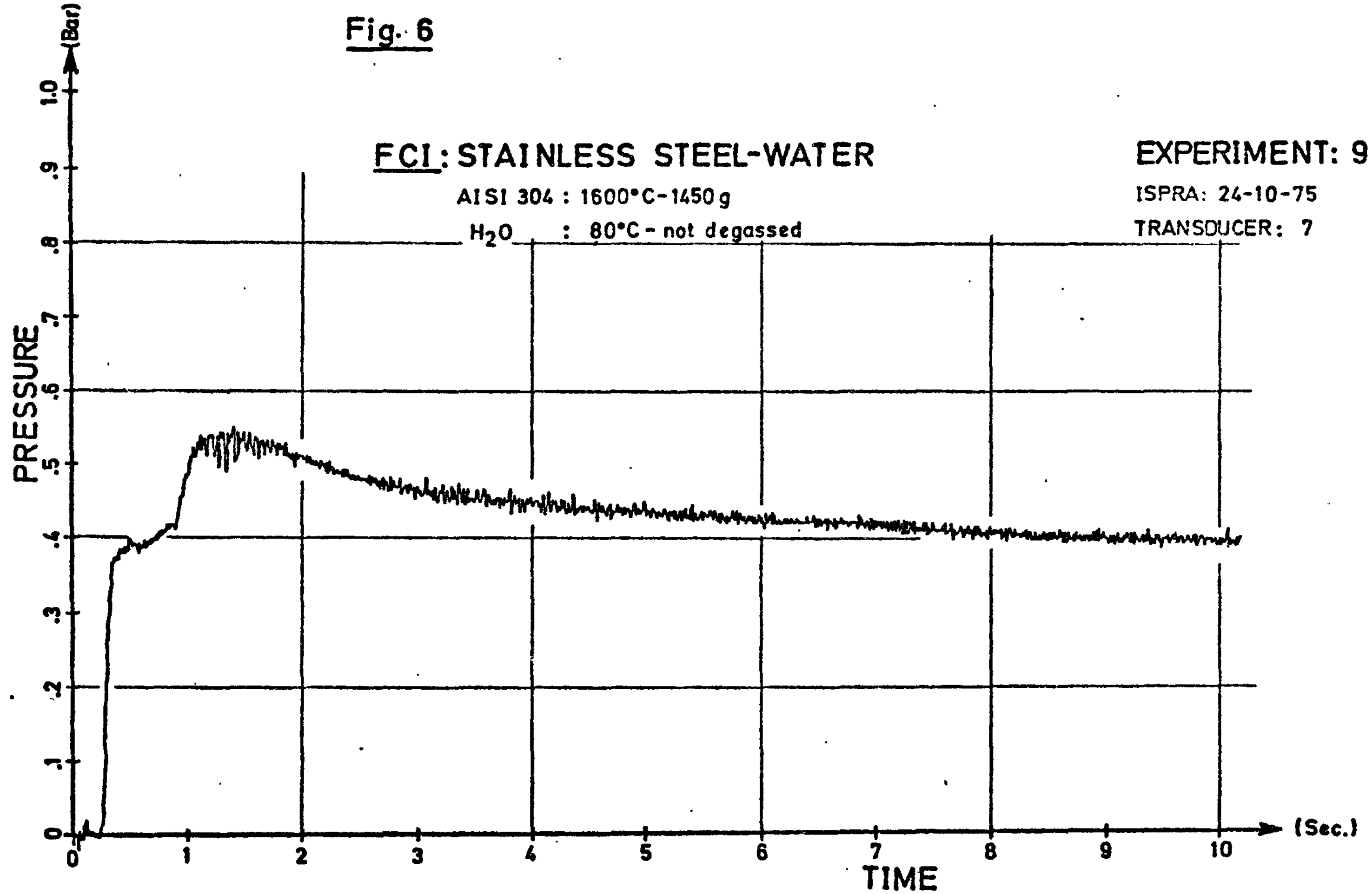
Fig. 4 Surface of the melt

Fig. 5



FCI: STAINLESS STEEL-WATER
AISI 304 : 1600°C-1450g
H2O : 20°C- not degassed
EXPERIMENT: 7
ISPRA: 30-9-75
TRANSDUCER: 1
PRESSURE
(Bar)
1.0
.9
.8
.7
.6
.5
.4
.3
.2
.1
0
0 1 2 3 4 5 6 7 8 9 10
(Sec.)
TIME

Fig. 6



FCI: STAINLESS STEEL-WATER
AISI 304 : 1600°C-1450 g
H2O : 80°C - not degassed
EXPERIMENT: 9
ISPRA: 24-10-75
TRANSDUCER: 7
PRESSURE
(Bar)
1.0
.9
.8
.7
.6
.5
.4
.3
.2
.1
0
0
1
2
3
4
5
6
7
8
9
10
(Sec.)
TIME

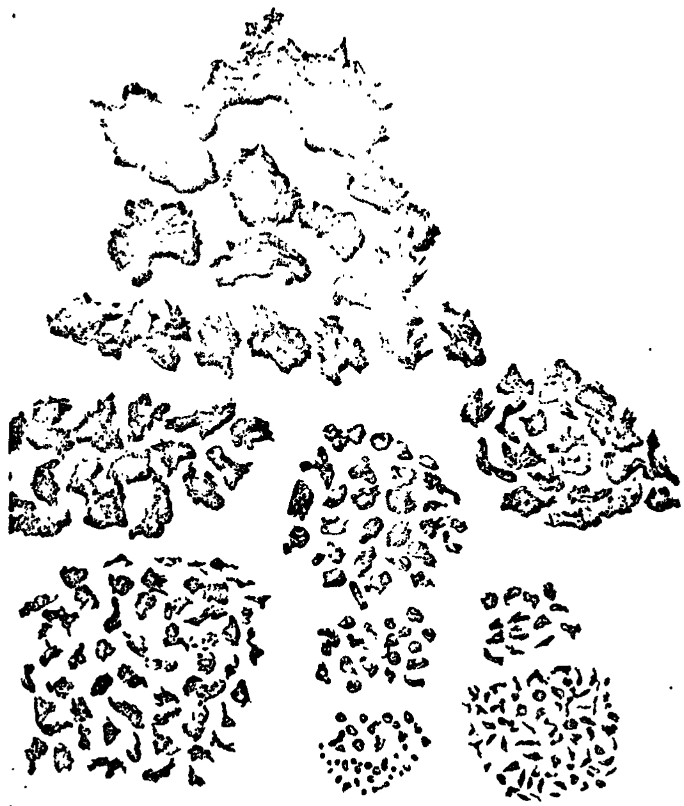


Fig. 7 Surface of the melt (stainless steel, T_{H_2O} : 20^{o}C)

Classification 2.2

Title 1 Fuel-Coolant Interactions (1)		Country UK
Title 2		Sponsor UKAEA Organisation Reactor Laboratory, Windscale
Initiated 1970 Status Continuing	Completed Last updating	Project leaders Dr. H. Lawton

General Aim

To predict and thus contain the mechanical effects following core melt-down.

Particular objectives

To observe the various phenomena when hot and cold liquids are brought into contact, with particular reference to a hot liquid which subsequently freezes.

Experimental facilities and programme

A low temperature rig (limited to about 1000°C) is now operational and initial results have been obtained using cold water and Bi_2O_3. Peak pressures in the range 1000-2500 psi have been recorded, with lower pressures in other pulses. A single test using tin just above the melting point has been carried out with no evidence of any interaction, although this was to be expected from the results of other workers. A further test would be carried out to investigate this unexpected finding.

The rig will be used to investigate a wide range of materials as quickly as possible rather than to investigate one system in depth. Materials to be used include boron dioxide, magnesium and silver, and possibly mercury - molten glass. Battelle has seen evidence of chemical reaction in the Al/H_2O system apparently causing reaction, and this mechanism should be borne in mind.

Project Status

The results of this series of tests will be reviewed about the end of 1975.

Reference documents

Internal documents.

Darby, Pottinger, Rees & Turner. Paper 7 to Crest Meeting on Fuel-Coolant Interaction. Grenoble. January 1972.

Classification 2.2

Title 1 Fuel-Coolant Interactions (2)	**Country** UK
Title 2	**Sponsor** UKAEA **Organisation** Culham Laboratory
Initiated 1972 **Completed** **Status** Continuing **Last updating**	**Project leaders** Dr. T. Dullforce

1 **General Aim**

To predict and thus contain the mechanical effects following core melt down.

2. **Particular Objectives**

To identify and quantify the various phenomena when particular hot and cold liquids are brought into contact.

3. **Experimental Facilities and Programme**

The work uses gram quantities. Heat transfer regimes and dispersion mechanisms are studied. High-speed cine films (500 frames per second) have been made and studied. Initially the system molten/tin distilled water has been studied; other materials are planned.

Reference Documents

Internal documents.

D. Buchanan, T. A. Dullforce, Nature 245, Sept. 1973. Mechanism for Vapour Explosions.

Title 1 Fuel - water thermal interaction	Country: JRC Sponsor: BMFT and CEC Organization : JRC ISPRA Establishment
Initiated : 1973 Completed : December 1976 Status : progressing Last updating : March 1975	Project leader : H. Kottowski

1) General aim

Assessment of possible pressures and mechanical energy releases due to the fuel/coolant interactions accompanying core melt-down accidents.

2) Particular objectives

Collection of experimental data on the thermal interaction of molten fuel (UO_2), or reactor structural materials (stainless steel, Zircaloy, Inconel, etc.) with water. The experimental results will be compared with theoretical model predictions to gain a better understanding of the interaction phenomena.

3) Experimental facilities and programme

Two facilities are approaching completion :

3.1. The Tank Facility

The core-melt material (up to 4 kg) is prepared in a crucible in a furnace (operating pressure up to 25 atm; 3000^{o}C) and dropped through a fall-guide into a reaction tank of 300 l containing 200 l H_2O with a temperature up to 230^{o}C (pressure 25 atm). Instrumentation is provided for the measurement of the pressure and temperature excursions accompanying interaction.

The debris is analysed after each experiment. About 40 experiments will be necessary to cover the range of the various parameters involved.

3.2. The Channel Facility

This facility allows the measurement of the pressure excursion, the displacement of the liquid in the channel and the vapour production as a function of the coolant temperature, the mass of molten material (up to 150 gr) and the blanket pressure (up to 25 bar). The following materials are foreseen to be investigated : stainless steel, Zircaloy, Inconel and UO_2.

4) Project status

1. Progress to date :

The Tank Facility :

- Fabrication and assembling of the supporting structure and the whole circuitry including furnace, interaction tank, pumps, vessels, valves and vacuum system is completed.
- Testing of the interlock system completed.
- Instrumentation for measuring and recording pressures and temperatures during the interaction in the tank has been commissioned.
- Adaptation of a high speed camera in order to visualize the interaction process (at least at low system pressure) is underway.
- Preliminary experimental studies of a filter system to collect and split the debris produced during the thermal interaction are underway.
- The mechanical device for catching and turning the crucible to drop the melt in the centre of the reaction tank has been fabricated and tested.
- A high frequency induction furnace for degassing the core melt materials has been adopted.

The Channel Facility :

- The mounting of the test-rig was started at the end of 1974
- The instrumentation for pressure, temperature and void measurements has been prepared.
- The crucible and heating system for preparing molten UO_2 are being tested in the light of the successful experience on UO_2/Na interactions.
- Theoretical developments for the assessment of "mild" fuel/coolant interactions are underway.

2. Essential results

The first experiments are expected in mid-1975.

5) Next steps : Completion of the test facilities and initiation of the experiments.

6) Relation to other projects :

These studies are part of the "core meltdown" analysis programme of the BMFT. The reference numbers of the project dealing with the same subject are :

RS 72a, RS 726, RS 73, PNS 4241, RS 48/1, RS 74 a, RS 746, PN 4243, RS 71, RS 79, RS 80, PNS 4311, PNS 4242.

7) Reference documents :

G. Fröhlich, H. Kottowski, F. Toselli
Theoretische und experimentelle Untersuchungen über die Wechselwirkung geschmolzener Materialien und Kühlmittel
IRS-Seminar, November 1974

Quarterly progress reports

JRC 1974 Safety Programme Progress Report

8) Degree of availability : Freely available

9) Budget : The expected total cost of the experiment, including the investment and the running costs are :

BMFT : about 123 000 UA
CEC : about 210 000 UA

10) Personnel : 10 men/year

11) Additional information : -

Classification: 2.3	
Title 1 (Original Language): Forschungsprojekt Coreschmelzen: 5. Untersuchung der metallurgischen Wechselwirkung zwischen Schmelze und RDB-Wand (RS 74 a - I.1.5, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Investigation of Metallurgical and Chemical Interactions Between Molten Phases of Reactor-Core-Material and the Reactor Pressure Vessel	Project Leader: Dr. Peehs
Initiated (Date): November 1972 Status: Completed Completed (Date): July 1975 Last Updating (Date): December 1975	

General Aim

Within the research program the core melting-phenomena after the two-fold improbable case of a loss-of-coolant accident and a simultaneous drop out of the emergency cooling system basic data were evaluated for the measurement of material constants of the molten phases and theoretical considerations of the reactor in the post accident stage.

Particular Objectives

It was the objective of these investigations

a) to determine the metallurgical behaviour of reactor-core material during melting and to investigate the constitution of the molten phases.

b) to study the interaction of the molten phases of the reactor core material with the pressure vessel of a LWR.

Experimental Facilities / Research Program

The first subtask in the program was to define the characteristic overall composition of the molten phases of core materials of LWR. As this composition varies from the beginning to the equilibrium stage, the core-melting had to be defined for beginning (Corium A) and for the equilibrium stage (Corium E).

The direct study of "Corium"-melting behaviour as second subtask was to be done by melting-experiments in a crucible within an inertgas atmosphere. Subsequent to the melting experiments metallographic and microprobe analysis as well as remelting experiments by direct microscopic observation had to be carried out to identify liquidus and solidus temperatures of "Corium", the disintegration mechanism of the high-melting components and the phase-composition in the "Corium".

To investigate the interaction of liquid "Corium" and reactor vessel materials as the third subtask a special test facility had to be developed. To achieve representative test conditions the possible quantity of materials to be used for interaction should be in the range of 1 - 2 kg. The foreseen parameters for the experimental study were the temperature gradient across the interaction zone, the temperature and the interaction time.

Project Status / Progress to Date

The investigations of the interaction of Corium A and the RPV wall material and the oxygen influence have been completed experimentally.

Project Status / Essential Results

At temperatures over 2000 ^{o}C the material loss from Corium during its primary liquefaction will be 20 - 25 % of the molten mass. With the molten samples the different evaporation behaviour of the Corium components has been analyzed. The evaporation behaviour of Corium depends on the O-activity of the gas phase and the steel content of the melt.

The melting temperatures of the RPV material in contact with Corium A and Corium E at 1400 °C was experimentally confirmed. Low temperature liquefactions of RPV material due to alloying effects will not occur.

Next Steps

Work on this project has been completed.

Relation with Other Projects

RS 154 Coremelting: Investigation of the Interaction between Molten Core and Concrete

RS 160 Experimental Investigation of the Interaction between UO_2-Steel Molten Core and Graphite

RS 183 Energy Balances after Hypothetical RPV Failure with Respect to Concrete Destruction

Reference Documents / Degree of Availability

No reports available.

Classification: 2.3		
Title 1 (Original Language): Forschungsprojekt Coreschmelzen: 5. Untersuchung der metallurgischen Wechselwirkung zwischen Schmelze und RDB-Wand, Feasibilitystudie (RS 74 b - I.1.5, Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Feasibility Study on Experimental Fuel Element Slumping Investigations With Respect to Coremelting		Project Leader: Dr. Peehs
Initiated (Date): 1. 1. 73	Completed (Date): 30. 4. 75	
Status: Completed	Last Updating (Date): 31. 12. 75	

General Aim

The objective of the R & D - work was to establish a study about the feasibility of large technical experiments to investigate fuel element slumping and the interaction between melt of burned down fuel and vessel wall of reactor pressure vessel material.

Particular Objectives

The feasibility of large technical experiments was to be tested with respect to the melt down of burned out fuel elements in the hot cells. The heat, necessary for the melting down of the fuel, was to be provided by the decay energy of the fission products and secondary heating elements. Another objective was to find out if the volatile fission product release during melt down and fuel slumping. For this purpose the released fission products were to be identified and continuously measured during the experiments.

Experimental Facilities and Program

1) Identification of possible reactor and the fuel element for preirradiation
2) Dimensioning of the test bundle and check of the hot cell equipment available.
3) Design study for the experimental devices.

<u>Project Status/Progress to Date/Essential Results</u>

The BWR VAK is best reactor facility for pre-irradiation of test elements. The proximity of the reactor to the hot cells in Karlstein, allows to start the meltdown test already 100 h after reactor shutdown. For the meltdown experiments the middle part of a segmented element can be used with an active length of 1000 mm. The test facility "BAUTZ" has a total height of 2510 mm. Since the decay heat of a fuel rod of 0.4 % of the max. heat rate, available 100 h after reactor shutdown, is too low for test element slumping in a cool environment, it is necessary to simulate the irradiation equilibrium of an inner element, in the reactor core by auxiliar heating elements. Because of the small decay heat of 0.4 % the support heaters must be heated up to the fuel-element slumping temperature of 2300 °K. Consequently only refractive metal heaters can be used. The meltdown test needs therefore to be executed in an inert-gas atmosphere.

In the design of the test facility "BAUTZ" results of a safety analysis were integrated, resulting, that lack of electric power supply and leakages in the different coolant loops are controllable.

<u>Next Steps</u>

Work on this project has been completed.

<u>Relation with Other Projects</u>

RS 154: Coremelting: Investigation of the Interaction between Molten Core and Concrete

RS 160: Experimental Investigation of the Interaction between UO_2-Steel-Molten Core and Graphite

RS 183: Energy balances after hypothetical RPV Failure with Respect to Concrete Destruction

<u>Reference Documents/Degree of Availability</u>

M. Peehs, K. Mollwitz, W. Würtz
Studie über die Durchführbarkeit von Kernschmelzversuchen mit abgebrannten Brennelementen in einer HEISSEN ZELLE
Abschlußbericht Förderungsvorhaben BMFT RS 74 B
Kraftwerk Union Aktiengesellschaft (Nov. 1975)

Company Confidential

Classification: 2.3	
Title 1 (Original Language): Experimentelle Untersuchung der Wechselwirkung UO_2-Stahl-Kernschmelze mit Graphit (RS 160 - I.1.5, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Experimental Investigation of the Interaction Between UO_2-Steel-Molten Core and Graphite	Project Leader: Dr. Peehs
Initiated (Date): February 1975 Completed (Date): December 1975 Status: Completed Last Updating (Date): December 1975	

General Aim and Particular Objectives

The tests conducted under RS 74 A have shown, that graphite can resist considerably a melting core, which contains zirconium. This effect results from a ZrC-protective layer, which constitutes of the Zr-content of the corium.

For the fast sodium cooled reactor the behaviour of a Zr-free molten corium has to be investigated. Therefore in addition to the results of RS 74 A the barrier properties of graphite will be studied, using a typical fast reactor composition of the molten materials.

Experimental Facilities

Within this task the same experimental facilities are used as with the investigation of the metallurgical and chemical reaction between Corium and RPV-material.

Research Program

1. The chemical reactions in the system UO_2-Graphite will be investigated, using various graphite qualities and gaseous atmospheres.

2. The chemical reactions in the system Steel-UO_2-Graphite will be investigated, using various graphite qualities and gaseous atmospheres.

Project Status / Progress to Date

A standard test was worked out experimentally; the compatibility of UO_2, Steel 1.4981 and UO_2 + Steel 1.4981 on graphite was investigated.

Project Status / Essential Results

UO_2 will be dissolved completely at 1980 °C in an UO_2-Steel-Graphite-system; pure UO_2 pellet is liquidated on graphite at 2375 °C. Microanalysis showed that FeCrNi-, UFe- and UFeCr-phases have been produced. The experimental results show, that a rather strong oxigen loss due to an CO-generation occurs during the melting process. This results together with material evaporation in a rather high weight loss during the UO_2 dissolution. The carbon activity in the melt reaches within minutes its final value. The interaction of the melt and the graphite mass stops.

Next Steps

The tests are completed. A final report with a detailed evaluation of experimental results will be prepared.

Relation with Other Projects

A 74 A: Investigation of Metallurgical and Chemical Interactions between Molten Phases of Reactor-Core-Material and the Reactor Pressure Vessel.

Reference Documents / Degree of Availability

No reports available.

Classification: 2.3	
Title 1 (Original Language): Kernschmelzen: Untersuchung der Wechselwirkung zwischen Kernschmelze und Reaktorbeton (RS 154 - I.1.5; Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Core Melting: Investigation of the Interaction Between Molten Core and Concrete	Project Leader: Dr. Peehs
Initiated (Date): 1. 2. 1975 Status: Continuing	Completed (Date): 30. 9. 1976 Last Updating (Date): 31. 12. 1975

General Aim and Particular Objectives

In continuation of the investigations of the interaction between molten core and concrete the following questions will have to be answered:

- How does typical reactor concrete behave on different temperatures up to its melting point?
- Which metallurgical processes occur when "Corium" contacts concrete?
- What happens with the concrete structure, when a representative mass of corium (kg-range) attacks the concrete under accident conditions?

Experimental Facilities

Within this task the same experimental facilities are used as with the investigation of metallurgical and chemical reactions between Corium and RPV-material (RS 74 a). To investigate the thermal shock behaviour of concrete a plasma torch with an electrical input up to 40 kW will be available.

Research Program

The research program is devided into the following subtasks:

- Compilation of literature on high temperature behaviour of concrete
- Investigation of concrete up to its liquefaction

- Determination of degassing characteristics of concrete
- Thermal shock behaviour of concrete
- Corium concrete interaction
- Theoretical evaluation of the experimental results

Project Status/Progress to Date

A literature study on the properties of concrete has been completed. A standard concrete mixture was evaluated and a procedure for hardening and aging was worked out. Concrete samples were produced for isothermal tests, for dilatometric tests, differential thermoanalysis and thermogravimetric analysis.

The Corium components for interaction tests with the concrete were defined. The test series to investigate Corium/concrete - interaction was started with three melting tests with large samples (in the kg-range) under inert conditions.

Project Status/Essential Results

Differential thermoanalysis and thermogravimetric analysis have shown, that the weight loss of the concrete after heating results from water loss. Up to 120 °C only physically bound water will be lost, at 500 °C and 800 °C chemically bound water.
The dilatometric tests have shown, that the thermal expansion of the concrete will be heavely influenced by the SiO_2-agregates.

Next Steps

- Continuation of the investigations on the high temperature behaviour of concrete
- Continuation of the investigations on the interaction between concrete and molten corium.

Relation with Other Projects

RS 74 a: Investigation of Metallurgical and Chemical Interactions between Molten Phases of Reactor-Core-Material and the Reactor Pressure Vessel

Reference Documents/Degree of Availability

No reports available.

Classification: 2.3	
Title 1 (Original Language): Energiebilanzen nach hypothetischem RDB-Versagen unter Berücksichtigung der Betonzerstörung (RS 183 - I.1.5, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Energy balances after hypothetical RPV failure with Respect to Concrete Destruction	Project Leader: H. Goetzmann
Initiated (Date): 1. 9. 1975 — Completed (Date): 31. 5. 1977 Status: Continuing — Last Updating (Date): 31. 12. 1975	

General Aim

In continuation of the theoretical investigations of the energy balances within a pressure vessel and on the containment walls, the progression of the melt in the concrete structures will be studied.

Particular Objectives

A computer code will be developed to describe the destruction of the concrete. Additionally, the energy balances and the pressure build-up within the containment will be studied, considering the energy and mass transport in the containment atmosphere.

Experimental Facilities

No experimental facility necessary.

Research Program

1. Problem related theoretical investigations:

1.1 Study of the existing knowledge of the available destruction models

1.2 Definition an formulation of the heat transport model

2. Energy balance for the RPV surrounding:

2.1 Definition of the region in which contact with the molten core can occur after hypothetical core melting

2.2 Setting up the energy balances

2.3 Consideration of the conditions which have to be fulfilled in order to keep the molten core as long as possible within the containment

3. Energy balance for the containment after a hypothetical RPV destruction

3.1 Calculation of the energy and mass transport to the containment wall

3.2 Energy balance and computation of the pressure build-up in the containment

4. Sensitivity study regarding the parameters, which influence the accident course

Project Status/Progress to Date/Essential Results

A two dimensional heat conduction model was defined in r-Θ-geometry. In the final stage a coupling of this model will be necessary with the thermohydraulic model of the TU Hannover (RS 166); therefore the geometry and the boundary conditions were harmonized with respect to the coupling. Work on the realization of the heat conduction model has been started.

Next Steps

The heat conduction model will be programmed and tested. The experimental results of RS 154 will be prepared for use in the program.

Relation with Other Projects

RS 72 a Theoretical determination of energy balances for core melting: Balance boundary "Containment walls" for BWR and PWR

RS 72 b Core Meltdown Program
Theoretical Evaluation of the Energy Balances within a Pressure Vessel

RS 154: Core melting: Investigation of the Interaction between Molten Core and Concrete

RS 166: Continuation of the Investigations of the Thermohydraulic Behaviour of the Molten Core

Reference Documents and Degree of Availability

No reports available.

Classification: 2.3	
Title 1 (Original Language): Konstitution und Reaktionsverhalten von LWR-Materialien beim Coreschmelzen (PNS 4244 - I.1.5, Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Constitution and Reaction Behaviour of LWR Materials at Core-Melting Conditions	**Project Leader:** H. Holleck
Initiated (Date): January 1974 **Status:** continuing **Completed (Date):** December 1976 **Last Updating (Date):** December 1975	

General aim

Theoretical and experimental investigations a) of the chemical interactions between core components during melting, b) of the constitution of the melt in dependence on the time after melting, oxidation state and temperature as well as on the content of concrete, c) reaction behaviour of the fission products in the melt.

Particular objectives

1. Phase equilibria in multicomponent systems with core components
2. Reaction studies in complex systems and different arrangements to investigate the miscibility of melts, the oxidation and vaporization behaviour and the distribution of fission products
3. Compositions of phases occurring during the melting process.

Project status

Progress to date

a) The phase equilibrium studies in the ternary systems U-Zr-O, U-Fe-O and Zr-Fe-O are completed. Sections of the quaternary system U-Zr-Fe-O are investigated in the moment.

b) The reaction studies are carried out with Corium A and with a mixture of Corium A and E in inert atmosphere; with Corium E in inert, lightly and strongly oxidizing atmosphere.

c) Simulated fuel rods, molten down in inert and oxidizing atmosphere were examined metallographically, by x ray diffraction and by microprobe analysis.

Essential results

a) The previously in many parts estimated isothermal sections of the ternary systems U-Zr-O, U-Fe-O and Zr-Fe-O were confirmed experimentally. Isothermal sections of these systems were established at 1000°C, 1500°C and 2000°C. In the quaternary system U-Zr-Fe-O the section UO_2-Zr-Fe - respresenting the compositions of a core melt in inert atmosphere - shows at 1500°C the following mean phase fields:

$UO_2+ZrFe_2+L_1$; UO_2+ZrFe_2; $UO_2+ZrFe_2+L_2$; UO_2+L_3.

The phase relations determined in the phase equilibrium studies are in good agreement with the phases observed in the more complex reaction studies.

b) Reaction studies with Corium A in inert atmosphere show that only small amounts of especially Fe vaporize and that in the molten state one metallic (melting region ∿ 1750-2180°C) and one oxidic melt (melting p. ∿ 2240°C) occurs. With increasing steel content one observes at compositions of Corium A/ Corium E 1/1 up to Corium E one metallic melt. In lightly oxidizing atmosphere once more a phase separation occurs in one oxidic (melting p. ∿ 2400°C) and one metallic (melting p. ∿ 1650°C) melt. If all the phases of Corium E are oxidized only one oxidic melt occurs (melting p. ∿ 1900°C).

c) The metallurgical examination of simulated molten rods shows at a section of a fuel rod heated up in inert atmosphere up to 1670°C, that melting can occur below the melting point of Zry at the contact UO_2-Zry. Zr penetrates with preference into UO_2. In oxidizing atmosphere the cladding oxidizes and the melting points are shifted to higher temperatures. Here however the concept of heating must be changed as low melting U,W-oxide phases influence the melting process.

Next steps

a) Phase equilibrium studies in sections of the quaternary system U-Zr-Fe-O: UO_2-α(Zr,O)-Fe; UO_2-ZrO_2-Fe (representing ligthly oxidizing atmosphere), UO_2-ZrO_2-Fe_3O_4 (representing strongly oxidizing atmosphere).

b) Reaction studies of different compositions in the system UO_2-Zry-steel, simulating different atmospheres and including concrete.

c) Metallurgical examination of simulated fuel rods molten down in bundles.

Reference documents

Report	KFK 2101	(1975)	p. 115	(german)
"	KFK 2195	(1975)	p. 338	(german)
"	KFK 2220	(1975)		(german)
"	KFK 2227	(1975)		(german)
KFK-Nachr. 7 (3)		(1975)	p. 50	(german)

Degree of availability

Unrestricted distribution

Classification 2.3

Title 1 Control of molten core debris (1)	Country UK
Title 2	Sponsor UKAEA Organisation Culham Laboratory
Initiated 1972 Completed Status Continuing Last updating	Project leaders

1. General Aim

To have the ability to retain within the containment molten core debris following a core melt-down.

2. Particular objectives

To provide a suitable theoretical model and calculation of the free convective movements of a self heated liquid.

3. Programme

A program has been written to calculate the free convection of a uniformly heated liquid in a channel of rectangular cross-section, the liquid being cooled at the top and bottom surfaces. This program resembles the work of Jahn and Reinecke, and the methods of calculation which they describe are used. Three quantities are calculated at all points of a mesh covering the cross-section, the vorticity, the stream function and the temperature. Equations for the time rate of change of the vorticity and the temperature are used to time step the calculation, and solving Poissons equation gives the stream function when the vorticity is known. The calculation starts with the temperature distribution due to conduction alone and a random vorticity. The calculations made so far show that after a short interval of time convective motion starts and grows exponentially.

4. Next Steps

The model will be corrected and improved following comparison with experiments.

5. Reference documents

Internal documents:

326

Classification 2.3

Title 1 Control of Molten Core Debris (2)	Country UK
Title 2	Sponsor UKAEA Organisation AERE Harwell
Initiated 1972 Completed Status Continuing Last updating	Project leaders R. G. Bellamy

1. General Aim

To have the ability to retain within the containment molten core debris following a core melt-down.

2. Particular Objectives

To provide experimental observations on the free convection of a self heating liquid particularly to enable prediction of heat fluxes at the upper and lower liquid surfaces.

3. Experimental Facilities

Three experimental rigs are involved using weak acids and ohmic heating. The first rig, with a cooled upper surface, has demonstrated that turbulent convection substantially enhances the conductive heat transfer by as much as a factor of 40. The second rig employs both upper and lower cooled surfaces as does the third, under construction, which will enable Rayleigh numbers appropriate to molten UO_2 to be attained. (The power density will be only about 0.3% of that in molten fuel, but the depth of liquid will be much greater).

4. Project Status

A two-dimensional code is employed for flow in a channel of rectangular cross-section and a code dealing with turbulent aspects is under development. The experimental and theoretical work should provide an understanding of the basic heat transfer mechanisms involved.

In the real accident situation there would be other uncertainties, for example, about the boundary conditions - the molten layer may be enclosed in a solid crust, but this crust may be weak and break up. There may also be a danger to support structures from thermal radiation. So far is has been assumed that boiling is avoided. A further possibility which may need considering could be a suspension of small fuel particles, maintained in suspension by turbulence.

Reference Documents.

Internal documents.

3. EXTERNAL INFLUENCES

PROJECT TITLE . Seismic monitoring network	CLASSIFICATION 3.1
SPONSORING COUNTRY : ITALY	ORGANISATION : ENEL
DATE INITIATED : 1973 DATE COMPLETED · -	PROJECT LEADER : F. Capozza

Description :

1. General Aim

Definition of reference earthquake for nuclear power plant sites.

2. Particular Objectives

Collection of data necessary to:

- characterize earthquakes in the different Italian regions;
- obtain a new correlation between acceleration and earthquake intensity in order to utilize the large amount of hystorical data available in Italy.

3. Experimental facilities and program

The seismic network shall consist of 152 monitoring points distributed in the whole Italian territory with the exception of Sardinia.
Each monitoring point shall be equipped with an accelerograph capable of recording a maximum acceleration of 1.00 g and with a threshold of 0.01 g.
The accelerographs are generally located inside electrical substations and installed on concrete columns directly anchored to the foundations.

4. Project status

At present 98 accelerographs have been installed.
Furthermore a computer program has been developed which enables to obtain the seismic spectra (acceleration, velocity and displacement) and their envelopes from instrumental recordings

5. Next steps

Completion of network.

6. Relation to other projects

3.1 (CNEN programs)

7. Reference documents

E. Iaccarino "Possibilità di rilevare un terremoto sul territorio italiano per mezzo di una rete di accelerografi". CNEN RT/Prot. (71) 36

PROJECT TITLE : SEISMOTECTONIC RESEARCHES	CLASSIFICATION 3.1
SPONSORING COUNTRY : ITALY	ORGANISATION : CNEN
DATE INITIATED : January 1975 DATE COMPLETED : In progress	PROJECT LEADER : G. MAGRI

Description :

Seismotectonic researches for nuclear plants site evaluation:

1) Geomorfological and cronostratigraphical studies of marine and subaerial deposits of late Pleistocene to find out:
   - active faults,
   - altimetric changes between land and sea.
2) Correlations between earthquakes (epicentrum, ipocentrum, etc.) and active tectonic dislocations.
3) Studies on earthquakes origins with experimental measurements.

Related projects: 3.1 (other programs)

PROJECT TITLE : Seismic instruments development for site evaluations	CLASSIFICATION 3.1
SPONSORING COUNTRY : ITALY	ORGANISATION : CNEN
DATE INITIATED : May 1974 (present phase) DATE COMPLETED : In progress	PROJECT LEADER : R. CERVELLATI

Description : 1) In the frame of a collaboration CNEN-ENEL, a network of accelerometers has been set up all over Italy with the aim of recording the accelerations of earthquakes. The "time-histories" will be employed in the characterization of the design earthquake. Up to May 1975 about 2/3 of the 150 accelerometers expected to complete the network have been installed.

2) A seismometric equipment has been set up and operated in various sites where the construction of nuclear plants is planned, in order to have a characterization of sites in the seismological field.

The present set up is being enlarged by the addition of seismometric equipments located a few miles apart so to allow the determination of the earthquakes ipocenter.

Reports

Reports obtained in a previous phase of the program are available.
Related projects: 3.1 (other programs)

PROJECT TITLE : Researches on sands liquefaction	CLASSIFICATION 3.1
SPONSORING COUNTRY : ITALY	ORGANISATION : CNEN
DATE INITIATED : May 1975 DATE COMPLETED : in progress	PROJECT LEADER : G. MAGRI, S.POLINARI

Description. Experimental research on correlations between seismic parameters and sands liquefaction.

The program includes: determination of sands density; study of correlations between seismic characteristics and density of sands.

PROJECT TITLE : Study on the possibility of predicting earth quakes by hydrogeochemical methods	CLASSIFICATION 3.1
SPONSORING COUNTRY : ITALY	ORGANISATION : C.N.E.N.
DATE INITIATED : January 1975 DATE COMPLETED : ==	PROJECT LEADER : Mario Dall'Aglio

Description :

It has been demonstraded that various premonitory geochemical phenomena occur before earthquakes. In particular the composition of the deeply circulating waters (e.g. thermal waters) can change some weeks or months before the destructive seismic mouvement.

Some hydrothermal italian systems are regularly checked in order to study the variation of water composition in relation to seismic activity.

PROJECT TITLE : SEISMIC DESIGN FOR NUCLEAR COMPONENTS, SYSTEMS AND STRUCTURES	CLASSIFICATION 3.1
SPONSORING COUNTRY : ITALY	ORGANISATION : AGIP NUCLEARE S.p.A. MILANO - ITALY
DATE INITIATED : May 1975 DATE COMPLETED : October 1975	PROJECT LEADER : Ing. Paolo GRILLO

Description :

Design methods of components, systems and structures for nuclear plants based on their tridimensional analysis with time-history and design spectra.

PROJECT TITLE : STUDIES OF SITE ENGINEERING	~~CLASSIFICATION~~ 3.1 - 3.5
SPONSORING COUNTRY : ITALY	ORGANISATION : CNEN
DATE INITIATED : November 1974 (present phase DATE COMPLETED : in progress	PROJECT LEADER : S. POLINARI

Description : Studies on parameters occurring in the evaluation of sites for nuclear plants.

The program is organized into the following tasks:

- analysis of earthquakes strong motion records
- dynamic response analysis of soil
- analysis of soil-structures interactions
- experimental and theoretical determination of the vibration characteristics of nuclear power plants structures
- development of codes for above analysis
- statistical analysis and studies of exceptional meteorological events
- seismic prevision by random techniques.

Reports

Reports obtained in a previous phase of the program are available.

Related projects: 3.1 (other programs).

	Classification 3.2/11.4.
Title 1 Flystyrt på containment-bygning	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2 Aircraft impact on containment building.	**Project leader** Per Lundsager Scientists:
Initiated (date) March 1975 **Status**: progressing **Completed**: (date) 1976 **Last updating** (date) February 1976	Per Lundsager S. Krenk

1. General aim

1.1. Investigations on structural consequences of an aircraft impact on a containment building.

1.2. Comparison of several FEM-codes available to Risø and evaluation of their potentials in this type of analysis.

2. Particular objectives

2.1. To study the transient response of an idealized axisymmetric containment structure subject to an impact force of the linearized Riera type.

2.2. To study the influence on the displacement and stress results of the element types available in a number of FEM-codes.

3. Experimental facilities and programme

None

4. Project status

4.1. Progress to date

A number of analyses have been carried out using 3 linear and 1 nonlinear code. 3 types of shell elements and 2 types of isoparametric solid elements have been applied.

4.2. Essential results

The stress results seem to a remarkable extend to depend on proper matching of element type and grid layout.

5. Next steps

Evaluation of results.

Internal and external reporting.

6. Relations with other projects

No direct relations.

7. Reference documents

Dynamic Analysis of Aircraft Impact Using the Linear Elastic Finite Element Codes FINEL, SAP and STARDYNE.
Per Lundsager, Steen Krenk.
Risø-M-1817, Aug. 1975.
Presented at the ELCALAP Seminar, Berlin, Sept. 1975.

8. Degree of availability

Project information is freely available

Classification: 3.2	
Title 1 (Original Language): Das Verhalten von Reaktorbauteilen gegen Splittereinwirkung (RS 102-7 - I.3.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Gesell-schaft, EMI
Title 2 (english): Behavior of Reactor Specific Materials and Component Parts at the Impact of Fragments, Splinters and Projectiles of Different Mass and Velocity	Project Leader: Hoffmann
Initiated (Date): 1.9.1973 Status: continuing Completed (Date): 31.12.1976 Last Updating (Date): December 1975	

1. General Aim

The increasing need for nuclear power energy and the actual size of nuclear power plants in twin-blocks of up to 1300 MW, each, forces to investigate and to improve the security means (for people against radioactivity) especially in densely populated regions and near industrial facility concentrations.

Within the scope of the reactor safety program extreme accidents and failures in nuclear power plants are analyzed as to initiation, processing and destruction effects with the aim to improve the actual safety of light water reactors (PWR and BWR)and to optimize - mainly from an economic point of view - the actual safety installations (materials, components, systems).

2. Particular Objectives

In connection with the legally required high security of reactor facilities against accidents from inside and outside (failures of the facility itself, dynamic loadings by aircraft-crashes and fragments released by explosions) and against intended actions by third hand (military interferences, sabotage) the present research program RS 102-7 theoretically and experimentally deals with the essential aspects of impacts projectiles, fragments and splinters on reactorspecific materials and component-parts (especially with concrete slabs, reinforced concrete and steel). The dynamic resistance of slabs - up to their complete destruction at ultimate loads - is investigated for penetration- and perforation-processes with the aim to find and check similarity laws wich scale the physical behavior of model- and prototype structures. This task includes to find the range in which the physical projectile- and target-parameters are valid and to adapt them to the reactor field.

3. Experimental Facility and Program

The destruction effects (perforation, penetration, spallation and scabbing) which are expected to occur at the impact on concrete, reinforced concrete and special steel targets are experimentally investigated in the velocity range from 100 m/sec up to 2.5 km/sec in dependence of projectile mass.

The simulation facility was already mentioned and shown in a picture in the last-year-report (A74). Now the facility is full in action and a series of tests had been done.

The targets to be used are reactor specific concrete slabs and construction steels with different thicknesses and compressive strengths. Furthermore, the composition of the concrete slabs and also their steel-reinforcement were varied.

4. Project Status

The expert in charge became Ill, for more than half a year now and the sickness will endure still for an uncertain time, causing delays in the realization of this program.

The successor, now in charge had to acquaint himself with the topic first. Especially the theoretical works on penetration- and perforation laws for concrete and their limitations of validity as well as the scientific evaluation of the available, experimental facts obtained during the first tests were summarized in a report "Penetrations- und Perforationsgesetze für Beton und ihre Gültigkeitsgrenzen" (Penetration- and Perforation Laws for Concrete and their Validity Limitations). After a revision, this report will be printed now.

4.1. Progress Todate

The required slab sizes and slab fixtures are determined through tests in the impact-chamber with concrete slabs of our production.

During the first tesrphase a total of 54 concrete slabs of grades Bn 250 and 350 with various reinforcements and mesh sieve holes were studied with regard to their resistance behavior against projectile impact. For these studies simple cylinder shaped projectiles with different calibers and length/diameter relations were chosen. In all cases the same projectile material (C 110 W 1) has been used. In the following the series of tests performed are specified:

1. Impact on concrete slabs of grades Bn 250 and 350. The tests parameters were varied as follows:
   Projectile mass m = 20,6 and 41,1 g; velocity v = 150 up to 2000 m/s; diameter D = 1,5 cm; length L / diameter D = 1 and 2. This series of tests was intended to check the validity range of often used penetration formulae while testing the various evaluation methods. For these concrete qualities several tests results are given in the literature.

2. In the following two test series, made with concrete slabs of grade Bn 350, the applicability of model- and scaling laws when using commercial material (aggregates, cement,steel,matting) for the manufacturing of the slabs have been checked. The impact velocities were within the range of 100 up to 900 m/s. In the following table the most important parameters for the full scale slabs and the model slabs and the projectiles are listed:

Table

	Full Scale	Model 1 : 2
Concrete grade (compressability)	Bn 350	Bn 350
Concrete specific gravity	2,3 g/cm^3	2,3 g/cm^3
Mesh sieve holes	B 16	B 8
Steel reinforcement	100x100x8x8	50x50x4x4
	(Handmatte)	(E-strichmatte)
	St 50/55 Rk	St 50/55 Rk

The distances between and the number of the reinforcementmats varies with the model scale and the slab thickness respectively.

Projectile caliber	40 mm	20 mm
Projectile L/D	2	2
Projectile material	C 110 W 1	C 110 W 1

The thicknesses of the slabs range between 10 and 40 cm, depending upon impact velocity and the diameter of the impacting projectile. The tests slabs are square with 70 cm length; this size is necessary to exclude boundary effects on the impact phenomena.

4.2. Essential Results

The tests confirm fully the applicability of model- and scaling laws when the most important model laws are observed.

From the test series the following results were also derived, which are of importance for the intended full scale tests: the hitherto chosen slab sizes are not sufficient anymore when projectile calibers are larger then 50 mm and impact velocities over 600 m/s. Boundary effects falsify the characteristic data such as penetration depth and crater volume. The consistency of the test results with the results of the empirical formulae is for cylinder sphaped projectiles with $L/D \leq 1$ and $L/D \approx 10$ insufficient, even for approximates.

The figures 1 a and 1 b show a typical penetration crater and a scabbing crater on the backside of the test slab. The evaluation of the test slabs regarding penetration depth of the projectiles is shown in figure 2 as well as its summarized results. The diagram shows the normalized penetration depths p/D (D = projectile diameter) over the impact velocity v. The values of the full scale test and from the model tests are situated on the same curve. The deviations in the test result values are caused by accidental conditions by impact, such as touching or penetrating reinforcement steels. The validity of the model- and scaling laws is shown by the good consistency of full scale test and model test.

5. Next Steps

In order to supplement and recheck the test values found, a further number of tests with concrete slabs of the same quality (compressability, reinforcement, aggregate) are necessary. More impact tests on concrete slabs with large L/D projectiles (jet engine shafts) are scheduled. Some tests have shown, that such projectiles represent a special danger for concrete constructions.

Furthermore impact research on special steels (typical steel for reactor construction) is planned.

6. Relations With Other Projects

Do not exist besides some cooperation with other experimental workers of the Institute.

7. Reference Documents

1) P. Rauser — Klassifizierung von Projektilen, Splittern und Fragmenten bei Schadensfällen an Reaktoranlagen

(Classification of Projectiles, Splinters and Fragments at Accidents in Nuclear Power Plants)

EMI-Bericht 2/75

2) H. Langheim, P. Rauser — Penetrations- und Perforationsgesetze für Beton und ihre Gültigkeitsgrenzen

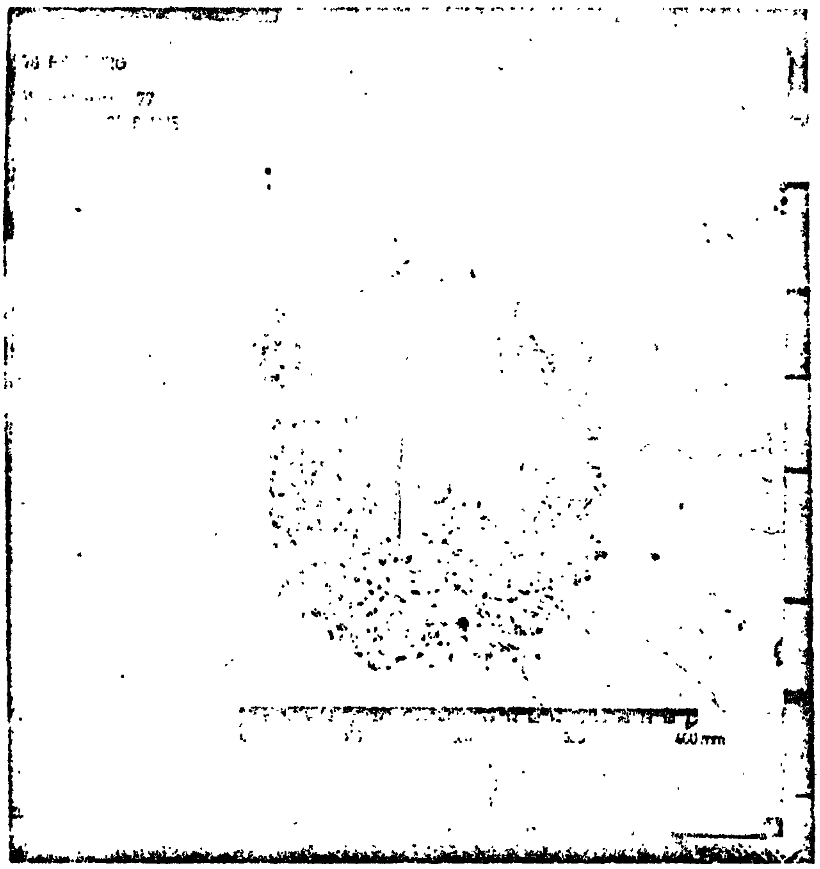
(Penetration- and Perforation-Laws for Concrete and their Validity Limitations)

EMI-Bericht 22/75

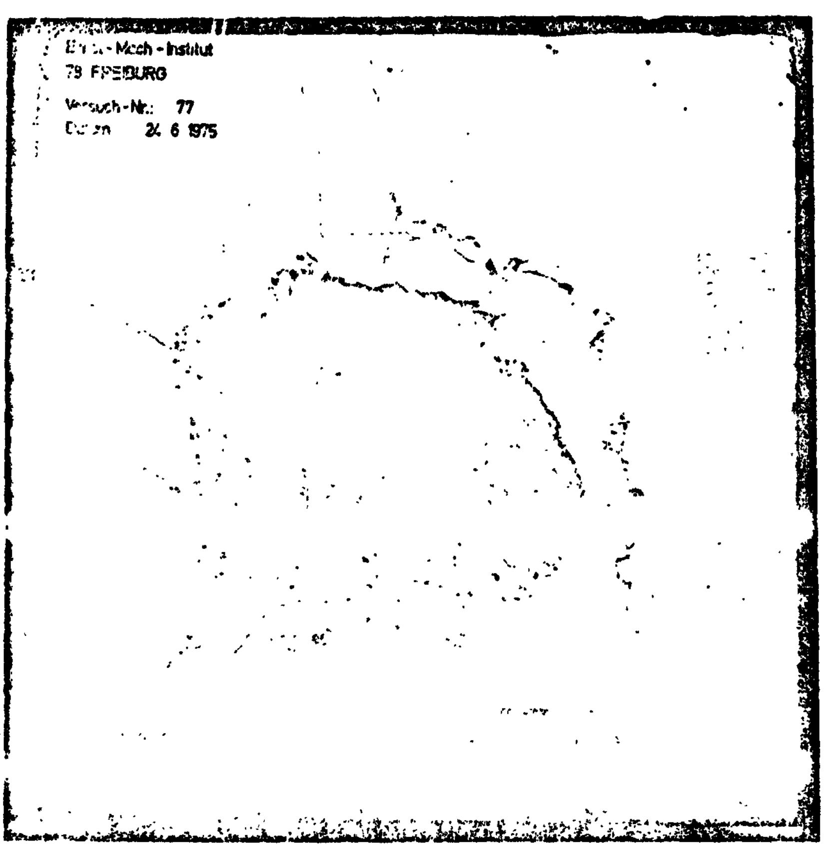
8. Degree of Availability

1) available with permission of BMFT

2) is being printed.



a



b

Fig. 1

Impact crater (a) and scabbing crater (b) in a concrete slab 700x700x200 mm, Bn 350; $\beta_{w\,28} = 408\ kp/cm^2$.
Steel projectile caliber 15 mm, L/D = 2; v = 1700 m/s

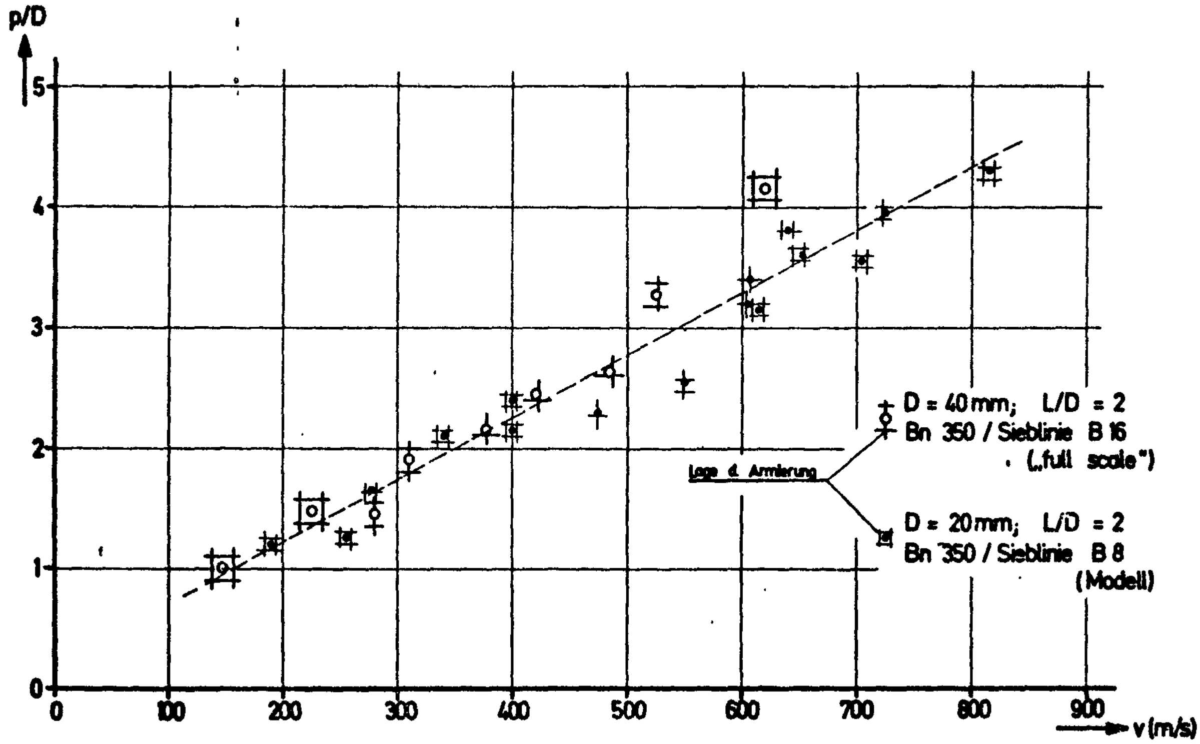


Fig. 2 Penetration depth in concrete Bn 350 as a function of the impact velocity

Classification: 3.2	
Title 1 (Original Language): Grenztragfähigkeit von Stahlbetonplatten bei hohen Belastungsgeschwindigkeiten (z.B. Flugzeugabsturz) und: Untersuchungen der Widerstandsfähigkeit von Betonstrukturen gegen Flugzeugabsturz (RS 165 und 149 - I.3.2., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** HOCHTIEF, Ffm. BWB, Koblenz
Title 2 (english): Ultimate Bearing Capacity of Reinforced Concrete Plates under Time-Dependent Loads (i.e. Aircraft Crash) and: Investigations of the Resistance of Concrete Structures to Crashing Aircrafts	**Project Leader:** Riech (coordination) Weymar
Initiated (Date): July 1, 1975 and Oct. 10, 1974 **Status:** continuing **Completed (Date):** June 30, 1978 and April 30, 1977 **Last Updating (Date):** December 1975	

1. General aim

In connection with research works performed on the field of reactor safety, one of the subjects covering "External Events" gives special emphasis to the investigation of the behaviour of the outer containment of nuclear power plants under aircraft crash loading. For the loading case "Aircraft Crash" it is demanded that no failure of safety components will occur and no radioactive substances may escape. This requirement will be met by an appropriate dimensioning of the outer containment of the structures surrounding the nuclear components so that a crashing aircraft cannot penetrate the outer walls.

Since the loading case Aircraft Crash entails high load peak values within short periods, the knowledge of the kinetic ultimate bearing loads is required for a safe and economic design of the plates and shells being used for nuclear power plants, i.e. the best utilization of all safety reserves.

Both combined research projects RS 165 and RS 149, are dedicated to the theoretical and experimental investigation of the essential problems of the loading case Aircraft Crash :

1. Investigation of impact load/time characteristics during the impact of deformable missiles;
2. Investigation of the kinetic bearing behaviour of reinforced concrete plates.

2. Particular objectives

The preceding and accompanying theoretical investigations performed in the scope of research program RS 165 aim at recording the following items:

- the impact of deformable missiles
- the physically nonlinear material behaviour of reinforced concrete structures under time-dependent loading
- the influence of finite deformations
- the three-dimensional problem in the area of load introduction.

The project RS 149 comprises:

- provision, installation and testing of the missile accelerator
- construction of a target abutment
- production of approximately 24 model missiles and the same number of reinforced concrete test plates
- procurement and installation of the measuring instruments.

3. Experimental facilities and program

For the following reasons the scale of the experiments was chosen as large as possible:

Small scale structures adjusted to laboratory conditions would cause difficulties on the following fields:

- selection of the granular size of the aggregates
- measures securing the composite of steel and concrete
- registration of the kinetic stress distribution (shock waves) within the structure (the velocity of the shock waves is independent of the geometric scale)
- installation of stirrups in the reinforcement and evaluation of their effectiveness.

The mentioned difficulties would require special activities and compromise settlements and would entail additional falsifying values or such effects which cannot be taken into account by a realistic theoretical treatment. The interpretation of the results would become very difficult or even impossible, especially with regard to the separation of the essential influence factors.

The realization of large scale experiments was facilitated by the cooperation of the Federal Ministry for Research and Technology and the Federal Ministry of Defense.

At the site of the Bundeswehr-Erprobungsstelle 91 (BWB/E.St.91) at Meppen a gas operated accelerator will be built by which missiles of a maximum diameter of 600 mm, a length of up to 7 m and a mass of up to 1000 kg may be accelerated to a maximum speed of 300 m/s.

Specially constructed projectils are used as model missiles, consisting of concentric tubes of varying length. The deformation behaviour of those tubes grants the impact load/time characteristics similar to those which may be expected in an aircraft crash. Quadratic reinforced concrete walls of 6 m edge length and up to 1,20 m thickness are used as targets. An abutment will be constructed in order to fix the target plates.

4. Project status

4.1 Progress to date

4.1.1 RS 165 (Hochtief AG, Abt. Kerntechnischer Ingenieurbau)

- Layout of the target abutment and the measuring instrumentation:
  The technical layout of the target abutment for the experimentary plant has been completed. The design drawings were delivered to BWB/E.St.91.
  Under consideration of the available measuring instruments of E.St.91, a measuring program was established by Hochtief in cooperation with E.St.91. This schedule informs about the measuring sizes, -methods, -installations and those instruments which still have to be procured.
- Preparing theoretical works:
  The investigations and calculations of the following individual items were continued:
  1. Dimensioning of the reinforced concrete plates

2. Physically nonlinear material behaviour of reinforced concrete structures under time-dependent loading.

With regard to the calculation of physically nonlinear problems the following works have been performed:
Provision and testing of triangular elements, which permit a sufficiently exact approximation of the displacements and stress resultants by use of the finite-element-method.
Procedures for the calculation of physically nonlinear time-dependant problems have been developed and partially tested. The following methods were alternatively used for considering the physically nonlinear behaviour:

a. Use of nonlinear moment-curvature relationships
b. integration of the stress distribution over the thickness of the reinforced concrete plate. A nonlinear stress-strain relationship is considered for the concrete as well as for the reinforced steel.

Different stiffnesses in the area of loading and unloading are taken into account.

4.1.2 RS 149 (BWB/E.St.91)

- The construction of the experimental plant on the area of the test site of E.St.91 has been started.
- The construction of the track system for the acceleration machine and the construction of the subsidiary building were finished by the end of 1975.
- The calculation and construction of the acceleration machine were completed by Messrs. Schwartzkopff, Bonn.
- The mounting of the bogie for the accelerator has been started.
- A part of the nonpresent measuring instruments, which are required for the tests, has been procured or ordered.

4.2 Essential results

First results cannot be expected prior to the completion of the total experimental plant.

5. Next steps

5.1 RS 165 (Hochtief AG)

- The measuring program will be presented to an independent expert.
- Four reinforced concrete test plates for the first series of measurements will be dimensioned and designed.
- The calculation and construction of the model missiles for the first four tests have to be renewed since calculations of Messrs. Schwartzkopff, Bonn, gave higher pressures in the expansion tube of the accelerator as originally assumed for the dimensioning of the model missiles.
- The theoretical investigations will be continued with the aim of improving the developed methods of calculation.

5.2 RS 149 (BWB/E.St.91)

- The work of Messrs. Schwartzkopff, Bonn, will be continued according to the time schedule and will be finished by the end of June 1976.
- The start of constructing the target abutment, the measuring and observation dug-out, the retaining wall and the working area in front of the target abutment will be effected in February 1976.
- The experiments for checking the mathematical models being used for the calculation of the impact load/time characteristics of deformable missiles will be performed during the last quarter of 1976.

TNO - IWECO	CLASSIFICATION: 3.2 / 3.3 7.1.
TITLE: Responsieberekeningen voor reactorgebouw	COUNTRY: NETHERLANDS SPONSOR: Ministry of Social Affairs ORGANIZATION: TNO-IWECO
TITLE (ENGLISH LANGUAGE): Dynamic Response of Reactor structures (building and containment)	PROJECTLEADER: Meyers
INITIATED: June 1974 — COMPLETED: end 1976 STATUS: progressing — LAST UPDATING: February 1976	SCIENTISTS: Van Beek Geertsema

1. General aim

   Development and application of calculational tools for the evaluation of structural response of reactorbuildings under dynamic loading conditions.

2. Particular objectives

   The calculations are directed to the evaluation of the effects of
   (1) pressure/blast waves of gasexplosion in the vicinity of the site
   (2) impact of strking aircraft on the reactor (containment) buildings.

3. Expermental facilities and programme: none

4. Project status

   A model was developed to describe the non-linear behaviour of reinforced concrete with a finite-elements technique. A special subroutine for the MARC-computercode describes this model. Response-calculations have been performed on a simple beam-model. Calculational results show a good comparison with experiments.

5. Next steps

   Application of the computational method to a typical reactorbuilding (containment).

6. Relation with other projects: -

7. Reference documents

   "Beschrijving van het niet-lineaire gedrag van gewapend beton"
   ("Description of the non-linear behaviour of reinforced concrete")
   by ir. H. Geertsema, June 1975, TNO-IWECO-reportnr. 11261/1.

8. Degree of availability

   Through Ministry of Social Affairs.

9. Budget — approx. Hfl. 270.000

10. Additional information — total duration approx. 2 years.

Classification: 3.3	
Title 1 (Original Language): Entstehung chemischer Explosionen und deren Wirkung auf sicherheitstechnisch wichtige Reaktorkomponenten (RS 102-6 - I.3.2., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** Fraunhofer-Gesellschaft, ICT, EMI
Title 2 (english): Formation of Chemical Explosions and their Effects upon Reactor Components with Important Safety Functions	**Project Leader:** Dr. Pförtner Hoffmann
Initiated (Date): 1.9.1973 — **Completed (Date):** 31.8.1976 **Status:** continuing — **Last Updating (Date):** December 1975	

1. General Aim

Experiments with unconfined ignitable clouds of explosible gases are performed in order to supply information on the hazards to nuclear installations from gas explosions resulting from accidental gas leak or spill in the course of production, storage and transport of explosible gases near nuclear power plants.

2. Particular Objectives

(a) Studies of gas dispersion after a leak or spill of explosible gas from liquid-gas containers or pipes and determination of the amount of ignitable mixture at any time after the spill.

(b) Studies concerning the explosion and detonation limits of explosible gases. Determination of flame propagation rates in unconfined vapor clouds with linear and spherical geometry, investigating the possibility of a deflagration-to-detonation-transition. Determination of the propagation functions of the pressure waves and shock waves respectively generated by an explosion and assessment of their impact on a reactor facility.

3. Experimental Facilities and Program

A weather-hall of 6 x 12 m^2 ground area is used for the gas dispersion tests. Liquid gas will be spilled onto a heatable platform of 3.7 m diameter placed at one end of the hall. The platform allows spills either on water, sand or concrete. The gas evaporation rate or source strength is measured by weighing the platform with a force transducer. The extension of the cold gas cloud is observed by filming the foggy region in scattered light and measured by several traverses with 8 suction probes placed at a plane at varying distances from the spill point. Analysis is performed by infrared-light-absorption measuring devices. At the front side of the hall near the platform there are facilities to produce a uniform stream of air with wind velocities up to 2.5 m/s. The exit cross section of the blower section is 2.5m high by 4 m broad.

Three gases will be spilled in amounts of 2, 5 and 10 kg each. At first liquid propane ($20^{o}C/8.5$ bar) is released in a way similar to the case when a tank is hurt getting a big hole in the bottom. Next liquid ethylene ($-104^{o}C$) and liquid methane ($-161^{o}C$) will be released from a dewar at atmospheric pressure as a "batch"-spill.

The extension of the flammable region over the first minutes and the amount of gas mixture within the flammability limits will be calculated and compared with computational predictions basing on a Pasquill-diffusion-model for an area source growing with time during evaporation.

4. Project Status

4.1 Progress to Date

(a) The blower with diffuser and the platform are being mounted and will soon be ready for service. The experimental set-up has been prepared. Realistic evaporation rates for cryogenic hydrocarbon gases on different ground materials are being collected from the literature in order to feed the computational model.

(b) Based upon several experiments for testing the measuring technique a series of 30 balloon experiments was carried out to determine the propagation functions of spherical shock waves in air in the case of detonating ethylene-air mixtures. The gas mixtures were ignited by a microsecond igniter and 50 g tetryl. At 15 different locations at distances between 5 and 50 m from the center of detonation pressure transducers were installed for the registration of the pressure-time dependence of the blast waves. The pressure transducers were mounted in such a way that the pressure sensitive area of the transducers and the ground surface were complanar.

The following parameters were varied:

- volume of the balloons 1.5 m^3, 15 m^3 (spherical) and
  1/2 x 1.5 m^3, 1/2 x 15 m^3 (hemispherical)
- position of the balloons
  on the ground (measurement of free propagation)
  above the ground (investigation of the reflexion of shock waves at the ground)
- location of ignition
  centrally, eccentrically (at the balloon surface)
- ethylene concentration
  5, 5.5, 6.53 (stoichiometric) and 10 Vol.%

In some experiments high speed cameras were used for the optical registration of the detonative event within the balloons.

In a second series of tests the amount of explosive was determined which is sufficient for the initiation of a detonation in a stoichiometric ethylene-air mixture ("critical" amount of explosive). The experiments were carried out with 1.5 m^3 balloons as well as with 15 m^3 balloons to see if there is a dependence of the critical amount of explosive on the gas volume. Additionally the influence of the location of the ignition was examined (centrally, eccentrically). Pressure transducers within and outside the balloons as well as a high speed camera made it possible to decide whether there was a detonation or not.

In another series of tests stoichiometric ethylene-air mixtures in thin walled PE tubes (length 27 m, diameter 0.5 and 0.8 m) were ignited by exploding wire or pyrotechnical bridge-wire-igniter at one end of the tube to collect informations about flame propagation rates. In some experiments the tube diameter was reduced over a distance of 2.5 m to produce eventually an acceleration of the flames. Additionally to the stoichiometric ethylene-air mixtures with an $O_2 : N_2$ ratio of 20:80 mixtures with a ratio of 40:60 were investigated. The flame propagation rates were measured by means of 10 phototransistors, which were installed outside the tube at intervals of 2.5m.

The pressure waves produced within the tube were registrated with 10 pressure transducers. Additionally most of the experiments were filmed with a high speed camera.

4.2 Essential Results

(a) No results yet

(b) The data of the balloon experiments were completely analysed and documented in a report (see point 7.), the essential contents of which are the propagation functions of shock waves in air, i.e. peak overpressure Δp, duration of the positive pressure phase t^+ and impulse I^+ as a function of the distance from the detonation center considering the appropriate scaling laws. As to the peak overpressures there were no recognizable differences between centrally and eccentrically ignited balloons. It can be stated however that t^+ as well as I^+ show an asymmetrical behavior in the case of eccentrical ignition. In contrast to the gas mixtures with an ethylene concentration of $\geqq$ 5.5 % mixtures with 5 % ethylene did not detonate; this means that the lower detonation limit of ethylene-air mixtures is between 5 and 5.5 %, at least for the selected ignition energy.

In the case of 5.5 and 10 % mixtures the peak overpressures were lower than for stoichiometric mixtures; this may be due to the lower content of explosion energy in those mixtures.

The critical amount of explosive as defined above lies between 6 and 8 g of PETN, i.e. ignition with 8 g resulted in a detonation, ignition with 6 g did not, independent of balloon size or location of ignition.

A preliminary analysis of the data obtained from the experiments with PE tubes shows that the flame propagation rates are between 5 and 8 m/s for stoichiometric ethylene-air mixtures. For those mixtures with a $O_2 : N_2$ ratio of 40:60 the rates are higher by about a factor of 10. In those regions of the tube where the tube diameter has been reduced a temporary acceleration of the flame front could by observed.

5. Next Steps

(a) Measurement of wind velocity distribution and turbulence in the weather-hall. Propane-gas-spills.

(b) The experiments with PE tubes will be continued with other deflagrative ignition sources, e.g. with jet flames to get additional insight into the process of flame acceleration. Moreover, experiments with larger gasvolumes ($> 100\ m^3$) and deflagrative ignition are planned in order to decide whether the results obtained till now are also valid for larger amounts of gas.

6. Relation with other Projects

RS 102-09 Diffraction of Shock Waves on Reactor Buildings, Fraunhofer-Gesellschaft, Ernst-Mach-Institut, 1973-1976

7. Reference Documents

-RS-report E 3/75 (Ernst-Mach-Institut, Freiburg)
Experiments with explosible gas/air mixtures in a detonation chamber (RS 102-06-1)

-RS-report RS 102-06-3
Propagation functions of spherical shock waves in air in the case of detonating ethylene-air mixtures.

Classification: 3.3	
Title 1 (Original Language): Beugung von Druckwellen um Reaktorgebäude (RS 1C2-9 - I.3.2., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** Fraunhofer-Gesellschaft, EMI
Title 2 (english): Diffraction of Shock Waves on Reactor Buildings	**Project Leader:** Hoffmann
Initiated (Date): 10.1973 — **Completed (Date):** 31.12.1976	
Status: continuing — **Last Updating (Date):** December 1975	

1. General Aim

For the security of nuclear power plants also dangers coming from outside have to be taken into account. One of them are the dynamic loads of the reactor containment caused by pressure waves. They are formed in all cases where energy is transferred by air, for example when a chemical or nuclear explosion takes place.

Therefore it is the aim of this project to find out the pressure-time-history of the pressure waves which are reflected and diffracted, when hitting a nuclear power plant by means of model experiments in the shock tube.

2. Particular Objective

Because there exist no data on pressure loads and their pressure-time-history running around a combination of buildings the objectives of this program are

a) to measure precisely the pressure loads of the single reactor buildings and to correct, if necessary, the data given in the standard literature

b) to get information on the shielding-, focusing- and reflection effects occurring in a reactor plant hit by a shock wave

c) to try to elaborate some general rules for the design of reactor plants in order to protect them optimally against shock waves.

3. Experimental Facilities and Program

The two kinds of experiments

a) shadowgraph experiments on small two-dimensional structures

b) measurements of pressure-time-history at defined positions of the power plant are done in a shock tube. Details of this shock tube and the method of getting results are given in the annual report A 74.

4. Project Status

4.1. Progress to date

A literature report elaborated at the beginning of the year has not yielded any new aspects. So the experimental program has been started at the following state of knowledge:

1) There is no information on the pressure loads of combined structures

2) Informations on the pressure loads of single buildings (quader, sphere, cylinder) can be found in some old manuals (Draft Engineering Manual, Air Force Design Manual). Particularly at the rear side of the structures they are not sufficient.

The shadowgraph pictures provided in the experimental program habe been performed completely. With their help the complement of the models for the large shock tube with pressure transducers has been carried out and the most important pressure-time-histories have been interpreted. Of particular interest are measurements concerning the time which the MACHstem takes ro overrun a semicylinder and three three-dimensional models (reactor itself, reactor with a longer cylindrical part, reactor dome).

After many technical preparations the first part of the pressure measurements (reactor and turbine buildings as single buildings and in combination) has been performed. Shock strengths were 1,2 and 1,4 , corresponding to a peak reflection overpressure of 0,45 and 1,0 bar. From the pressure-time histories given by the pressure transducers at the reactor the maximum pressures were evaluated and plotted against various geometrical parameters. These histories clearly show the influence of the turbine building on the load of the reactor. The pressure-time-histories with the highest pressures were completely interpreted by the shadowgraph pictures.

4.2. Essential Results

From the shadowgraph pictures about the semicylinder and about the three-dimensional models it could be concluded:

1) In the rear part of the structure the overrunning time of the MACHstem around the structures clearly deviates from the statements of the Engineering Manual.

2) In the pressure region being of interest here there is only a slight dependence of the overrunning time normalized by D/U on the shock strength and the cylinder length (D = diameter of the reactor, U = shock velocity).

3) The overrunning time over the semicylinder and over the dome is nearly the same, so the focussing point on the rear side is expected being positioned on the boundary between dome and cylindrical part of the reactor.

The most important results of the pressure measurements are:

1) There ist no pressure increase at the transition point from regular to MACH reflection with respect to zero-degree incidence.

2) At some configurations of the complex reactor and added turbine building the maximum pressure exceeds the peak reflection overpressure by far. The highest pressure was 1,54 times the peak reflection overpressure.

   The pressure-time-history where this high pressure occurs is shown in Fig. 1. It is compared to the corresponding one which is obtained at the single reactor building.

5. Next Steps

It is planned to evaluate the pressure-time-histories also with respect to their course, not only to their maximum pressure. But it is feared that there many difficulties will overcome because the frequency behavior of the reactor is here of great importance.

To get more information about the pressure field at the critical arrangements which are known now, interferometer experiments will be done.

Furthermore, the second part of the pressure measurement program with the building complexes reactor/auxiliary supply/swichgear and reactor/turbine building/auxiliary supply/swichgear is performed.

In 1976 the research project should be finished by a final report.

6. Relations with other Projects

Up to date there no data from other projects influencing this program. Later on the results from RS 102-6 may yield valuable information on the shock strengths to be expected.

7. Reference Documents

a) Literaturrecherche II

b) Statusbericht RS 102-9

8. Degree of Availability

Both reports mentioned in 7. exist as a manuscript.

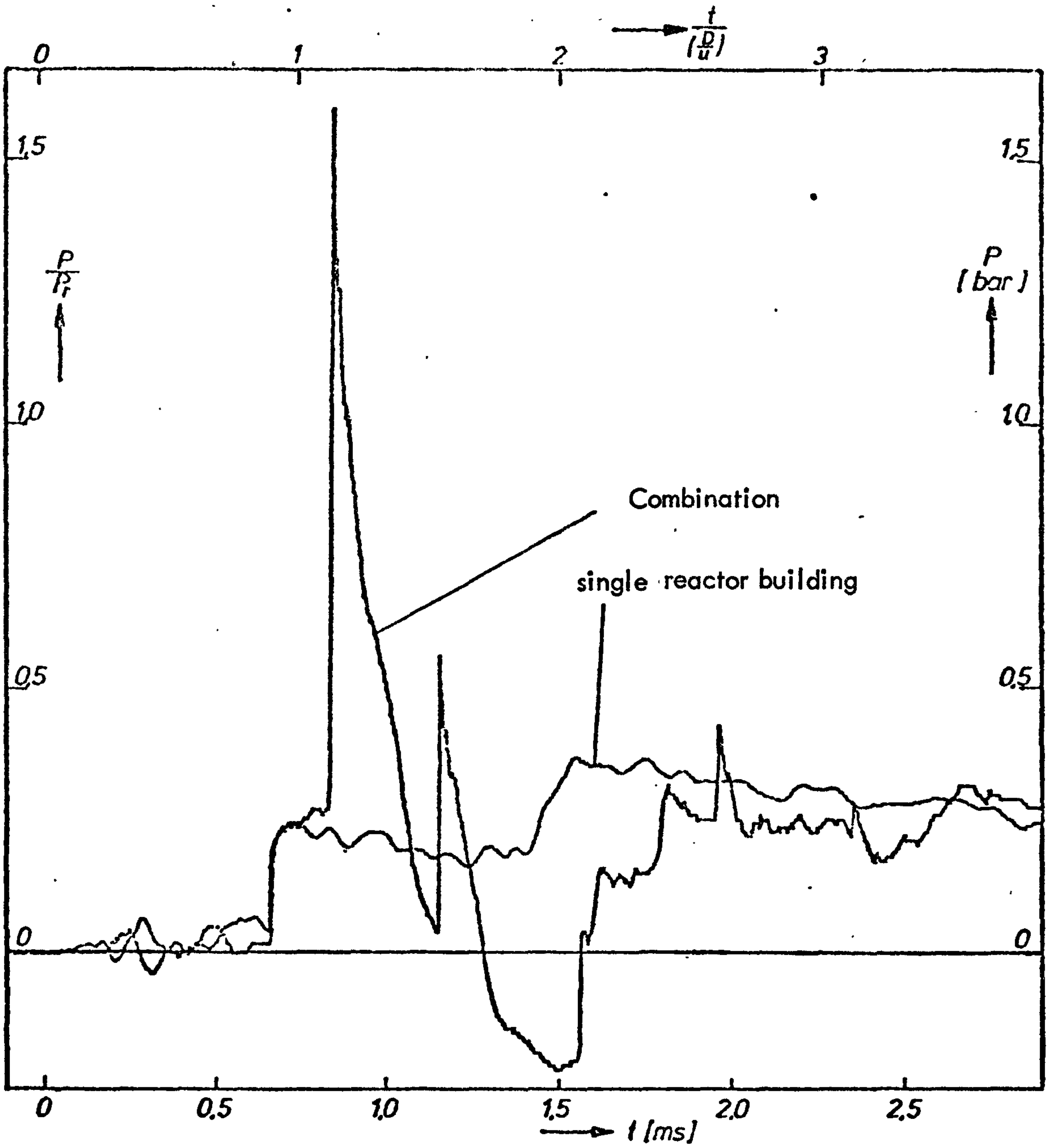


Fig. 1

p = local overpressure

p_r = peak reflection overpressure at the front of the reactor building = 1 bar

t = time; t = 0 - arrival of the wave at the front of the reactor building

D = reactor diameter = 30 (cm) $\frac{D}{u}$ = 0,76 (ms)

u = shock velocity

Pressure-time-history on the reactor building at $\Phi = 135^{\circ}$

Comparison single building-combination

Classification: 3.3	
Title 1 (Original Language): DICE THROW-Vorstudie (RS 173 - I.3.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: SDK, Lörrach
Title 2 (english): DICE THROW-Feasibility Study	Project Leader: H. Hofmann
Initiated (Date): 30.7.1975 Status: finished Completed (Date): 31.12.1975 Last Updating (Date): December 1975	

1. General Aim

In autumn 1976 reinforced concrete buildings shall be tested under pressure waves originating from explosions. These tests will take place at the test grounds "White Sand", New Mexico, USA, under the project name DICE THROW (4-nations project). An explosion of 500 tons TNT is planned, which will mainly serve military purposes.

The Bundesminister für Forschung und Technologie (BMFT = Federal Minister for Research and Technology) considered to take part in this project, mainly being interested in detailed analysis concerning the

- formation and propagation of pressure waves in atmosphere and ground
- reflexion and flow of pressure waves around the buildings of nuclear power plants
- maximum bearing capacity, safety against tumbling and vibration of structures induced by pressure waves

with respect to large structures. Another point he is interested in, is the quantification of the respective safety margins.

A review of design methods worked out during former nuclear weapon tests as well as a control of the results by means of advanced computational and measuring methods were of further interest. As a basis for the decision on whether a participation of BMFT is useful, pertinent and organizational conditions had to be clarified by this feasibility study under the aspect of nuclear safety considerations.

2. Particular Objectives

-

3.1 Experimental Facilities

Not relevant.

3.2 Research Program

-

4. Project Status

4.1 Progress to Date

Since a participation of BMFT with own objectives requires an adaption to the general test philosophy, first it was necessary to work out a concept and the possibilities given by the test arrangement. As a basis, the technical documents ot the Bundesminister für Verteidigung (BMVg = Federal Minister of Defence) and the Infrastrukturstab der Bundeswehr (InfrastrStBW = Infrastructure Staff of the German Federal Armed Forces) were used. BMVg also takes part with own objectives.

Furthermore, the possibilities and limits of an additional participation could be discussed in personal conversations with the competent representatives of the designing and performing institutions - Defense Nuclear Agency (DNA) and Waterways Experiment Station (WES). These discussions took place at the test grounds "White Sand" and in Albuquerque, New Mexico, on the occasion of a DICE THROW pilot test in August 1975 (explosion of 100 tons TNT).

To simplify matters and for the sake of clarity, first essential results have been summarized in the draft version of two research proposals - "Experimental Investigations and Preliminary Computational Analysis", which have been submitted for judgement to the BMFT. The proposals include the design of a characteristic containment building, the corresponding pre-dimensioning and a distribution of mass. The final report is available, too.

4.2. Essential Results

4.2.1. Possibilities of the Large Scale Tests

The high energy release resulting from an explosion of 500 tons TNT allows to conduct tests with large buildings of nuclear power plants, e.g. containments, under reduced scale problems and at different excitation levels - low, medium and high - depending on the distance between the test building and the explosion center - about 100 to 300 m.

By the choice of an excitation level, tests in the linear and nonlinear region of structural behaviour are possible. Such large-scale tests have never been performed up to now and would be a valuable extension of present practice. For safety reasons, however, the performance of those tests will scarcely be possible in Germany.

4.2.2. Technical Performance

The large-scale test DICE THROW especially will present

- a better information on the structural behaviour of large buildings
- a comparison between modern computational and experimentally results, particularly a check of the extrapolation of computational methods and theoretical fundamentals to large structures and thus
- an extension of the range of knowledge required for nuclear reactor safety considerations with respect to the large nuclear power plant structures.

On account of a participation of BMFT, it is possible to build two additional characteristic large scale containments and to test them under the influence of pressure waves resulting from explosion.

Text to be continued on next page.

Furthermore, the aboveground and underground cubic buildings erected by BMVg may be used for the objective of BMFT (e.g. for cable channels etc.). DNA may provide up to 300 channels for the instrumentation. As a completion, free-field records of other participants will be available in exchange for own data.

In addition material properties necessary for preliminary analyses and pre-dimensioning of the structures will be determined and made available on the test grounds. Results and evaluations of the pilot tests may be obtained under data exchange agreements.

From the technical point of view a participation in the large-scale test DICE THROW is recommended.

The final recommendation still depends on the clarification of several conditions. These include e.g. binding cost estimates for the structures to be ordered in the USA and further details concerning the efficiency of the measuring technique.

5. Next Steps

The project "DICE THROW-Feasibility Study" is completed.

6. Relation to Other Projects

At the time there are no other projects in execution concerning the behaviour of nuclear power plant structures under the influence of explosions with respect to tests of such a large scale.

7. Reference Documents

- Abschlussbericht RS 173 "DICE THROW-Vorstudie zur analytischen und experimentellen Erfassung von Reaktorbauwerken unter simulierter (Gas-) Explosionsbelastung im Grossmaßstab", Oktober 1975
- Anträge an den BMFT vom 3. September 1975
  - DICE THROW - Experiment   BMVg - Rü III 8 -
  - DICE THROW - Analytik   SDK

- Reise- und Besprechungsbericht vom 26. August 1975
  - Informationsgespräche DICE THROW, Albuquerque, N.M., USA
  - Vorversuche zu DICE THROW (100 t TNT) auf dem Testgelände "White Sand"

8. Degree of Availability

-

PROJECT TITLE : Preliminary Research on Tornado Effécts on Nuclear Plants	CLASSIFICATION 3.5
SPONSORING COUNTRY : ITALY	ORGANISATION : UNIVERSITY OF PISA
DATE INITIATED : 1/6/1974 DATE COMPLETED : 1st Report 15/10/1974	PROJECT LEADER : M. MARINI

(in progress)

Description :

The program is studying the effects of tornadoes on main physical nuclear plants on the basis of data collected in the United States. A parallel research on the theoretical evaluation of a tornado characteristics is in progress. The direct and induced effects of tornadoes on structures has been examinated, with particular reference to the nuclear plants. The behaviour and the effects of some tornadoes in Italy in 1974, were studied making an attempt for a preliminary analysis of the experimental data. The problem of a classification of the tornado intensity on the basis of the provoked damages will be examinated to correlate, if possible, the various physical parameters.

The program has been sponsored by C.N.E.N.

Title 1 (original language)	COUNTRY : FRANCE
Etude analytique des transitoires accidentels des réacteurs P.W.R.	SPONSOR : CEA
	ORGANIZATION
	CEA
Title 2 (english)	Project leader
Analytical study of accidental transients for P.W.R	CEA/DSN/SETS J.P.MERLE Scientists :
Initiated (date) 1973 Status : progressing	Completed : (date) Last updating (date) Janvier 1975

1. **But général**

   Etude de la réponse des principaux paramètres de fonctionnement d'un réacteur à une perturbation d'origine accidentelle.

2. **Objectifs particuliers**

   Etudes des principaux transitoires accidentels qui interviennent dans l'analyse de sûreté des réacteur de puissance.

3. **Installations expérimentales**

   ———

4. **Etat du projet**

   Code de calcul SIRENE (P.W.R) en cours de tests (comparaison avec les résultats du constructeur).

5. **Prochaines étapes**

   Extension des possibilités de programme (quant aux types de perturbation pouvant être traitées).

4. <u>POWER TRANSIENTS</u>

CLASSIFICATION
4

TITLE 1	CODES DE CALCULS DES TRANSITOIRES ACCIDENTELS	COUNTRY FRANCE SPONSOR E.D.F/SEPTEN ORGANIZATION E.D.F.
TITLE 2	CALCULATION CODES FOR ANTICIPATED TRANSIENTS	Project Leader E.D.F./SEPTEN/T
Initiated 1973	Completed 1975	Scientists M. LARMINAUX
Status	Codes opérationnels. Last updating: 20.01.75	

I - GENERAL AIM

Etude des conséquences de régimes transitoires accidentels de dimensionnement.

II - PARTICULAR OBJECTIVES

- Code BABEL: études d'accidents en régime symétrique. Le code comporte une représentation simplifiée de la chaudière nucléaire et de la partie secondaire, un calcul de pression et de niveau est fait dans le pressuriseur et dans la partie secondaire du générateur de vapeur.
- Code ETINCEL : études d'accidents dissymétriques. Le code (modèle 2 boucles) comporte la représentation de deux canaux dans le coeur et peut prendre en compte des mélanges plus ou moins importants dans les zones d'entrée et de sortie. Il permet l'étude d'accidents conduisant à des inversions de débits primaires.

III - EXPERIMENTAL FACILITIES AND PROGRAMME

Néant.

...

IV - PROJECT STATUS

4.1 - Progress to date

Codes élaborés,

4.2 - Essential Results

Etudes des régimes transitoires pour les rapports de sûreté,

V - NEXT STEPS

VI - RELATION WITH OTHER PROJECTS

Néant,

Mélange dans la cuve d'un PWR des écoulements provenant des diverses boucles,

VII - REFERENCE DOCUMENTS

VIII - DEGREE OF AVAILABILITY

La communication des codes ou leur utilisation pour des études appliquées doivent faire l'objet de contrats cas par cas.

Classification: 4	
Title 1 (Original Language): Untersuchung von Betriebstransienten bei Versagen des Schnellabschaltsystems (ATWS-Studie) (RS 153 - I.1.8., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Investigation of Operation Transients during Failure of the Scram System (ATWS-Study)	Project Leader: G. Frei
Initiated (Date): 1. 11. 1974 — Completed (Date): 31. 7. 1976	
Status: Continuing — Last Updating (Date): 31. 12. 1975	

General Aim

The aim of this project is to determine the dynamic behaviour of PWR and BWR plants during different anticipated transients with failure of the shut down system (ATWS = Anticipated Transients Without Scram). The investigations shall show, whether limiting-values of the fuel elements, the core and the loop components are exceeded.

Particular Objectives

The design conditions for a second scram system can be specified on the basis of the results of this investigation if necessary. For this purpose some pre-investigations shall be executed.

Project Status/Progress to Date

PWR:

The consequences of a failure of the shut down system were investigated on the transient behaviour, corresponding to the RSK-guide lines. The analysis were conducted for the power plant Grafenrheinfeld and is valid for all 1300 MW-reactors of the present generation. The analysis started from normal operation of the reactor; it was assumed that only the shut down system failed and all the other systems were operating well.

The accident analysis was concentrated on the new core with high boron content (about 1100 ppm at full power), because the density effect for power reduction is rather low in this case. Besides this, the Dopplereffect has its maximum value for a now loaded core. Other reactivity effects were not considered in this study.

The case of feedwater break down was investigated, considering a dead time of the steam generator safety valves of 50 and 500 msec respectively. After this the opening of a life steam line safety valve was investigated at the end of a power cycle (strong negative temperature backfitting).

To verify the computer code LOOP 7, some commissioning tests of Biblis A under full power were recalculated:

- Reactor-Scram
- Loss of auxiliary power
- Load rejection to zero power with rod injection

The most important process data were compared with calculated values. Some boundary conditions however, e. g. variable speed of the coolant pumps, time behaviour of the reactivity after control rod injection had to be adepted to the test series. Some initial values, e. g. water content of the pressurizer and coolant pressure, had to be compared and adepted to values at the begin of the test.

Some corrections were made with respect to the time behaviour of the instrumentation. The influence of the undercooling water in the pressurizer was investigated, which has to be considered during the auxiliary power case or after scram.

BWR:

The behaviour of a typical BWR power plant was investigated for the KKB and KKI power plants, corresponding to the RSK guide lines (8 transient cases, after failure of the shut down system). As protective actions the controlled slow down of the pumps to minimal (rate: 10 °C/sec), speed, and the collective electromechanical insertion of the control rods (insertion time: 120 seconds) were considered.

The simulation code DRAMP was improved in some details.

Project Status/Essential Results

PWR:

The reslut of the ATWS-study was that the pressure of the primary system does not exceed ~10 % of the design pressure value in all cases; the transients were limited by voiding of the core. In spite of film boiling in some cases, the maximum fuel rod surface temperature will be below 650 °C. It can be stuted, that a second shut down system is not necessary.

The recalculation of the data from the commissioning tests of Biblis A with the code LOOP 7 used for the ATWS-calculations showed rather good agreement between experimental and theoretical values.

BWR:

The ATWS-investigations show, that in most cases the pressure does not exceed ~10 % of the design pressure. The highest calculated pressure value is 112 % of design pressure.

In most cases film boiling is avoided. Only in some cases of "loss of main heat-sink" critical heat flux is exceeded. However in these cases the maximal calculated cladding temperatures are below 700 °C for the 8 x 8 cases (KKI) and below 850 °C for the 7 x 7 cases (KKB). Maximal fuel center temperatures are in these cases 2150 °C (8 x 8) respectively 2840 °C (7 x 7).

The maximal amount of steam blown into the pressure suppression system within the first 200 seconds following the start of the transients is less than 50 full-power seconds, resulting in a 12-K temperature increase of the pool-water (After this period the reactor is subcritical).

In cases of loss of feedwater flow or onsite power the core shroud (fuel-bundles, upper core plenum, separators) remains completely filled with water. Outside the shroud the level, where the start of the emergency core cooling systems is initiated, is defined only for some serials. Therefore the depressurization system is not started.

Next Steps

PWR: Additional work is still under consideration.

BWR: Investigation of ATWS accidents; Improvement of the computer codes; Parameter variations: rod input time, slow down time of the pumps.

Relation with Other Projects

see RS 153 - A 74

Reference Documents/Degree of Availability

No reports available.

PROJECT TITLE : Dynamic studies for safety analysis	CLASSIFICATION 4 _ 8
SPONSORING COUNTRY : Italy	ORGANISATION : CNEN
DATE INITIATED : 1962 DATE COMPLETED : in progress	PROJECT LEADER : M. Di Bartolomeo

Description : Concern the development of analog and hybrid models for nuclear power plants for dynamic studies, stability problem accident analysis, etc.

Facilities

- 2 analog computers EAI PACE 231/R
- 1 hybrid computer EAI 8945
- 1 hybrid computer EAI PACER 700

Reference documents:

1) P.Giordano - A.Mathis - G.Melucci
Dynamics and control studies for a steam-generating pressure-tube reactor
Doc.CNEN RT/ING(65)13 - Sept.1965

2) A.Mathis - B.Musso - E.Turrini
Accident Analysis for a fast source reactor
Energia Nucleare - Vol.15 - n.7 - Luglio 1968

3) P.Giordano - A.Mathis - B.Rimini
Analog methods for studying the space-time dynamics of nuclear reactors
Proceedings of 5th Congress of the Int.Ass. for An.Comp.
Losanna 1967.

4) A.Mathis
The Use of hybrid computers in the italian CNEN nuclear program.
Conference on "The Effective Use of Computers in the Nuclear Industry.
Knoxville, Tenn.(USA) April 21-23, 1969.

5) P.Giordano - A.Mathis - O.Modonesi
Use of analog and hybrid computers in the design of CIRENE type nuclear power plant.
Enlarged Halden Programme Group meeting on Computer Control
LOEN,(Norway) - May 29th - June 2nd, 1972.

4-8 CNEN

Classification 4.1

Title 1 ANDYCAP: Tré dimensional dynamisk model af kogendevands reaktor kerne.	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 ANDYCAP: 3-D-dynamical model of a BWR-core.	Project leader: P. Skjerk Christensen
Initiated: 1969 Completed: 1972 Status: in use, being improved Last updating: Currently	Scientists: P. Skjerk Christensen

1. General aim

The purpose of the model is to describe and follow transients in a BWR core due to pertubations of process variables in timescale 1-100 seconds.

2. Particular objectives

The project is particularly aimed at normal and abnormal conditions in the reactor. The model is based on a three dimensional nodal description of the core as the neutronic part whereas the hydraulics model consists of a number of parallel one dimensional channels coupled at the lower and upper plenum. A recirculation loop containing a pump is included. In practical calculations the number of nodes has to be limited to some 2000, and the number of hydraulic channels to 30, due to the computer time which on a CDC-6600 is a factor of 100 times the reactor time, strongly depending on the character of the transient. The transients can be initiated by control rod movement, steam load disturbance, feed water disturbance, and main circulation pump disturbance.

3. Experimental facilities

4. Project status

1. Progress to date: A version of the code is in use
2. Essential results:

5. Next steps

Work is in progress directed to speed up the code.

6. Relation with other projects

7. Reference documents

To be issued

8. Degree of availability

Not available

Classification 4.1

Title 1 Kontrolstavs udskydning i en kogendevands-reaktor	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 Rod ejection transients in BWR	Project leader: P. Skjerk Christensen
Initiated: 1974 Completed: Status: in progress Last updating: currently	Scientists: B. Thorlaksen

1. General aim

The aim is to construct a model which describes the reactor transient following the ejection of one of the control rods in a BWR. The model is an extension of the ANDYCAP code which describes the dynamics of a 3-D nodal model of a BWR-core and pressure vessel. The movement of the control rod is calculated assuming critical flow in the guide tube, and the velocity limiter on the rod has been taken into account.

2. Particular objectives

1. The power distribution is calculated in three dimensions, but the local peaking factor has to be found by adequate box calculations.
2. The fuel model includes the possibility of a steam explosion initiated by molten fuel.
3. The consequences of the transient pertinent to the core and pressure vessel are calculated.

3. Experimental facilities

4. Project Status

1. Progress to date: A model for the movement of a control rod after a failure of the housing has been made.
2. Essential results

5. Next steps

6. Relation with other projects

7. Reference documents

The project is a Ph.D. study and the final report will be issued autumn 76.

8. Degree of availability

The thesis will be freely available.

Classification 4.1, 4.2, 4.3

Title 1 PWR-stations dynamik model	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 PWR: A PWR power plant dynamics model	Project leader:
Initiated: 1972 Completed: Status: in progress Last updating: Currently	Scientists: P. la Cour Christensen P. Skjerk Christensen

1. General aim

The goal of the project is to describe and follow transients in a power plant comprising a PWR. The transients may be initiated by any process variable in- or outside the plant.

2. Particular objectives

The plant model must be able to calculate the transients in real time which however limits the number of space meshes. Furthermore, the model must be able to perform interactive calculations which means that the user is able to study immediately the results of his perturbations on the model. At last, the model must be able to serve as a tool used by investigation of control systems.

The model includes a one-dimensional core model and a single cooling loop comprising a circulation pump, a steam generator of the U-tube type, a pressurizer, and a boron injection system. The neutronic model is based on diffusion theory with a single prompt and three delayed neutron groups. The steady state is found by purely digital calculations while the transients are calculated mainly by analogue elements while some neutronic solutions still are calculated by digital techniques.

3. Experimental facilities

4. Project status

1. Progress to date: The models for the steady state and the transients are almost finished and the two parts have been coupled together. Simple transients have been run.
2. Essential results:

5. Next steps

6. Relation with other projects

7. Reference documents

In preparation Risö report no. 318

8. Degree of availability

PROJECT TITLE : Overpower tests	CLASSIFICATION 4.1
SPONSORING COUNTRY : Italy	ORGANISATION : CNEN
DATE INITIATED : March 1974 DATE COMPLETED : February 1976	PROJECT LEADER : C. Lepsky

Description :

1. General aim

Fuel cladding interaction and mechanical properties of irradiated cladding after rupture determined by power excursion.

2. Particular objectives

Investigating modes of rupture and material properties by neutron-radio graphy, dimensional analysis, etc. of irradiated cladding subjected to increasing power ramps up to power burst. Investigate the influence of gap (150, 230, 310 cold gap).

3. Experimental facilities and programme

Irradiation (Halden reactor) and post-irradiation examinations at CSN Casaccia Center of CNEN.

4. Project status

4.1 Progress to date

Irradiation completed at Halden (IFA 131) up to 30.000 MWD/t.
Non destructive post-irradiation tests of 6 rods prior to overpower test already completed.
First overpower test on 1 rod already performed including non destructive post-irradiation analysis of the same rod.
Experimental procedure and facilities for subsequent overpower tests already set-up.

./.

5. Next steps

Overpower tests performance will begin in 1975 on 16 rods with irradiation up to 30.000 MWD/t.

6. Relation with other projects

Interamp programme at Studsvik on standard irradiated rods, at different burn-up levels. This programme will start in September 1975 and will continue for three years.

7. Degree of availability : to a limited extent.

4.1 CNEN

TH - Delft	CLASSIFICATION: 4.1 4.2 4.3
TITLE: Ontwikkeling van een hybried computermodel voor de simulatie van storingen en ongevallen in een drukwaterreactor.	COUNTRY: NETHERLANDS SPONSOR: Ministry of Social Affairs ORGANIZATION: TH - Delft
TITLE (ENGLISH LANGUAGE): Development of a hybrid-computermodel for the simulation of transient and accident conditions in a PWR	PROJECTLEADER: Latzko
INITIATED: Oct. 1974 COMPLETED: 1978 STATUS: in progress LAST UPDATING: january 1976	SCIENTISTS: Bruens

1. General Aim

   Development of a calculational tool wich can compute the plant response to various transients and accident conditions (excl. LOCA) for a PWR. Provide the possibility to evaluate the effectiveness of control and protection systems under these conditions.

2. Particular Objectives

   Development of a hybrid-computermodel of a PWR. The nuclear core and the steam generator will be the basic modules. These and the other parts of the primary and secundary system will be modelled such that they can be easily adapted to any type of PWR.

3. Experimental facilities and programme: -

4. Project Status

   The following simulation programs are finished:
   - hybrid reactor core model describing the neutron-kinetics and the thermal behaviour
   - steam generator computer modules describing the thermal/hydraulic behaviour of the evaporator.

5. Next steps

   - computer model of the steam generator recirculation
   - hybrid model of the pressurizer
   - digital models of the turbine, reheater and generator
   - digital model of the preheater
   - coupling of the different computer programs.

6. Relations with other projects: -

7. Reference documents: -

8. Degree of availability

   through Ministry of Social Affairs.

<table>
<tr><td colspan="3"><u>Classification:</u> 4.3</td></tr>
<tr><td colspan="2"><u>Title 1 (Original Language):</u>
3D-Transientenprogramm - Modifizierung eines 3D-Transientenprogramms für den SWR
(RS 178 - II.1.2, Jahresbericht A 75)</td><td>COUNTRY:
BRD
SPONSOR:
BMFT
ORGANIZATION:
KWU, Offenbach</td></tr>
<tr><td colspan="2"><u>Title 2 (english):</u>
Modification of a 3 Dimensional Transient Code for BWR</td><td rowspan="2"><u>Project Leader:</u>
Dr. Lockau</td></tr>
<tr><td><u>Initiated (Date):</u>
September 1975
<u>Status:</u>
Continuing</td><td><u>Completed (Date):</u>
August 1977
<u>Last Updating (Date):</u>
December 1975</td></tr>
</table>

General Aim and Particular Objectives

For the accident analysis of a BWR a point kinetic or one dimensional model was used. Therefore rather conservative factors had to be regarded, when local effects were investigated. With a 3 dimensional calculation this conservative factors can be corrected, without loosing safety margin. Therefore a 3 dimensional program shall be developed, which gives more realistic details for the calculation of unsymmetric incidents in the core.

Experimental Facilities and Research Program

The central activity concentrates on the development of a thermo-hydraulic boiling-channel module for the 3 dimensional transient model. As a first step the physical model and the efficiency of the numerical method shall be reviewed. This concerns primarily the void coefficient of reactivity, which influences the power and power density distribution strongly by negative feedback. The 3 dimensional model must be able to treat 50 - 100 boiling channels in parallel.

Project Status / Progress to Date

As a first step the Standard Code FRANCESCA-MULTICHANNEL was investigated, which can be used for the calculation of 10 parallel channels, but without any feedback to neutron physics.

The tests showed that extension is possible in principal. The code however must get an overlay structure in order to save storage in the computer.

Some tests runs of CISE were recalculated in order to compare the physical models. The data were derived from pipe tests with quickly closing valves, which allowed to determine the void coefficient as a function of the flowrate, the pressure and the subcooling.

Project Status / Essential Results

The FRANCESCA model has already been compared with a stationary thermohydraulic model, which uses the Dix-correlation (improved Levy model); this gave the best results, comparing different pipe- and bundle tests. Another check was the recalculation of the CISE tests. The parameter field was

$$p = 50 - 90 \text{ bar}$$
$$g = 110 - 220 \text{ g/cm}^2\text{/sec}$$
$$\Delta T_{sub} = 4{,}38 \text{ and } 10{,}2\ ^{o}C$$

The recalculated results of FRANCESCA agreed well the measured values. The recondensation parameter was confirmed.

Next Steps

The recalculation of thermohydraulic experiments will be continued.

The codes IQSBOX and FRANCESCA will be coupled; the first step concerns organisation problems. The integration of the 3 dimensional system into a common physical program system will be discussed.

Relation with Other Projects

No relation with other projects.

Reference Documents / Degree of Availability

No reports available.

5. BEHAVIOUR, TRANSPORT AND RELEASE OF RADIOACTIVE SUBSTANCES

CLASSIFICATION

5.1

TITLE 1	ACTIVITE DES PRODUITS DE CORROSION	COUNTRY FRANCE SPONSOR E.D.F. ORGANIZATION E.D.F.
TITLE 2	RADIO-ACTIVITY OF THE CORROSION PRODUCTS	Project Leader E.D.F./DER/TEGAEH
Initiated juin 1974	Completed 12/1975	Scientists M. BUREAU BERGE
Status	Last updating : 20.01.75	

I - GENERAL AIM

Activité des produits de corrosion.

II - OBJECTIFS PARTICULIERS

Diminution de l'activité déposée sur les parois des ciucuits primaires afin de faciliter les interventions.

III - EXPERIMENTAL FACILITIES AND PROGRAMME

- étude de la dissolution des produits de corrosion (boucle SEPAI-E.D.F. CHATOU)
- étude du taux de relâchement en cobalt de différents alliages présents dans le circuit primaire (boucle SEPAI)
- essais de décontamination d'une boucle SEMA.

IV - PROJECT STATUS

4.1 - Progress to date

- début des essais sur SEPAI,
- campagne de mesures sur SEMA.

4.2 - Essential Results

La campagne de mesures effectuée à SENA a permis de montrer que les produits de corrosion, qui se dissolvent du fait de l'abaissement de température, sont efficacement retenus sur résines synthétiques.

V - NEXT STEPS

- fin des essais de dissolution sur SERAI,
- étude du taux de relâchement,
- essai de décontamination d'une boucle de SENA.

VI - RELATION WITH OTHER PROJECTS

Néant.

VII - REFERENCE DOCUMENTS

VIII - DEGREE OF AVAILABILITY

E.D.F.

Classification: 5.2	
Title 1 (Original Language): Versuche zur Erfassung und Begrenzung der Freisetzung von Spalt- und Aktivierungsprodukten beim Coreschmelzen (PNS 4243 - I.1.5., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Experiments on Determination and Limitation of Fission and Activation Product Release during Core Meltdown	Project Leader: Dr. Albrecht, DI. Perinic
Initiated (Date): 1972 / Completed (Date): 1978 Status: continuing / Last Updating (Date): December 1975	

1. General aim

Determination of the release fraction of the radioactive core inventory for various coremelting conditions.

2. Particular objectives

Quantitative investigation of the release of fission and activation products during core heat-up and from a liquid core melt, including also concrete; characterization of the physical and chemical behavior of the released products; development of techniques for reducing the release.

3.1 Experimental facilities

a) Melting plant SASCHA
   A high-frequency induction furnace in which Corium melts of about 1oo g can be produced in inert as well as in oxidizing atmosphere; extension of the facility for production of kg - melts is presently prepared

b) Transport and collection systems for the released products

c) Facility for production of slightly active Fissium (in construction)

3.2 Research program

Release experiments with
- 3o g of inactive Corium for testing and optimizing the experimental facilities
- 3o g of Corium containing activated steel and Zircaloy for determination of the release of Fe, Cr, Ni, Mn, Zr, Sn and U; variation of the parameters maximum temperature, atmosphere, pressure, flow rate
- 3o g of Corium containing slightly active Fissium with simulated burn-up in the range of 1o.ooo - 5o.ooo MWd/t; same parameters as above
- 3o g Fissium-Corium with additions of CaO, SiO_2, concrete and other materials
- 3oo/∿3ooo g Fissium-Corium with a reduced number of parameter values depending on the results of the previous experiments

4. Projectstatus

4.1 Progress to date

The installation of the melting plant SASCHA and the transport and collection system for the released products have been completed. After conducting performance tests with these facilities a series of preliminary experiments were carried out
- to study the behavior of corium melts in thoria crucibles
- to develop a draw-off system for the released products from the hot crucible region
- to find out if the vapors above the melt will influence the temperature measurement by causing a selective absorption of the wavelengths detected by the two-color pyrometer
- to determine the fraction of released products which is lost by deposition in the transport system before reaching the filters

In addition to these experiments the technique for analyzing inactive Corium deposits on the filters were tested and further improved.
Preparations for the production of Fissium have been started by installation of five shielded boxes in which the fuel and the simulated fission products will be weight, ground, mixed, sintered and finally sealed in Zircaloy capsules.

4.2 Essential Results

a) The investigations on the compatibility between Corium and ThO_2-crucibles have confirmed earlier results that a melt of 24oo - 26oo°C can be kept in the crucible hot longer than about 15 min. After this time, the crucible walls are washed out so far - especially in the range of the melt surface - that the melt will penetrate and run out.

b) For drawing-off the released products from the hot surroundings of the crucible a spherical vessel of Duranglass behaved quite satisfactory. After melting tests with temperatures of more than 27oo °C it was not broken nor did it show any appearance of softening. An essential advantage of this glass system is that the occurances above the crucible (flames, eruptions etc.) can be better observed than if a system of ceramic or metallic material is used.

c) Melting experiments with various steel components (Fe, Cr, Ni) in an argon atmosphere and with Corium under air did not show any selective influence of the vapors on the intensities of the wavelengths used for the temperature measurement.

d) Three release experiments have been carried out with 3o g of Corium under various atmospheres, with pressures in the range of 1.5 to 2.o bar and maximum temperatures > 25oo °C. The quantitative evaluation is not yet completed, but from the visual inspection of the glass system and the filters it seems, that in the case of argon and steam atmosphere the total amount of released material as well as the transport losses were relatively low.
After the experiment in air, however, the glass was covered by a dense brown layer which became less dense with increasing distance from the crucible. Hot molten particles which were thrown out from the melt during the oxidation phase had penetrated into the glass surface at several spots whereas the glass system remained almost optically clear in the first mentioned experiments.

5. Next steps

The most important work to be done in the near future will be

- quantitative evaluation of the release experiments described above
- continuation of these experiments (cf. 3.2.)
- application of a condensation nuclei counter for determination of the aerosol concentration during the release experiments
- completion of the installations for the Fissium production

6. Relation with other projects

PNS-4241: Experimental investigation of the meltdown phase of UO_2-Zircaloy fuel rods under conditions of failure of emergency cooling

PNS-4244: Constitution and reaction behavior of LWR materials at core melting conditions.

7. Reverence documents

Report KFK - 21o3 (1974) p. 144-64 (in German)
Report KFK - 213o (1975) p. 64-66 (in English)
p. 251-6o (in German)
Report KFK - 2195 (1975) p. 82-85 (in English)
p. 318-37 (in German)
Report IRS-F-21 (1974) p. 1o5- 7 (in German)
Report IRS-F-23 (1975) p. 137-39 (in German)
Report IRS-F-24 (1975) p. 195-98 (in English)
Report IRS-F-26 (1975) p. 135-38 (in German)

8. Degree of availability

Unristricted distribution

Classification 5.2	
<u>Title 1</u> Berekening van hoeveelheden radioactiviteit vrijkomend bij een ernstig reactorongeval	<u>Country</u> The Netherlands <u>Organization</u> KEMA
<u>Title 2</u> Calculation of the quantities of radioactivity released as a result of a serious reactor accident	<u>Physicist</u> K.P. Termaat

1. General aim

In analysing the risk of a nuclear power plant for the surrounding population one has, amoung other things, to consider the quantity and the nuclide spectrum of the radioactivity released to the environment as a result of a reactor accident with a non zero probability.

2. Particular objectives

Calculations are performed with the programme CORRAL developed by Battelle NW USA. Minor changes were introduced.
The basic objective has been to contribute in the risk analysis referring to an enlarging nuclear power programme in this country.

3. Experimental facilities

Not applicable.

4. Project status

Necessary calculations have been performed. An improvement in applied input data to the programme CORRAL could be usefull.

5. Next steps

Not applicable.

6. Relation with other projects

Equivalent calculations are performed in the USA as a contribution to the Rasmussen-study (WASH-1400).

7. Reference documents

See 2 and 6.

8. Degree of availability

Internal report.

Classification: 5.3	
Title 1 (Original Language): Untersuchungen zur Wechselwirkung von Spaltprodukten und Aerosolen in LWR-Containments (PNS 4311 - I.1.4., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Radioactive Pollutants in the Post Accident Atmosphere of the LWR-Containment	Project Leader: Dr. W. Schöck
Initiated (Date): January 1972 Status: continuing Completed (Date): December 1977 Last Updating (Date): December 1975	

1. General Aim

The deposition of radioactive aerosols inside the containment building between formation and release to the environment has been recognized as a means of mitigating the radiological consequences of hypothetical accidents. To be able to describe and assess the various attenuation and removal mechanisms adequately will reduce unnecessary conservativism in the safety analysis.

2. Particular Objectives

The objective of the project is to describe the removal of airborne particulate radioactivity from the post accident atmosphere of an LWR-Containment. For this purpose a computer code NAUA will be developed on the basis of an experimentally verified numerical model.

3. Experimental Facilities and Program

After completion of a preliminary model NAUA Mod1, the experimental facility is specified and constructed. Phase 1 of the experimental program is intended to produce the necessary input information for the model and to verify the model assumptions.

Phase 2 comprises experiments with aerosols produced from corium like materials and will be followed by extrapolation to containment situations.

4. Project Status

4.1. Progress to Date

NAUA Mod1 has been completed containing steam condensation and the aerosol removal processes coagulation, diffusion, sedimentation and thermophoresis. The conception of the experimental vessel and the instrumentation is completed.

4.2. Essential Results

The Mod1 was extended to contain an arbitrary steam source. The effect of generating a bimodal aerosol size distribution by heterogeneous steam condensation has been studied extensively under various conditions. A qualitative experimental proof of this distribution was gained.

The first calculations with Mod1 exhibit a rather rapid decay of the dry particle peak of the bimodal distribution due to coagulation with the droplet peak. Thus the decay of the droplet peak will play the main role in the removal of the airborne particles.

5. Next Steps

A sensitivity study with Mod1 will lead to the construction of the experimental facility and to the definition of the experimental program. Then the experiments of phase 1 can be started.

6. Relation with Other Projects

PNS 4243

7. Reference Documents

PNS semi annual report 2/1974, KFK 2130 (1975)(German)
PNS semi annual report 1/1975, KFK 2195 (1975)(German)

8. Degree of Availability

Unrestricted distribution

Classification: 5.3	
Title 1 (Original Language): Krypton- und Xenon-Entfernung aus der Abluft kerntechnischer Anlagen (PNS 4240 - II.6.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Separation of Krypton and Xenon from the Offgas of Nuclear Facilities	Project Leader: R. v. Ammon E. Hutter
Initiated (Date): January 1972 Completed (Date): December 1976 Status: continuing Last Updating (Date): December 1975	

1. General Aim

Development of a process for the separation of the fission noble gases from the offgas of future reprocessing plants.

2. Particular objectives

The operation of a pilot plant should demonstrate the noble gas separation with a simulated inactive offgas and thus provide data for the lay out of the offgas purification system of a large reprocessing plant. The method used is low temperature rectification. Two prepurification steps of the offgas before the rare-gas separation are also studied on a semi-works scale: reduction of oxygen and nitrogen oxides with hydrogen and separation of water and carbon dioxide by adsorption on molecular sieves.

3.1 Experimental facilities

A low temperature rectification unit (gas throughput 50 Nm^3/h) has been constructed. Molecular sieve adsorption and O_2-, NO_x-reduction units will be installed on the same scale. Both prepurification steps are also studied on a laboratory scale (1 Nm^3/h throughput).

3.2 Research program

Prepurification steps: Advantages and drawbacks of two alternatives for the O_2-, NO_x-reduction (catalytic or thermal reaction) will be evaluated. The influence of catalytic poisons like fission iodine and organic phosphorus compounds on the stability and activity of the catalysts is being studied. - Adsorptive capacities of molecular sieves for CO_2 and H_2O in the presence of trace amounts of NO_x and NH_3 have to be determined. - A method for the measurement and controlled decomposition of ozone formed in the rectification columns is to be found.

Low temperature rectification: the effectivity of the two separation columns will be determined, and thus data concerning the achievable Kr-decontamination of the process-gas as well as the Kr-enrichment and achievable Xe-purity in the second column will be obtained.

4. Project status

4.1 Progress to date

The low temperature rectification pilot plant has been laid out and constructed, a safety analysis for the operation of the plant has been carried out and plans for the simulation of the heating effect of Kr-85 have been considered. The molecular sieve adsorption pilot plant has been ordered.

The laboratory scale adsorption test apparatus has been installed. 11 commercial catalysts for the reduction step have been tested in the laboratory. One catalyst (0,3 % Ru on Al_2O_3-carrier) has been selected for further tests. Criteria were maximum NO-reduction and minimum NH_3 production. A series of poisoning tests on this catalyst has been carried out.

4.2 Essential results

All 11 noble metal catalysts studied reduce NO and O_2 with H_2 somewhat in excess of the stöchiometric amount down to values <1 ppm at temperatures $\geq 400^\circ C$. However, the more active of them (mainly those on a Pd or Pt basis) also accelerate the formation of NH_3 from the carrier gas N_2 and H_2. Eccessive NH_3-yields are obtained there. Thus, a Ru-catalyst with medium specific surface area (~ 100 m^3/g) has been selected where NH_3-formation is ≤ 20 ppm at $500^\circ C$. Samples of this catalyst poisoned with varying amounts of tributylphosphate (TBP) showed some increase of the temperature of beginning reaction and some increase of the NH_3-formation, but no decrease of the NO-reduction at 400 - $500^\circ C$.

5. Next steps

The low temperature rectification plant will be installed during summer of 1976. The test program will start soon afterwards. The molecular sieve pilot plant will be installed at the beginning of 1977.

Laboratory tests on the adsorption properties of molecular sieves towards rare gases and trace impurities in the presence of large amounts of H_2O will start early in 1976. Laboratory studies on the catalytic poisoning and reaction kinetics of the O_2-NO_x-reduction with H_2 will continue. The pilot plant will be specified and ordered late in 1976.

6. Relation with other projects

PNS 4112: Development of Exhaust Air Filters for Reprocessing Plants

7. Reference Documents

a) R. v. Ammon, W. Weinländer, E. Hutter, G. Neffe und C.H. Leichsenring: Entwicklung der Krypton-85-Abtrennung aus dem Abgas der großen Wiederaufarbeitungsanlage, KFK-Nachrichten 7 (3), 63 (1975) (in German)
b) E. Hutter, G. Neffe, R. v. Ammon, W. Weinländer und C.H. Leichsenring: 1. Halbjahresbericht 1975 PNS, KFK 2195 (1975) (in German, English abstracts)
c) R. v. Ammon, Jahreskolloquium 1975 PNS, KFK-Report 2244 (1975)

8. Degree of availability

Available through GfK

Classification: 5.3	
Title 1 (Original Language): Spalt-Jod-Abscheidung in Kernkraftwerken und Wiederaufarbeitungsanlagen: Störfall-Umluftfilter zur Abscheidung von Spaltprodukten aus der Sicherheitsbehälter-Atmosphäre (PNS 4110/4111 - II.6.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Fission Product Iodine Removal in Nuclear Power Plants and Reprocessing Plants: Post Accident Recirculation Air Cleanup for Fission Product Removal from the Containment Atmosphere	Project Leader: J.G. Wilhelm H.G. Dillmann
Initiated (Date): 1971 Status: continuing	Completed (Date): 1976 Last Updating (Date): December 1975

1. General aim

Fission product removal from the containment atmosphere by post-accident recirculating air filter systems.

2. Particular objectives

Removal of fission product iodine and aerosols.

3. Experimental facilities and research program

3.1. Technical test rig for testing of original filter components (iodine and aerosols) under simulated accident conditions.

Gas flow : up to 2000 m^3h^{-1}

temperature: up to 200°C

r. h. : up to 100 % at temperatures $\leq$ 151°C

pressure : up to 5 bar

Laboratory scale rigs for testing of sorption materials.

3.2. Testing of sorption materials in annular filter beds for operation under steam atmosphere under pressure and at high temperature.

Examination of aerosol filters under simulated accident conditions. Testing of single technical components e. g. moisture separators, heaters, prefilters under simulated accident conditions.

Irradiation tests on fiber mats and aerosol filters, respectively.

Testing of complete filter systems under simulated accident conditions and, if necessary, improvement of the systems (in cooperation with other partners).

Experimental investigations concerning Ru, Cs, Te filtration from the containment atmosphere.

4. Project status

4.1. Progress to date

A prototype post-accident recirculation filter has been tested under various conditions in the filter test rig.
The adsorber material AC 6120 as well as molecular sieves were examined for transportation of the impregnation by the air-steam mixture and for their stability, respectively, under simulated accident conditions. A fundamental concept was elaborated for an aerosol generator working under simulated accident conditions. Investigations have been made to find noncorrosive test aerosols.

4.2. Essential results

The prototype post-accident recirculation filter was exposed to hot air of $160^{0}C$ and to steam up to 1 bar at temperatures between 108^{0} and $160^{0}C$. After 18 days of operation it yielded removal efficiencies of $\geq$ 99.98 % for $CH_3{}^{131}I$ (residence time 0.2 sec, Ø air flow 1200 m^3h^{-1}).
An iodine filter element was exposed for more than 10 days to an air-steam mixture of $160^{0}C$ and 72 % r. h. at a pressure of 4 bar. After that time a penetration of < 1 % for radioactive methyliodide was found (residence time 0.2 sec).
The laboratory experiments with AC 6120 performed in superheated steam over a period of 48 h yielded no measurable transportation of impregnation under different conditions up to temperatures of $200^{0}C$.

5. Next steps

Experiments under simulated accident conditions will be continued with higher pressure, relative humidity and temperature of the air-steam mixtures.
Investigations of the sorption materials are to be continued.
Tests of aerosol filters under simulated accident conditions will be carried out after installation of the aerosol generator in the technical filter test rig.

6. -

7. Reference documents

Report KFK 2130 (1975) p. 77 (german with english abstracts)

Report KFK 2195 (1975) p. 98 (german with english abstracts)

8. Degree of availability

Unrestricted distribution

Classification: 5.3	
Title 1 (Original Language): Spalt-Jod-Abscheidung in Kernkraftwerken und Wiederaufarbeitungs-anlagen: Entwicklung von Abluftfiltern für Wiederaufarbeitungs-anlagen (PNS 4110/4112 - II.6.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Fission Product Iodine Removal in Nuclear Power Plants and Reprocessing Plants: Development of Off-Gas Filters for Reprocessing Plants	**Project Leader:** J.G. Wilhelm J. Furrer K. Jannakos
Initiated (Date): 1971 **Completed (Date):** 1976 **Status:** continuing **Last Updating (Date):** December 1975	

1. General aim

To remove fission product iodine and other contaminants from off-gas of fuel reprocessing plants a filtration process is under development capable to retain the fission product iodine, including ^{129}I, in a form ready for final storage.

2. Particular objectives

Filters have been developed for decontamination of the off-gas from large reprocessing plants. The filter systems should be able to handle the majority of the fission product iodine and other contaminants from spent fuel elements, including contaminants in the form of aerosols. The loaded iodine adsorber material should be in a form ready for final storage of the ^{129}I without needing additional processing.

The final objective is to develop, build and test a prototype off-gas filter system retaining the aerosols and iodine from the off-gas of the reprocessing plants.

3. Experimental facilities and program

3.1. Different test rigs in reprocessing plants and laboratories including a large pilot rig.

3.2. Analysis of reprocessing off-gas in SAP Marcoule and WAK (reprocessing plants). Testing of AC 6120/H_1 iodine sorption material in off-gas from reprocessing. Construction of facilities for testing of filter elements under simulated condition of off-gas from reprocessing.

Development and testing of a halogen detector for the measurement of the iodine concentration in the dissolver off-gas.

Design and construction of a prototype iodine filter for large reprocessing plants for a volume flow of 250 Nm^3/h. Test operation performed with the prototype includes testing of the remote handling system as well as the transport of the exposed filters to the final storage facility.

4. Project Status

4.1. Progress to date

The test of the highly impregnated iodine sorption material AC 6120 was continued in the laboratory under simulated conditions of a 1500 t/a uranium reprocessing plant. The influence of NO in nitrogen as the carrier gas on the removal efficiency of the sorption material has been especially examined.

Several experiments on the removal efficiency in iodine sorption of AC 6120 were carried out in the original off-gas of the French reprocessing plant SAP Marcoule while the filter system was installed on the one side directly behind the dissolver and on the other side behind the iodine desorption column.

The first iodine filter developed at the LAF II was installed in the off-gas line of the dissolver in the Karlsruhe reprocessing plant.

The filter system for the dissolver off-gas handling test rig of the IHCh was specified and ordered with an engineering firm.

The conception ot the prototype off-gas filter system was selected and a lock and transport system allowing to replace filters was designed and subjected for testing.

Five alternative solutions were set up in order to find the appropriate filter concept. The method of selection based on the evaluation of performance criteria. According to the selected concept a filter drum was designed and constructed. The lock of the filter system has been designed and realized. Preliminary tests have been made.

4.2. Essential results

The influence of NO in N_2 as the carrier gas will be harmful if there is no oxygen or NO_2 present to reoxidize the reduced $AgNO_3$ impregnation.

The experiments in the real dissolver off-gas showed the high removal efficiency and the high capacity of the iodine sorption material. The removal efficiencies of this material installed directly behind the dissolver in the off-gas behind the adsorption-desorption column were greater than 99.9 %. In both cases the material had a high loading capacity. The filter installed in the dissolver off-gas line of the Karlsruhe reprocessing plant is loaded with the total iodine released from the dissolver solution. The removal efficiency after an operational period of now 2.5 months is still > 99.99 % (residence time: 0.4 s).

With respect to the prototype filter system first tests covering the filter drum and the lock system have yielded satisfactory results. They related to the tightness and loading of the filter drum and to the performance of the double cover of the lock.

5. Next Steps

After startup of the test rig it is scheduled to test the iodine filter, the filter drum and other components of the off-gas filter system.

6. Relation with Other Projects

The iodine removal from the dissolver off-gas will be a necessary step for gas-cleaning with respect to the ^{85}Kr-seperation process (PNS 4140).

7. Reference documents

KFK 2195, p. 105 (1975)
KFK 2262, to be published in 1976

WILHELM, J.G.; DEUBER, H.; DILLMANN, H.G.; FURRER, J.
Spaltjodabtrennung in Kernkraftwerken und Wiederaufarbeitungsanlagen
Jahreskolloquium des Porjektes Nukleare Sicherheit, Karlsruhe
17. November 1975
KFK-2244 (November 1975) S. 65 - 84
WILHELM, J.G.
Die Reduktion der Jodemissionen aus kerntechnischen Anlagen durch Jodfilter.
NUCLEX 75, Basel, 7. - 11. Oktober 1975,
Kolloquium D2, Paper 3; 10 S.
KAEMPFFER, R.; WILHELM, J.G.; (Hrsg.)
Laboratorium für Filterrechnik
Jahresbericht. 1974
KFK-2165(Juni 1975).

8. The distribution of the KFK report is not restricted.

Classification: 5.3	
Title 1 (Original Language): Spalt-Jod-Abscheidung in Kernkraftwerken und Wiederaufbereitungsanlagen: Abluftfilter an Reaktoren, Alterung und Vergiftung von Jod-Sorptionsmaterialien (PNS 4110/4114 - II.6.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Fission Product Iodine Removal in Nuclear Power Plants and Reprocessing Plants: Exhaust Air Filters at Nuclear Installations, Ag ing and Poisoning of Iodine Filters	Project Leader: J.G. Wilhelm, H. Deuber
Initiated (Date): July 1971 Completed (Date): 1976 Status: continuing Last Updating (Date): December 1975	

1. General aim

Improvement in the operational behaviour and removal efficiency of exhaust air filters.

2. Particular objectives

Provisions to control the effects of solvent loading on the removal efficiency of iodine filters.

3. Experimental facilities and research program

3.1. Laboratory scale test rigs for the measurement of removal efficiencies of solvent loaded iodine sorption materials.
Test rigs in nuclear power plants.

3.2. Determination of poisoning compounds for iodine filters.
Measurement of retention times of poisoning compounds on iodine sorption materials.
Design of iodine filter assemblies with sufficient operational time.

4. Project status

4.1. Progress to date.
The influence of solvent loading on the removal efficiency of iodine filter charcoal was examined for the exhaust air filter in the shutoff rooms of an LWR nuclear power station. The first laboratory studies for the determination of the retention times of poisoning compounds were carried out.

4.2. Essential results

The loading of solvent in the impregnated charcoal of an exhaust air filter varied between 7.3 and 12.4 wt %. The removal efficiency for ^{131}I in form of CH_3I was substantially reduced by adsorption. The measurements with upstream activated filter charcoals in a nuclear power station showed that the solvent components xylene and toluene occurring in large quantities under the usual conditions of shutoff room exhaust air filtering cannot be retained sufficiently long. The high-boiling components of the petroleum-gasoline fraction, which also occur in the exhaust air and substantially contribute to iodine sorption filter loading through pollutants, were retained over a period of several months by relatively low-depth upstream filter beds.

5. Next steps

The measurement of retention times under controlled conditions of coadsorption of solvents and water will be continued, also prefilters will be tested in nuclear power plants.

Identification of Radioiodine Species in the Off-Gases of Nuclear Installations

1. General aim

Improvement in the assessment of the environmental impact of radioiodine in the off-gases of nuclear installations.

2. Particular objectives

Identification of the radioiodine species.

3. Experimental facilities and research program

3.1. Laboratory scale rigs for the generation of iodine species and testing of sorption materials.

3.2. Development of iodine samplers for the discrimination of elemental iodine, methyl iodide and possibly other iodine species as hypoiodous acid; application of discriminating iodine samplers in off-gases of nuclear installations.

4. Project status

4.1. Progress to date.

A discriminating iodine sampler for the application in off-gases of nuclear power stations is being developed. Different materials eligible as selective sorption materials for elemental iodine are tested with respect to their removal efficiencies for elemental iodine and methyl iodide. - Experiments on the volatilization and removal of hypoiodous acid have been made.

4.2. Essential results.

AC 6111, impregnated with potassium iodide, proved to be the most suitable of the investigated selective sorption materials for elemental iodine under certain conditions. When tested in the entire range of parameters, which is of significance for the examination of off-gases from nuclear power stations, it caused a negligible removal of methyl iodide. - An iodine species was volatilized from solutions of carrier-free iodide and elemental iodine which was difficult to remove with AC 6120. This species did not exhibit the properties expected for hypoiodous acid.

5. Next steps

The tests of AC 6111, impregnated with potassium iodide, will be continued with elemental iodine in the significant range of parameters. Measurements in the off-gases of nuclear power stations will be made. The development of a discriminating iodine sampler applicable in off-gases of reprocessing plants will be started.

7. Reference documents

Report KFK 2130 (1975) p. 96 (german with english abstracts)
Report KFK 2195 (1975) p. 113 (german with english abstracts)

8. Degree of availability

Unrestricted distribution

Classification 5.3

Title 1 Gas Phase Trapping Studies (1)		**Country** UK
Title 2		**Sponsor** UKAEA-CEGB **Organisation** Reactor Laboratory, Windscale
Initiated 1973 **Status** Continuing	**Completed** **Last updating**	**Project leaders** J.J.Hillary

General Aim

Improvement and standardisation of aerosol trapping with particular reference to normal emissions from reactors.

2. Particular Objectives

To provide design data relative to the possible problem of removing Sulphur 35 from normal emissions.

3. Project Status

Some difficulty with analytical techniques has been encountered, and until finally resolved, this has made quantitative interpretation of results somewhat uncertain. Nevertheless, it appears that useful trapping efficiencies could be achieved with suitably impregnated charcoal using a coolant - oxygen (20% mixture). Rather poorer efficiencies appear to obtain with coolant which was diluted with a large excess of air, a condition more closely representing the likely operational requirements.

Classification 5.3

Title 1 Gas Phase Trapping Studies (2)	**Country** UK
Title 2	**Sponsor** UKAEA
	Organisation Reactor Laboratory, Windscale
Initiated 1972 **Completed**	**Project leaders**
Status Continuing **Last updating**	J. J. Hillary

1 **General Aim**

Improvement and standardisation of aerosol trapping, with particular reference to normal emissions for reactors.

2. **Particular Objectives**

To define and thus control the qualities of charcoal which affect ageing.

3 **Experimental Facilities and Programme**

Apparatus is being set up for controlled static ageing tests of a large number of samples with typical atmospheric impurities.

4 **Project Status**

Apparatus is being commissioned.

Classification 5.3

Title 1	Gas Phase Trapping Studies (3)	Country: UK
Title 2		Sponsor: UKAEA Organisation: Windscale Reactor Laboratory
Initiated: 1972 Status: Continuing	Completed Last updating	Project leaders J.J.Hillary

1 General Aim

Improvement and standardisation of aerosol trapping, with particular reference to normal emissions from reactors.

2 Particular Objectives

To define and thus control the manufacturing variables, which affect the ability of charcoal to retain methyl iodide.

3. Experimental Facilities and Programme

A rig is in use in which methyl iodide at a defined concentration is passed through well characterised charcoal samples.

4. Project Status

About 500 samples of charcoal have been characterised and tested. The results are now being analysed statistically for correlations.

5. Reference Documents

Not expected until completion of analysis.

Classification 5.3

Title 1 Gas Phase Trapping Studies (4)		**Country** UK
Title 2		**Sponsor** UKAEA **Organisation** Windscale Laboratory
Initiated 1973 **Status** Continuing	**Completed** **Last updating**	**Project leaders** J. J. Hillary

1 **General Aim**

Improvement and standardisation of aerosol trapping with particular reference to normal emissions from reactors.

2. **Particular Objectives**

To define the efficiency with which I131 may be removed from ventilating air at ver low concentrations.

3. **Experimental Facilities and Programme**

A rig is under construction for the mixing process under controlled conditions.

4. **Project Status**

During normal operation of a power reactor, radioactive iodine 131 may be released to the surrounding atmosphere; there may be interest in monitoring down to levels as low as 10^{-15} ci/m^3 (implying about 10^{-22} kg/m^3). At plant outlets, the efficie of trapping plant may be of interest down to 10^{-11} ci/m^3. These are very much low than the levels of 10^{-7} kg/m^3 used in trapping experiments.

Preliminary work is aimed at the development of the measurement technique.

Reference Documents

Not yet available.

PROJECT TITLE : Testing of the Filters used in Nuclear Plants	CLASSIFICATION 5.3
SPONSORING COUNTRY : ITALY	ORGANISATION : C.A.M.E.N.-S.Piero a Grado (PISA) UNIVERSITY OF PISA
DATE INITIATED : End of 1967 DATE COMPLETED : End of 1976	PROJECT LEADER : S. LANZA (University) M. MAZZINI (University) G. SARNO (CAMEN)

a Description :

Research program:

To set up methods for testing of HEPA and iodine filters, both in laboratory and in situ, with reference to standard and accident conditions.

Facilities:

A rig and the related fittings for testing: HEPA filters by NaCl, DOP and condensation nuclei methods.

- charcoal filters by Freon and methyl iodide methods.

A rig for testing methyl iodide trapping efficiency of granular beds of materials such as charcoal and molecular sieves under rigidly controlled experimental conditions.

Reference documents:

1. G.CURZIO, A.GENTILI
Rimozione di gas nobili prodotti per fissione negli impianti nucleari.
Convegno sulle attivita' di ricerca nel campo della sicurezza degli impianti nucleari ed i metodi di calcolo e di prova di strutture soggette a vibrazioni.
Pisa (21 - 26 Settembre 1960).

2. S.LANZA et alii

Il controllo dei sistemi filtranti installati negli impianti nucleari.

Atti del Convegno sulle attivita' di ricerca nel campo della Sicurezza degli Impianti Nucleari svoltosi a Pisa - 21-23 settembre 1970. CNEN, Serie Simposi. Roma, 1971.

3. A.GENTILI

Caratterizzazione di filtri di carbone attivo per la ritenzione di gas nobili radioattivi prodotti per fissione.

XVII Congresso Nazionale dell'Associazione di Fisica Sanitaria e Protezione contro le radiazioni (Monte Porzio Catone 5-7 ottobre 1971).

4. S.LANZA, M.MAZZINI

Influenza dell'avvelenamento con vapori nitrici sulla ritenzione di CH_3I da parte di zeoliti argentate.

Istituto di Impianti Nucleari dell'Universita' di Pisa - RL 111 (72).

Tipografia Editrice Pisana, Pisa (1972).

5. S.LANZA et alii

Testing Methods for Iodine Filters of Nuclear Plants.

Proceedings of the Seminar on Iodine Filter Testing sponsored by CCE at Karlsruhe (RFT), 4 ÷ 6 december 1973.

Related projects:

.5.3 (Removal of Iodine)

Remark:

Program is sponsored by CNEN. A few peculiarities are investigated under CNEN contract jointly with the CAMEN.

5.3 CAMEN/ Pisa Un.

PROJECT TITLE : Removal of Iodine from Containment Atmosphere by Sprays with PSICO 10 Facility	CLASSIFICATION 5.3
SPONSORING COUNTRY : ITALY	ORGANISATION : C.A.M.E.N.- S.Piero a Grado (PISA) UNIVERSITY OF PISA
DATE INITIATED : 1967 DATE COMPLETED : 1976	PROJECT LEADER : R. MIRANDOLA (Univers.) G. SARNO (CAMEN)

Description:

The program has been set up with the aim of collecting experimental information for a correct evaluation of the efficiency of spray systems used in several nuclear plants for the removal of iodine released in the containment after a LOCA.

Twelve runs on molecular iodine removal by sprays were carried out in the 95 m^3 PSICO 10 model containment vessel. Both service water and a water solution containing 1% sodium thiosulphate were sprayed through different nozzles with, in some cases, recirculation of the sprayed solution and fractions of the model containment vessel volume not sprayed.

Further runs are scheduled for the next two calendar years in order to have information on the two hour dose reduction factor with longer fresh spray duration and with different nozzles and other spray and containment atmosphere conditions.

Reference documents:

1. B.GUERRINI, M.MAZZINI, R.MIRANDOLA, G.PETRANGELI
   PSICO 10:a facility for testing in the field of containment technology for nuclear plants.
   Ingegneria Nucleare - Marzo-Aprile 1969.

2. B.GUERRINI, S.LANZA, M.MAZZINI, R.MIRANDOLA
   Scalbatraio Center for research in nuclear safety.
   Nuclear Technology - April 1971.

3. B.GUERRINI, S.LANZA, M.MAZZINI, R.MIRANDOLA
Containment spray experiments with the PSICO 10 facility.
Energia Nucleare - Luglio 1972.

4. S.BARSALI, R.BOVALINI, F.FINESCHI, B.GUERRINI, S.LANZA, M.MAZZINI, R.MIRANDOLA
Removal of iodine from containment atmosphere by sprays: Research of the University of Pisa.
VII Congrès Internationale de la Société Francaise de Radioprotect
Versailles (France) 28/31 - Mai 1974.

5. S.BARSALI, R.BOVALINI, F.FINESCHI, B.GUERRINI, S.LANZA, M.MAZZINI, R.MIRANDOLA
Removal of iodine by sprays in the PSICO 10 model containment vessel.
Nuclear Technology - August 1974.

6. G.SARNO, S.MANFREDINI
Relazione sul ciclo di esperienze effettuate durante il 1974 sullo studio PE-1 (abbattimento dello iodio).
In the press (CNEN publication).

Remarks

The program has been sponsored by CNR and CNEN.

PROJECT TITLE . Techniques for Testing Charcoal Absorbers for Iodine and its Derivatives	CLASSIFICATION 5.3
SPONSORING COUNTRY : ITALY	ORGANISATION : ENEL
DATE INITIATED : 1970 DATE COMPLETED -	PROJECT LEADER : G. Sandrelli

Description :

1. **General Aim**

Development of methods to test the adsorption efficiency of charcoal absorbers for iodine and its alkyl derivatives.

2. **Particular Objectives**

The research has been concentrated on methyl iodide.

3. **Experimental Facilities and Programme**

All the tests are carried out at the Laboratories of the Politecnich Institute of Milan.

4. **Project Status**

Two experimental test methods have been developed: one (discontinuous) is based on an activation analysis of gas samples; in the other (continuous) a small gas flow is sent directly to a gas-chromatograph detector. The comparison between data obtained by using the two different techniques allows a better interpretation of the test results.
A mathematical model has also been developed to determine the methyl iodide concentration downstream of the absorber as a function of time. With the aid of this model, it is possible to avaluate the influence of operating conditions and to limit their negative effects.

5. **Next steps**

Extension of the tests to low methyl iodide concentrations such as those expected in the annulus of a double containmant system in the case of a LOCA.

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5. Reference Documents

- Facchini, Sandrelli, Teatini, Terrani, "A research carried out by Enel and Politecnico of Milan on the formation and the removal of Iodine Alkyl Derivatives".
Seminar on Iodine Filter Testing, Karlsrhue, December, 1973.

PROJECT TITLE : Fission produced radioactive noble gases treatment	CLASSIFICATION 5.3
SPONSORING COUNTRY : ITALY	ORGANISATION : UNIVERSITY OF PISA
DATE INITIATED : July, 1970 DATE COMPLETED : End of 1976	PROJECT LEADER : G. CURZIO

Description :

Research program:

- Charcoal beds adsorption characteristics determination in ideal work conditions.
- Evaluation of the dependence of the characteristics on the bed size and grain size.
- Evaluation of the effects of the decay heat, moisture, other impurities, pressure and temperature transients, etc.
- Charcoal filter tests.
- Comparative analysis of treatment devices.

Facilities:

- Charcoal bed testing facility
- Nuclear detector devices

Reference documents:

1. CURZIO G., GENTILI A.
Ritenzione di gas nobili su letti di carbone attivo.
Atti del XVI Congresso Nazionale dell'A.I.F.S.P.R., Firenze Settembre 1970. Firenze 1971.

2. CURZIO G., GENTILI A., MAINARDI C., PELLUNGRINI P.
Ritenzione dei gas nobili radioattivi prodotti per fissione negli impianti nucleari. Tip. Edit. Pisana, Pisa, 1972.

3. CURZIO G., GENTILI A.
Determinazione della densità granulare di materiali ad elevata porosità specifica. Il Giornale di Fisica, XIII, 4, 286, 1972.

4. CURZIO G., GENTILI A.
Noble gas adsorption characteristics of charcoal bed: Van Deemter's coefficient evaluation. An. Chem. 44, 8, 1544 (1972).

5. CURZIO G., GENTILI A.
Libération des gaz nobles par centres nucléaires: quelque remarques sur le fonctionnement des filtres de charbon de bois. VI^{e} Congres International de la Société Française de Radioprotection: "Tendances Nouvelles en Radioprotection". Bordeaux, 27-30 mars 1972, p. 233, Montrouge 1972.

6. CURZIO G., GENTILI A.
The effects of decay heat on adsorption characteristics of charcoal beds. Noble Gases Symposium, Las Vegas, 24-28 Sept. 1973.

7. CASTELLANI F., CURZIO G., GENTILI A.
Effects of moisture on Krypton adsorption characteristics of charcoal beds. To be published.

8. CURZIO G., GENTILI A.
Man-rem cost: a reverse evaluation.
To be published in "Environmental Studies".

The program has been sponsored by C.N.R. and will be sponsored by C.N.E.N.

PROJECT TITLE : Removal of krypton and xenon from the off-gas of Nuclear Plants.	CLASSIFICATION 5.3
SPONSORING COUNTRY : ITALY	ORGANISATION : CNEN
DATE INITIATED : 1975 DATE COMPLETED : 1978	PROJECT LEADER : G. BEONE

Description : The purpose of the project is to develop a process for the removal of radioactive kripton and xenon from gaseous effluents of nuclear reactors and reprocessing plants.

The selective-absorption process in liquid solvents is one of the promising methods under discussion from the point of view of the technological feasibility.

Macrocyclic polyethers, a new category of organic compounds capable to form inclusive compounds determining "sandwich structures", seem to be very promising for this purpose. The particular aim of the experimental research program is the choice of aromatic macrocyclic compounds, solid or soluble in non-polar organic solvents, and to verify the noble gases selective-absorption possibility.

Related projects: 5.3 (Pisa University)

Classification 5.4

Title 1 Konsekvenser af frigørelser af radioaktive stoffer til atmosfæren	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 Consequences of releases of radioactive substances to the atmosphere	Project leader: O. Walmod-Larsen
Initiated: 1972 Completed: Status: Progressing Last updating:	Scientists: S. Thykier-Nielsen P. Hedemann Jensen

1. General aim

Estimation of radiation doses to individuals and population, from releases of radioactive substances to the atmosphere under various environmental conditions.

2. Particular objectives

Development of models for calculation of

a. External gamma-doses from a cloud or plume of radioactive material.

b. Internal doses due to inhalation of radioactive material.

c. External gamma-doses from radioactive material peposited on the ground.

d. Population doses.

e. Global doses (both to individuals and population).

Furthermore the parameters in the models will be studied: Duration of release, atmospheric stability etc.

The consequences of different types of releases both accidental and normal will be investigated.

3. Experimental facilities and programme

None

4. Project status

A model has been developed for calculation of external gamma-doses, as well as internal doses due to inhalation of radioactive material. The Gaussion plume model is used assuming release from a point source.

Based on the model two computer programs has been written for the calculation of doses to individuals: GDOS (external gamma dosed) and INDOS (inhalation doses).

GDOS and INDOS is used as subroutines in the computer-program PLU48 which calculates population doses to a distance of 50 km from a point source.

5. Next steps

Further development as given in 2.

6. Relation with other projects

The models for calculation of external gamma doses and inhalation doses are described in the report:

a. "Modeller til beregning af eksterne gammadoser og inhalationsdoser fra frigørelser til atmosfæren af radioaktive stoffer", S. Thykier-Nielsen, Risø-M-1725.
   A comparison between GDOS and other models for calculation of external gamma doses are given in

b. "Sammenligning af matematiske modeller til beregning af eksterne gammadoser hidrørende fra radioaktivitetsfrigørelser til atmosfæren", Per Hedemann Jensen, Risø-M-1726.

8. Degree of availability

Available on an exchange basis.

Classification: 5.4	
Title 1 (Original Language): Untersuchungen zur ^{129}J-Radioökologie (PNS 4132 - II.6.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Investigation of ^{129}J-Radioecology	Project Leader: H. Schüttelkopf
Initiated (Date): anuary 1974 Status: continuing	Completed (Date): December 1976 Last Updating (Date): December 1975

1. General Aim

^{129}I will be the most important nuclide of radioiodine released from a nuclear fuel reprocessing plant. Due to its very long half-life of 1.6×10^7 years the behaviour of this nuclide on its critical pathways will be explored.

2. Particular Objectives

The ^{129}I radioecology can only be investigated in the environment of a nuclear reprocessing plant. The best analytical method to determine ^{129}I is the neutron activation analysis and therefore a research reactor is necessary for such investigations.

3.1 Experimental Facilities

The Karlsruhe Nuclear Reprocessing Plant WAK and the Karlsruhe Research Reactor FR2.

3.2 Research Program

The ^{129}I concentration in various samples in the environment of the WAK will be determined in order to get a complete picture of the ecological behaviour of this longlived radionuclide.

4. Project Status

4.1 Progress to date

The development of analytical methods was completed at the beginning of 1975. Waste air and waste water collection from WAK started in March 1975. Normally, air samples are collected weekly and water samples are collected and analyzed monthly. The waste

water cleaning systems include a distillation facility. The samples of the destilled waste water was cleaned in a sewage system and from this water monthly determinations of ^{129}I are performed too.
Starting in April 1975 ^{129}I was determined in a series of milk samples and in samples of thyroids of rabbits. During ten nuclear fuel dissolutions in WAK in May and June 1975 the distribution of ^{129}I in the facility was determined. The concentrations of ^{129}I were measured in feed solutions, in scrubber solutions and air. A second run of such determinations started in November 1975. In this run the determination of ^{129}I in medium active waste water was included.

4.2 Essential results
The determination of ^{129}I in the waste air of WAK shows that a large fraction of the ^{129}I-throughput is released with this waste air. Since a filter system has been installed in October 1975, the released ^{129}I-fraction has been reduced to below 1 % of the ^{129}I-throughput. The concentration of ^{129}I in waste water cleaned by distillation was about 10 pCi/l. The concentration in the waste water released from the Karlsruhe Nuclear Research Center was about 1 pCi/l. The results of the determination of ^{129}I in milk samples lie between 0.2 and 7 pCi/l. Between 1 and 10 pCi ^{129}I have been found in thyroids of rabbits. During the dissolution of nuclear fuels more than 99 % of ^{129}I are released from the feed solution. Scrubbing with water has no effect on the air concentration of ^{129}I. When scrubbing with 3 N NaOH between 50 % and 80 % of ^{129}I are retained. Since the sum of ^{129}I determined did not amount to 100 %, we have to suppose that between 10 and 50 % of ^{129}I are absorbed within the nuclear reprocessing plant. Part of it was released during the second half of 1975.

5. Next steps
The program of collection and analyses of air, water, milk and thyroid samples will be continued. A very extensive sampling of soil and plant material will be made in 1976 to determine the degree of ^{129}I-contamination of the environment of WAK. The concentration of ^{127}I in the air of the environment of the Karlsruhe Nuclear Research Center will be measured, starting at the end of 1976.

6. <u>Relation with other projects</u>

None

7. <u>Reference documents</u>

Report KFK 2045 (German)

H. Schüttelkopf, KFK-Nachrichten, 4, 1975

Semiannual reports in the series IRS-FORSCHUNGSBERICHTE

8. <u>Degree of availability</u>

Unrestricted distribution

Classification: 5.4	
Title 1 (Original Language): Theoretische und experimentelle Untersuchungen zur Ausbreitung radioaktiver Gase (PNS 4312 - II.6.4., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Theoretical and Experimental Investigations of the Atmospheric Dispersion of Radioactive Gases	**Project Leader:** Dr. Hübschmann, Dr. König, Schüttelkopf
Initiated (Date): January 1969 **Status:** continuing **Completed (Date):** December 1978 **Last Updating (Date):** December 1975	

1. General aim

Improvement of the knowledge about the atmospheric dispersion of single short-time radioactive emissions

2. Particular objectives

Tracer diffusion experiments are performed at the various stability categories, chemical tracer gases are emitted at heights between 50 and 200 m.

Theoretical evaluations of the measured data are performed in order to be able to predict the activity concentration pattern downwind of a power reactor after an accidental activity release.

The necessary meteorological instrumentation and the data recording equipment are defined for nuclear reactor stations.

3. Experimental facilities and program

A 200 m high meteorological tower is operated in the Karlsruhe Nuclear Research Center in order to collect comprehensive meteorological information in the lower atmospheric layer. Wind velocity and direction profiles as well as dry and dew point temperature profiles are measured across the tower height and stored on magnetic tape.

Diffusion experiments are carried out using tritiated water and /or halogenated hydrocarbons as tracers.

4. Project status

4.1 Progress to date

The meteorological data measurement at the now completely instrumented 200 m high tower, the data registration and record have been continued during the report period. Six diffusion experiments have been performed in 1975. Different chemical tracers have been emitted at heights of 60 and 100 m, In some experiments two tracers have been released simultaneously at the same level or at both levels. The gaschromatographic evaluation of the air samples has been improved and streamlined by an automatic injector and a peak integrator.

4.2 Essential results

The evaluation of the tracer experiments confirmed the validity of the results gained by previous experiments: the surface roughness increases the turbulence in the lower boundary layer and results in higher peak concentrations closer to the emitter. A set of diffusion parameter curves dependent on the surface roughness effect is developed.

Theoretical evaluations of measured meteorological data comprise the energy density spectrum of atmospheric turbulence in the frequency range $1{,}4 \cdot 10^{-4}$ to 0,125 Hz, definition of changes of the wind direction in contrast of the turbulent wind direction fluctuations, dynamic changes of atmospheric diffusion and its influence on the expected concentration pattern after an accidental activity release.

A ring of several dosemeters around a power reactor station has been designed, which measures the γ-radiation from an accidentally emitted radioactive off-gas plume. Such measurements may serve as an indicator of hazards to the population in the nearest residential areas.

The Reactor Safety Study WASH-1400 has been analysed in respect to the diffusion and dose calculations. This analysis serves as a base for applications to German power reactor sites.

5. Next steps

The diffusion experiments will be continued in the following years and will be extended to stable weather situations using automatic sampling stations.

6. Relation with other projects

Relation exists to several other projects within the "Projekt Nukleare Sicherheit" especially to PNS 4134 (Long-term Radiological Burden of the Environment Caused by an Accumulation of Nuclear Power Facilities).

7. Reference documents

Report KFK 2130 (1975) p. 300 (german with english abstracts)
Report KFK 2195 (1975) p. 366 (german with english abstracts)

8. Degree of availability

Unrestricted distribution

Classification: 5.4	
Title 1 (Original Language): Wanderung langlebiger Transurane im Boden und in geologischen Formationen (PNS 4412 - I.2.4, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Migration of Longlived Transuranium in the Soil and Geological Formations	Project Leader: Dr.Dippel Dr.Jakubick
Initiated (Date): ıuary 1973 Status: continuing	Completed (Date): December 1976 Last Updating (Date): December 1975

1. General Aim

For the case of a soil contamination by transuranium elements and for site evaluations of final depositories informations are required on behaviour and migration of plutonium.

2. Particular Objectives

In this respect the following points are of interest:

- applicability of tracer studies on real situations;
- influence of soil chemistry on the mobility of transuranium elements;
- behaviour of extremely low plutonium concentrations in natural waters.

3.2 Research Program

By field studies of Pu depth distribution,coming from the world wide fallout in natural soil,the mobility under natural conditions could be find out. By means of column studies in the laboratory the more specific aspects of migration may be estimated additionaly. From the comparision of both, the behaviour under real situations can be deduced.

4. Project Status

4.1 Progress to date

Due to a series of fundamental research activities on the interaction of transuranium elements with different parts of the environment a general view was attained in this field. They deal in detail with:

plutonium-sources; effect of environmental conditions, transportation mechanisms in the nature, plutonium reservoirs in the environments.

4.2 Essential Results

An evaluation of these problems from the point of view of geo-siences leads to the following results:

- plutonium in soil possesses the longest residence time,
- the knowledge of potential migration paths deserve a first order priority in siting of α-waste final depositories (case of Maxey Flats).
- the normally used relationsships may be applied for the evaluation of plutonium transfer in terrestrial ecosystems.

5. Next Steps

A field survey for locations which were not disturbed in the last 20 years having no other vegetation cover then grass and having different soil types will be started.

Simultaneously a line of glove boxes will be set up for the laboratory studies.

6. -

7. Reference documents

Report KFK 2130 (1975) p. 324 (german with english abstracts)
Report KFK 2195 (1975) p. 434 (german with english abstracts)

8. Degree of availability

Unrestricted distribution

Classification: 5.4	
Title 1 (Original Language): Langfristige radiologische Belastung durch eine Anhäufung kerntechnischer Anlagen (PNS 4134 - II.6.4., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Investigation of the Long-Term Radiological Environmental Impact Caused by an Accumulation of Nuclear Facilities	Project Leader: Dr. Bayer
Initiated (Date): ınuary 1973 — Completed (Date): June 1976 Status: continuing — Last Updating (Date): December 1975	

1. General aim

The main objective of this project is to assess and forecast the potential radiological burden that arises from an expanding nuclear industry, especially in regions where an accumulation of nuclear facilities is expected.

2. Particular objectives

It is planned to develop an extensive computer program to estimate :he release of radionuclides to the environment from the various facilities and the resultant radiological burden to the population in the Upper Rhine region.

This program will consider:

- the forecasted development of the nuclear industry during the next years
- the different release rates of nuclear power plants
- the transport of the released isotopes in the atmosphere and the hydrosphere
- the different "paths" of the radionuclides in the biosphere
- the different forms of food supplies, e.g. self-supply by farmers or central supply of urban population
- the population distribution and the age-dependent nourishment of the population.

A study is planned to learn how the different variables and their uncertainties will influence the results.

3. Experimental facilities and program

Not relevant.

4. Project status

4.1/2 Progress to date and essential results

After the incorporation of several additional atmospheric and hydrospheric transport effects the part of the computer program concerned by these effects is now nearly completed. Some subroutines concerning mainly the terrestrial transport and the uptake in plants have still to be improved. The relevant input data like site characteristics, release rates, deposition velocities etc. have been updated on the basis of newer reports and personal communications. Partial results of a few exposure pathways are already obtained.

5. Next steps

First complete results are expected in the near future. Sensitivity studies will be started in order to estimate the accuracy of the results.

6. Relation with other projects

Relation exists to several other project within the "Projekt Nukleare Sicherheit", especially to PNS 4312 (Theoretical and Experimental Investigations of the Atmospheric Dispersion of Radioactive Gases).

7. Reference documents

PNS - Semiannual Report 1975/1 (German with English abstracts)
Report KFK 2195 (1975) p. 129

PNS - Semiannual Report 1975/2 (German with English abstracts)
Report KFK (to be published)

D. Schiesser
Production and Release of Sr-89, Sr-90, Ru-103, Ru-106, Cs-134, Cs-135, Cs-137, Ce-141, and Ce-144 by Nuclear Power Plants and Reprocessing Plants and the Expected Radiological Impact untill the Year 2000 (German with English abstracts)
Report KFK 2153 (1975)

Reports in the series IRS-FORSCHUNGSBERICHTE

Semiannual reports:

Report period	Jan. - June 1975	IRS-F-26	
	July - Dec. 1975	IRS-F-28	(German)
Annual report	A 74	IRS-F-24	(English)

8. Degree of availability

Unrestricted distribution.

PROJECT TITLE : Quantitative evaluation of the release of radioactive substances in-to the environment	CLASSIFICATION : 5.4 - 5.5 - 5.6
SPONSORING COUNTRY : ITALY	ORGANISATION : C.N.E.N.
DATE INITIATED : January 1974 DATE COMPLETED : ——	PROJECT LEADER : Mario Dall'Aglio

Description :

In the vicinity of some nuclear plants has been carried out the study of the distribution and circulation of the natural isotopes of the radioactive elements which can be released by the nuclear plants, before the start of the industrial activity (e.g. Impianto "Fabbricazioni Nucleari", Bosco Marengo, Al).

After the beginning of the nuclear activity an environmental check can supply quantitative information about the pollution level due to the nuclear plants.

PROJECT TITLE : STUDIES ON THE CONTAMINATION OF THE SEA	CLASSIFICATION 5-4, 5-5,5-6.
SPONSORING COUNTRY : ITALY	ORGANISATION : C.N.E.N. and EURATOM
DATE INITIATED : 1957 DATE COMPLETED : (in progress)	PROJECT LEADER : Brondi Aldo

Description :

Studies of the factors wich influence the uptake, accumulation and loss of radioisotopes by different inorganic and organic constituents of the marine environment. The investigations are carried out on relevant radioecological and ecological factors in nature and under laboratory conditions.
Studies on thermal pollution from nuclear plants.

Classification 5.4	
Title 1 Gevolgen voor de omgeving van ongevallen bij kernenergiecentrales	**Country** The Netherlands **Organization** KEMA
Title 2 Environmental effects of nuclear power plants accidents	**Projectleader** B.Th. Eendebak

1. General aim

To analyse the risks of light water reactors on specific sites in the Netherlands.

2. Particular objectives

To study the effects of nuclear accidents as a function of site, population density, wheather conditions, etc.

3. Experimental facilities

Not applicable.

4. Project status

Computer code "MAKRO" is available.

5. Next steps

Not applicable.

6. Relation with other projects

This project was started by an order of the Minister of Economic Affairs to make a risk analysis of the fuel cycle in the Netherlands. This study was finished in June 1975.
See also the projects "Calculation of the quantities of radio-activity released as a result of a serious reactor accident" and "Failure analysis by application of event and fault trees".

7. Reference documents

Not available yet.

8. Degree of availability

Through the organization KEMA.

Classification 5.5	
Title 1 Bepaling van het aantal lekke splijtstofstaven en de kernpositie tijdens het reactorbedrijf	**Country** The Netherlands **Organization** KEMA
Title 2 Determination of the number of leaking fuel rods on the core position during operation	**Projectleader** J. Hoekstra

1. General aim

To reduce the wet-sipping time.

2. Particular objectives

- An increase of the off-gas activity combined with a one step control-rod movement indicates the position of ruptured fuel
- The release of certain fission-products during reactor start-up is a measure for the number failed fuel rods.

3. Experimental facilities

Dodewaard nuclear power plant.

4. Project status

Still in progress.

5. Next steps

Not applicable.

6. Relation with other projects

None.

7. Reference documents

None.

8. Degree of availability

Through the organization KEMA.

Classification 6 and 14

Title 1 Fejlanalyse af tekniske Systemer og vurdering af deres pålidelighed	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 Failure analysis of technical systems and evaluation of their reliability	Project leader D. Nielsen
Initiated 1970 Completed Status: In progress Last updating currently	Scientists J.R. Taylor O. Platz D. Nielsen

1. General aim

To develop analysis methods, or improve existing methods, for identification of potential faults that, if they occur, can lead to undesired consequences.

2. Particular objectives

1) To obtain a better knowledge of fault types and characteristics of fault propagation, and
2) to obtain experience with analytical modelling techniques for evaluation of reliability parameters.

3. Experimental facilities

4. Project status

4.1. Progress to date

a) The strategy and main steps of cause-consequence analysis based on the concept of critical events has been developed. The display format (used in connection with cause-consequence analysis and for documentation purposes) is the cause-consequence chart (CCC). The CCC is similar to the fault tree/event tree combination, but takes more details into consideration, and allows a systematic and complete build-up of accident trees.

6. FAULTS AND ACCIDENT COMBINATIONS

b) Statistical studies are carried out classifying incidents according to the number and types of failures involved in abnormal occurences reports of USAEC. Special emphasis is placed on classification of design errors according to cause, and methods for reducing design error frequency.

c) Development of analytical modelling techniques for evaluation of reliability parameters, taking testing and repair policies into account.

4.2. Essential results

The combined use of fault trees (cause chards) and event trees (consequence charts) has proven to be useful for coordinating expertise of specialists so that steps can be taken towards more meaningful risk analyses.

The study of the abnormal occurrences reports of USAEC indicates that design errors often play a significant role in process plant failure (the proportion of the failures which are considered as design errors is surprising high).

5. Next steps

Continuation of current works.

6. Relation with other projects

The project has relation to the Risø project concerning development of Monte Carlo computer programs for system reliability analysis.

7. Reference documents

D.S. Nielsen, "The Cause-Consequence Diagram Method as a Basis for Quantitative Accident Analysis", Report Risø-M-1374, 1971.

J.R. Taylor, "Sequential Effects in Failure Mode Analysis", Report Risø-M-1740, 1974.

8. Availability

Reports are available from the Library of The Danish Atomic Energy Commission, Risø, DK-4000 Roskilde, Denmark.

PROJECT TITLE : FAULT TREE ANALYSIS FOR NUCLEAR POWER PLANTS	CLASSIFICATION 6 - 14
SPONSORING COUNTRY : ITALY	ORGANISATION : CESNEF - POLITECNICO DI MILANO, MILANO, ITALY 20133
DATE INITIATED : 1973 DATE COMPLETED : 1975	PROJECT LEADER : S. GARRIBBA

Description :

Formal methods are established in order to achieve the determination of minimal cut sets from fault trees. Methods are based upon the segmentation of the tree, construction of minimal cut sets of subtrees and subsequent expansion into the minimal cut sets of the original tree. The method as compared with the traditional combinatorial techniques has the advantage of consenting (i) determination of the minimal cut sets of any order it may be required, (ii) hand calculations and interactive programming, (iii) direct or built-in sensitivity analyses.

7. CONTAINMENT AND ASSOCIATED SYSTEMS

Classification : 7.1.

Title 1 : Programme LOCA-2 : Evolution des pressions à courte terme dans les logettes de l'enceinte d'un réacteur PWR en cas d'un accident LOCA	Country BELGIUM
Title 2 : Programme LOCA-2 : A computer code to estimate the short term pressurization in the subcompartiments surrounding the primary system, in case of a loss of coolant accident	Organization TRACTIONEL
Initiated : June 72 Completed : April 73 (version 1) Last update : December 75 (version 4)	Project Leader . STUBBE

1. General aim: The layout of the subcompartments surrounding the primary system must be such as to ensure the integrity of the structural elements in case of a loss of coolant accident. Such layout must account for the short term overpressurization of the concrete structures surrounding the break location.

   The programme LOCA-2-V4 evaluates conservatively the pressure evolution in the subcompartments from which the forces acting on the concrete structures can be estimated.

2. Particular objectives : The blowdown mass and energy release data from blowdown codes (eg. Relap) are input to the code. The program can handle a maximum of 20 nodes with 40 interconnections. The calculation of the mass flow rates between compartments is for orifice flow (L/D < 50).

Three options are available for estimating the critical mass flow rates :

- Henry Fauske model (water, vapour + air)
- Moody model (for water - vapour only)
- Experimental model (cfr. code ZOCO V)

The effect of water entrainment can be simulated by specifying a water entrainment factor for each compartment, and a time evolution of these factors. This option enables one to estimate the depressurization rate of steam generators in case of a steamline break or feedwaterline break.

A code option is available to simulate fly-out panels and movable plug which may be used to provide proper shielding during normal operation and as pressure relieve panels in case of a LOCA. The programme contains thermodynamic correlations to deal with saturated and overheated vapour states.

4. Program Status :

1. Progress to date : the fourth version (LOCA-2-V4) is operational and is being used to guide the layout for the subcompartments of actual reactor systems.
   The short term pressure evolution of the different nodes is generated in graphical form for easy interpretation, and also stored on tape for easy access later.

2. Essential results : Extensive validation of the program results were performed on benchmark problems calculated with other codes (TMD, RELAP). The results indicate good agreement for constant values of the discharge coefficient. The value of the discharge coefficient, being tied to a certain model for the critical mass flow rate, reflects the conservatism of the results.

5. Next steps :

- The code LOCA-2-V4 is being modified to deal with flows in longer passages whereby inertial and frictional effects become important. By including a proper momentum conservation equation, the code may be used to simulate flow distribution around the reactor vessel and to estimate the reaction forces on the vessel in case of a primary break at the outlet nozzles.
- Extention from 20 to 50 nodes is envisaged to deal with the fine detail of the flow around objects, whereby differend nodes must be used to simulate the flow pattern.

7. Reference Documents :

D. BROSCHE : ZOCO V, a computer program for the calculation of time and space dependent pressure distribution in reactor containments.
Nuclear Engineering and Design. Vol. 23 (1972)

K.V. MOORE ET AL
Relap-IV : Computer program for transient thermohydraulic analysis. IDO-83401 (1973)

F.J. MOODY : Maximum Flow rate of a single Component, two-phase Mixture.
Transactions of the ASME - February 1965.

R.E. HENRY, H. FAUSKE : The two-phase Critical flow of a one-component Mixtures in Nozzles, orifices and short tubes.
Journal of heat transfer - May 1971.

8. Degree of availability : Contact TRACTIONEL - BRUSSELS.

Classification : 7.1

Title 1 : PROGRAMME LOCA-3:Evolution de la pression à long terme dans l'enceinte d'une centrale nucléaire suite à une rupture du circuit primaire.	Country : BELGIUM
Title 2 : PROGRAMME LOCA-3:Long term pressure evolution in the containment of nuclear power plants, following a loss of coolant accident.	Organization TRACTIONEL
Initiated : July 1974 Completed : July 1975 Last update : January 1976	Project Leader : E. STUBBE

1. General aim : To calculate the pressure history in the containment following a loss of coolant accident. The compute code LOCA-3-V4 unables one to.:

   a. estimate the maximum pressure for which the containment integrity must be assured ;

   b. estimate a conservatively low containment back pressure to evaluate the efficiency of the ECCS ;

   c. evaluate the efficiency of different safeguard systems (spray, ventilation) ;

   d. evaluate the temperature gradients in the containment structure in order to estimate the stress levels in the concrete.

2. Particular objectives : The code was developped for calculating containment loading for actual power plants. This requires the simulation of the various components that influence the pressure history such as :

- Detailed description of all passive heat sinks available. The code is dimensionned for a maximum of 10 structures, each of which can contain up to 150 nodes with a variable spacing.

  Four different options are built in to calculate the internal heat transfer coefficient in case of LOCA, two of which are the widely used Tagami-Ushida correlation for integrity and ECCS calculations.

- Simulation of the operational safeguard systems, such as spray and cooling coils.

- Evaluation of the sump water temperature resulting from such sources as the spray, the spill-over flow rates, condensing flux, and the flashing fraction that goes to the sump. This temperature is important to determine the stress in the sump concrete structure and to evaluate the depressurization rate in the recirculation phase.

- During the recirculation phase, a proper evaluation of the temperature of the component cooling water is necessary in order to estimate the heat absorption capacity of the cooling coils and the cooling capacity of the residual heat removal heat exchangers.

4. Project Status :

   1. Progress to date : The fourth version (LOCA-3-V4) is fully operational and provides graphical output for the most important parameters (pressure, temperatures).

2. Essential results : Extensive validation of the code was performed and the results show good agreement with results from other codes such as CONTEMPT, COMPATE, COCO and ZOCO V.

5. Next steps : The code LOCA-3-V4 is continually being updated to follow the evolution in the models used to conservatively estimate the pressure evolution (e.g. FLASH options, including "pressure flash" or "temperature flash").
Work is proceeding to include the treatment of the post-reflood phenomena with FROTH. As the input data are usually given for a fixed downstream pressure, the post reflood mass and energy release rates must be adjusted to actual downstream pressure in the containment.

7. Reference documents :

D. BROSCHE : ZOCO V, a computer code for the calculation of time-and-space dependent pressure distributions in reactor containments.

Nuclear Engineering and Design 23 (1972)

L. RICHARDSON ET AL.
"CONTEMPT", A computer programme for predicting the containment pressure-temperature response to a Loss-of-coolant accident. IDO-17220 (1967)

F. BORDELON ET AL.
Containment Pressure Analysis Code (COCO)
WCAP 8326

8. Degree of availability : CONTACT TRACTIONEL - BRUSSELS

Classification 7.1, 7.2

Title 1	COUNTRY Denmark
	SPONSOR DAEC Risø
	ORGANIZATION DAEC Risø
Title 2 CONTAC-II. A containment transient analysis code	Project leader: Aksel Olsen
Initiated: January 1969 Completed: Feb. 70 Status: In use Last updating: March 1971	Scientists: N. Kjær-Pedersen V.S. Pejtersen

1. General aim

Development of a computer code for the calculation of pressure- and temperature transients in a PS-containment during a loss-of-coolant accident.

2. Particular objectives

The present code, CONTAC II, is written in FORTRAN IV for the Burroughs 6700 computer, and it is a revised version of the CONTAC-code written for the IBM-7094 computer

It calculates the transient pressures and temperatures in the two compartments of a PS-containment following a rupture of the primary system. The blow down mass flow rate and corresponding enthalpy may either be supplied to the code as input functions of time or be calculated internally in a one-node vessel representation using Moody's model for critical two-phase flow.

The code uses one-node representation of drywell and wetwell. Core spray, drywell spray, wetwell spray and wetwell pool cooling as well as heat transfer from and to walls and structures are incorporated.

3. Experimental facilities and programme

The CONTAC II code was used during the planning and experimental phase of the Marviken containment experiments, and a comparison between experimental data and calculation has been undertaken.

4. Project status

A comparison between a number of calculations and corresponding data from the Bodega Bay, Humboldt Bay and Marviken experiments show an overprediction of the maximum drywell pressure ranging from 5 to 25 per cent.

5. Next steps

A new updated version, CONTAC III, including an improved vent flow model, is expected to be ready in early 1975.

6. Relation with other projects

A multiroom containment code, CONTAC-M, is now under development.

7. Reference documents

N. Kjær-Pedersen:
CONTAC II, A Containment Transient Analysis Code
Risø, RD-Memo nr. 47 (1972)

N. Kjær-Pedersen
SIMPLI II, A Simulation Program Using Implicit Intergration
Risø, RD-Memo nr. 46 (1972)

8. Degree of availability

Available

Classification 7.1, 7.2

Title 1	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 CONTAC-M: A Containment Multiroom Transient Analysis Code	Project leader: Aksel Olsen
Initiated: 1974 Completed: Status: Progressing Last updating:	Scientists: K.L. Thomsen P. Hansen

1. General aim

Development of a multinode containment computer code for pressure and temperature transients during a loss-of-coolant accident.

2. Particular objectives

The code, CONTAC-M, is under development in FORTRAN IV for the Burroughs 6700 computer.
It is to calculate transient pressures and temperatures in a multiroom full pressure or pressure suppression containment following a loss-of-coolant accident. The blow down mass flow rate and enthalpy may be calculated by the code or be supplied as input. Heat transfer to or from structures and walls is to be included as well as the effect from core spray, drywell spray, wetwell spray and wetwell pool cooling. Some effects from non-thermal equilibrium is to be included.

3. Experimental facilities and programme

The code is to be tested against available experimental data, as for instance the Marviken data.

4. Project status

The formulation of some physical equations has not yet been decided, while various possible mathematical solution methods are still being considered.

5. Next steps

Further work along the lines indicated.

6. Relation with other projects

The code is in some respects based on the experience gained during the Marviken experiments as well as experience with the CONTAC code.

7. Reference documents

No reports available yet.

8. Degree of availability

	Classification 7.1/10.4/11.2/11.4
Title 1 BEAMDYN - Program til lineær 3-dimensionel statisk og dynamisk analyse af bjælkestrukturer.	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2 BEAMDYN - Program for the linear 3-dimensional static and dynamic analysis of beam type structures.	Project leader Per Lundsager Scientists:
Initiated (date) November 1975 Status: progressing Completed: (date) 1976 Last updating (date) February 1976	Per Lundsager

1. General aim

To develope a computer program for linear 3-dimensional static and dynamic analysis of beam type structures.

2. Particular objectives

2.1. To make simplified analysis of Nuclear Structures, especially for aircraft impact and earthquake loadings.

2.2. To investigate the properties of various FEM techniques e.g. time integration methods, eigenvalue extraction methods etc.

3. Experimental facilities and programme

None

4. Project status

4.1. A program layout has been made which allows for easy change or extension of essential program features. This is done by extensive use of subprograms.

4.2. The static and direct time integration parts have been programmed and are being tested. The modal analysis part is being programmed.

4.3. At this stage standard IBM/SSP routines are used where possible. Newmark β time integration is used with $\beta = 0.25$ in order to obtain unconditional stability.

4.4. The program is coded for an IBM 370/165 computer.

5. Next steps

Completion and test of the program in its initial configuration. Further steps have not been decided yet.

6. Relations with other projects

No specific relations for the time being, but the project is of general relevance to Risø's activities in structural analysis.

7. Reference documents

None until now.

8. Degree of availability

Project information will be freely available.

CLASSIFICATION 7.1

TITLE 1	ETUDE DE LA CONDENSATION EN TRANSITOIRE DE VAPEUR D'EAU EN PRESENCE D'AIR SUR UNE PAROI PLANE.	COUNTRY FRANCE
		SPONSOR E.D.F./SEPTEN
		ORGANIZATION C.E.A.
TITLE 2	STUDY OF THE CONDENSATION ON A WALL OF A MIXTURE OF AIR STEAM AND DROPLETS IN TRANSIENT CONDITIONS.	Project Leader CEN.G/DTCE/STT
Initiated 1975	Completed 1976	Scientists M. SUREAU E.D.F. M. COURTAUD CEA
Status	Last updating: 20.01.75	

I - GENERAL AIM

Pression dans l'enceinte de confinement lors d'un accident de rupture du circuit primaire.

II - PARTICULAR OBJECTIVES

L'étude de l'évolution de la pression dans l'enceinte de confinement lors d'une rupture du circuit primaire, fait apparaître deux pics de pression, l'un lors de la phase de dépressurisation, l'autre résultant de l'évacuation de l'énergie stockée dans le coeur et le circuit par les systèmes d'injection de secours. Tous les facteurs qui interviennent dans cette mise en pression doivent être connus aussi bien que possible. En particulier, la pression est limitée d'une part par le système d'aspersion, d'autre part par la condensation de la vapeur éjectée sur les parois de l'enceinte et sur les structures en acier ou en béton qui y sont contenues. Le programme expérimental a pour but d'essayer d'améliorer la compréhension des phénomènes physiques pour mettre en évidence les paramètres influents et établir des modèles permettant de mieux approcher les valeurs des coefficients d'échange.

III - EXPERIMENTAL FACILITIES AND PROGRAMME

Il s'agit de condenser sur des plaques planes de différentes natures (béton, acier, béton + acier) et d'état de surface variable, de la vapeur d'eau mélangée à de l'air dans des proportions variables.

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IV - PROJECT STATUS

4.1 - Progress to date

Dispositif expérimental en cours d'étude depuis octobre 1974.

4.2 - Essential Results

Néant.

V - NEXT STEPS

Le dispositif expérimental pourrait être opérationnel en octobre 1975. Exploitation du dispositif jusqu'en fin 1976.

VI - RELATION WITH OTHER PROJECTS

Néant.

VII - REFERENCE DOCUMENTS

Néant.

VIII - DEGREE OF AVAILABILITY

Propriété E.D.F. et C.E.A.

	CLASSIFICATION 7.1
TITLE 1 CALCUL DE LA PRESSION DANS L'ENCEINTE DE CONFINEMENT LORS D'UNE RUPTURE DE TUYAUTERIE PRIMAIRE (PWR).	COUNTRY FRANCE SPONSOR E.D.F. SEPTEN ORGANIZATION E.D.F.
TITLE 2 CALCULATION OF THE PRESSURE IN THE CONTAINMENT, FOLLOWING A LOSS OF COOLANT ACCIDENT (PWR).	Project Leader E.D.F./SEPTEN/T
Initiated 1973 Completed 1974 Status Codes opérationnels Last updating : 20.01.75	Scientists H. ROUX.

I - GENERAL AIM

Dimensionnement de l'enceinte de confinement des réacteurs PWR.

II - PARTICULAR OBJECTIVES

L'analyse des sollicitations mécaniques et thermiques des enceintes de confinement des tranches nucléaires PWR en cas de rupture de tuyauterie primaire ou secondaire est effectuée à l'aide des codes développés à E.D.F. :

PAREO 5 : Calcul des évolutions de pression et de tuyauterie dans l'enceinte à court terme et à long terme en phase de recirculation.

CATEM 6 : Calcul des évolutions de température et pression à l'intérieur des divers compartiments de l'enceinte pendant les premiers instants suivant la rupture.

FORMON : Calcul des forces appliquées aux parois extérieures de la cuve en cas de rupture de tuyauterie dans le puits de cuve.

III - EXPERIMENTAL FACILITIES AND PROGRAMME

Néant.

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IV - PROJECT STATUS

4.1 - Progress to date

Codes élaborés.

4.2 - Essential Results

Calcul des enceintes des tranches PWR françaises.

V - NEXT STEPS

Améliorations éventuelles des corrélations utilisées dans les codes à l'aide de résultats expérimentaux.

VI - RELATION WITH OTHER PROJECTS

VII - REFERENCE DOCUMENTS

- note E.D.F./SEPTEN TC 74.46 "Equations de base et utilisation du code CATEM 6".
- note E.D.F./SEPTEN TC 74.48 "PAREO 5 : modèle mathématique pour l'étude de l'évolution des pressions et températures dans une enceinte de réacteurs PWR après PDR".

VIII - DEGREE OF AVAILABILITY

- rapports § 7 : disponibles.

La communication des codes ou leur utilisation pour les études appliquées doivent faire l'objet de contrats cas par cas.

CLASSIFICATION
7.1

TITLE 1	ECOULEMENT DES BROUILLARDS	COUNTRY FRANCE
		SPONSOR E.D.F. SEPTEN
		ORGANIZATION C.E.A.
TITLE 2	MIST FLOW	Project Leader CENG/DTCE/STT
Initiated	Non défini	Completed	Scientists M. SUREAU E.D.F. M. COURTAUD CEA
Status	Last updating : 20.01.75	

I - GENERAL AIM

Détermination des écoulements dans l'enceinte de confinement en cas d'accident de dépressurisation.

II - PARTICULAR OBJECTIVES

Ecoulement d'un brouillard à travers orifices jusqu'à un blocage sonique.

III - EXPERIMENTAL FACILITIES AND PROGRAMME

Utilisation de l'installation d'essais MOBY-DICK.

IV - PROJECT STATUS

4.1 - Progress to date

Etude du dispositif expérimental.

4.2 - Essential Results

Néant.

V - NEXT STEPS

Engagement du programme en 1975.

VI - RELATION WITH OTHER PROJECTS

Etude de l'évolution des fluides diphasiques (MOBY-DICK - IHPV). Codes de calcul de pression enceinte.

VII - REFERENCE DOCUMENTS

Néant.

VIII - DEGREE OF AVAILABILITY

Propriété E.D.F et C.E.A.

Classification : 7.1

Title 1 (original language) Thermal shock experimental program	Country : FRANCE Sponsor : EDF FRAMATOME Organization FRAMATOME
Title 2 (english) Thermal shock experimental program	**Project leader** : Mr. DOYEN **Scientists** :
Initiated (date): 1975 Completed (date): 1977 Status: PROGRESSING Last updating (date):	

1. GENERAL OBJECTIVE

The main objective of this program is to gain a validation of LEFM (Linear Elastic Fracture Mechaniely) applied to a reactor vessel core shell submitted to thermal stress similar to those occuring after a LOCA. At the end of the reactor life.

2. PARTICULAR OBJECTIVES

2.1 Validation of LEFM

2.2 Test a SA 508 cl 3 shell with subcritical cracks and critical cracks with liquid nitrogen in order to verify the crack ini--tiation and the crack arrest conditions.

3. PROJECT STATUS

Some preliminary tests have been carried out for determining the heat transfer coefficient between a steel and liquid nitrogen.

Based on experimental data, finite elements calculations have been performed for determining the crack size which is susceptible to propagate under thermal stress induced by the cool down of the internal surface of the test piece by liquid nitrogen. The test piece is ordered.

4. NEAR TERM PLANNING

A detailed fracture toughness characterization of the material will be done prior to perform thermal shock test in order to determine the crack sizes.

5. RELATION WITH OTHER PROJECTS

Complementary of HSST program (thermal shock tests) carried out by ERDA (USA).

6. AVAILABILITY OF "RESULTS"

Property of FRAMATOME.

Classification: 7.1	
Title 1 (Original Language): Untersuchung der Vorgänge in einem mehrfach unterteilten Containment beim Bruch einer Kühlmittelleitung wassergekühlter Reaktoren (RS 50 - I.1.4., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Battelle-Institut, Frankfurt/M.
Title 2 (english): Investigation of the Phenomena Occurring within a Multi-Compartment Containment after Rupture of the Primary Cooling Circuit in Water-Cooled Reactors	Project Leader: Dr. Kanzleiter
Initiated (Date): May 15, 1971 Completed (Date): December 31, 1976 Status: continuing Last Updating (Date): December 1975	

1. General Aim

The objective of the present research project is to simulate in large-scale experiments rupture of the primary cooling circuit within the containment of a water-cooled reactor and to investigate the phenomena occurring within the containment. The experimental results are to be compared with the results of model calculations and will finally serve to improve the computer codes.

2. Particular Objectives

The investigations have been started with model-scale experiments in an experimental PWR containment. The volumetric model scale is about 1 : 64 relative to the 1200 MW reactor plant Biblis A.

Later investigations will include BWR-specific experiments and experiments with highly stressed concrete structures.

3.1. Experimental Facilities

The experimental facility consists essentially of

- a special model containment,
- a model coolant circuit,
- measuring instruments,
- data collecting and processing systems.

The model containment, built in conventional reinforced concrete, has a diameter of 12 m, a height of 12.5 m, a capacity of 580 m^3 and is designed for an internal pressure of 6 bar. The interior is divided by concrete walls and removable partitions into several compartments, which are interconnected through openings of adjustable size. For the first experiments a PWR-configuration with nine compartments was installed. The model scale of the compartment volumes and the overflow areas is about 1 : 64 relative to the 1200-MW-PWR-plant Biblis A. By exchanging the removable partitions it is possible to modify the interior of the containment and to simulate different containment shapes.

The components of the model coolant circuit are two vessels (simulating reactor vessel and steam generator), pipes of 150 and 200 mm in diameter which connect the vessels with the site of rupture, and an auxiliary circulation system. The maximum pressure is 140 bar at a temperature of 300°C. Rupture is effected by rupture disks.

The measuring instruments used at present contain about 200 measuring channels for pressure, differential pressure, temperature, density, mass flow, jet force and water level.

The signals from a maximum of 120 channels can be recorded up to a threshold frequency of 5 kHz: the other signals are recorded at a scanning rate of 1 Hz. The data are stored on analogue and digital magnetic tapes and are processed by a process computer. Output is effected by means of display, hard-copy unit and digital magnetic tape.

3.2. Research Program

A total of about 30 blowdown experiments are envisaged. The first 16 experiments planned simulate PWR primary-circuit ruptures. The conditions for the other experiments, including experiments with BWR-specific and PWR-specific steam line breaks, jet force experiments leading to an extremely high load on special concrete structures, and additional PWR primary-circuit breaks are not yet completely specified.

4. Projekt Status

4.1. Progress to Date

The experimental facility has been built and equipped for PWR experiments. To date it has been used for four trial runs and twelve experiments with single- and double-end pipe ruptures of 100 mm diameter in a steam generator compartment, the nozzle compartment, the pressurizer compartment, the reactor cavity and the refuelling cavity.

4.2. Essential Results

The initial conditions for the pressurized water in the model coolant circuit before rupture were 120 to 140 bar and 287 to 295°C. About 0.1 s after rupture the flow rate at the site of rupture reached its maximum of about 50,000 kg/m^{2}s. Maximum jet forces of approximately 140 x 10^3 N (diameter of the leak 100 mm) were measured 0.1 s after rupture over a period of about 0.1 s. From the compartment where the rupture took place, a water-steam-air mixture streamed through openings into the other compartments of the containment. Differential pressures between compartments were measured with maxima of up to a few bar 0.15 s to 0.5 s after rupture, depending on the compartments and transducers concerned. Approximately 30 to 40 s after rupture the blowdown finished and the pressure in the containment reached about 4 to 5 bar.

Model calculations in connection with these experiments are being carried out by university institutes, regulatory authorities and plant manufacturers. These model calculations lead in general to conservative results, but it was also found that it is not possible to achieve full agreement between experimental and theoretical results by merely varying the constants α_D and ξ_W (α_D = discharge coefficient, ξ_W = water carry-over coefficient).

5. Next Steps

The PWR experiments will be continued with double-end pipe ruptures in the nozzle compartment. Then experiments with BWR- and PWR-specific steam line breaks and jet force experiments with special concrete structures will follow.

6. Relation to other projects

RS 0016B Investigation into the Phenomena Involved in the Depressurisation of Water-Cooled Reactors. Experiments Using a Steel Vessel 11.2 m in Height with Internals. Battelle-Institut e.V.,Frankfurt

RS 0123A Safety Investigations performed at the decommissioned HDR plant

7. Reference Documents

(1-4) Quarterly reports in the series "IRS-Forschungsberichte" (in German)

IRS-F-25	January to March 1975
IRS-F-26	April to June 1975
IRS-F-27	July to September 1975
IRS-F-28	October to December 1975

(5) IRS-F-24 "Research Reports (Anual Reports)" (in English)

(6) BF-RS 50-23-1 "Beanspruchungsmessungen an einem Modell-Containment aus Stahlbeton", August 1975

(7) BF-RS 50-62-2 "Druck- und Temperaturverläufe bei Blowdown-Versuchen in einem Modell-Containment", October 1975

(8) BF-RS 50-62-3 "Experimental Investigations of Pressure and Temperature Loads on a Containment after a Loss-of-Coolant Accident", October 1975

(9) BF-RS 50-31-2 "Geometrie der Versuchsanlage I (Zeichnungen)" July 1975

(10) BF-RS 50-32-C4 "Vorläufiger Versuchsbericht C4", August 1975

(11) BF-RS 50-32-C5 "Vorläufiger Versuchsbericht C5", March 1975

(12) BF-RS 50-32-C6 "Vorläufiger Versuchsbericht C6", March 1975

(13) BF-RS 50-32-C8 "Vorläufiger Versuchsbericht C8", August 1975

(14) BF-RS 50-33-2 "Versuchsergebnisse vom Druckentlastungsvorgang im Primärsystem bei Einfachendbrüchen mit langer Rohrleitung (Versuche C1 bis C3) - Vorläufiger Bericht -", December 1974/April 1975

(15) BF-RS 50-33-3 "Strahlkräfte an der Bruchstelle - Vorläufiger Bericht"
April 1975

8. Degree of Availability:

Documents are available through

Institut für Reaktorsicherheit- Forschungsbetreuung
Glockengasse 2, D-5000 Köln 1

Documents (9) to (15) are available only under special agreement.

Classification: 7.1	
Title 1 (Original Language): Untersuchungen über die Auswirkungen des Ausströmens von Dampf- und Dampf-Wasser-Gemischen aus Rohrleitungslecks (RS 93 - I.1.4, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Reaction and Impingement Forces on Components and Structures Caused by a Two Phase Jet Discharged at Primary Pipe Breaks	Project Leader: Dr. Riedle
Initiated (Date): January 1973 **Status:** Completed	
Completed (Date): October 1975 **Last Updating (Date):** December 1975	

General Aim and Particular Objectives

In the safety analysis of a loss of coolant accident it is necessary to calculate the mechanical loading caused by the pressurized water discharging through the break. The discharging two-phase jet applies thrust on the supporting structures of the primary loop components and pipe restraints. The neighbouring walls and structures have to withstand the impact pressure caused by the impingement of the jet.

Experimental Facilities and Research Program

This program deals with quasi-steady state experiments which were carried out to investigate on a reduced scale a guillotine break and a slot break. During the guillotine break tests the discharging jet is directed on to a baffle plate, during the slot break tests two jets are directed against each other and are diverted by the simulated bottom of the ruptured pipe. The discharged fluid is saturated water; in several cases experiments with one-phase fluid were carried out.

The test parameter were varied in the following way:

discharge diameter D,	guillotine break:	5, 10, 25, 50, 65 mm
discharge diameter D,	slot break:	5, 10 mm
distance pipe outlet-baffle plate A	guillotine break:	0,25 to 10 D
distance between the pipe outlets A	slot break:	0,20 to 14 D
stagnation pressure:		10 to 100 bars

The instrumentation supplies following informations:

critical discharge rate,
thermohydraulic state at the pipe outlet,
pressure distribution on the baffle plate,

the contour of discharging two-phase jets as well as thrust forces.

Project Status / Progress to Date / Essential Results

The investigations concerning the critical flow of saturated pressurized water have shown the following results:

1. The critical flow rate measured agrees well with theoretical calculations of diverse models from literature. The hydraulic resistance of the pipe, the diameter of the circular outlet cross section and, connected with it, the flow regime are of particular importance. For small outlet dimensions the calculation with the Moody model (flow regime: annular flow) seems to be successful, while for large outlet a model with homogeneous flow conditions should be preferred (e. g. that by Linzer). These facts were confirmed by the measured results.

2. The thermohydraulic state at the outlet cannot be determined exactly by the measurement of the outlet pressure and temperature and by mathematical estimation of the steam content. These data do not allow exact conclusions about flow regimes actually occuring, boiling delay and thermodynamic non-equilibrium. In particular the measurement of the pressure a few millimetres upstream of the

outlet plane is subject to considerable errors.

3. The pressure distributions resulting from the impact of the two-phase jets on the baffle plates are similar to normal distribution curves. For all diameters tested it was possible to show the radial pressure distribution as a two-parameter-function $f\ (\frac{r}{D}, \frac{A}{D})$ if the local pressure on the plate was related to the maximal value in the jet centre. They are similar to corresponding normal distribution curves at the impact of one-phase jets - only the parameter distance between outlet and the baffle plate differs by factor of 6 to 10. The radial expansion of two-phase jets, disturbed by the baffle plates, and that of undisturbed two-phase jets are almost the same - they differ by factor of 5 from the corresponding one-phase jets. Finally, it must be mentioned, that the two-phase jet has no conical jet core and that it loses its effectiveness after a distance of about 5 D. In the case of a one-phase jet the core has a length of 5 D and a total range of 30 to 40 D.

4. As to the jet load and thus to the design of the structures subject to the impact of two-phase jets it is necessary to mention the differences between the individual break shapes. In the case of the guillotine break a thrust force acts on the outlet pipe and an equal and opposite force acts on the baffle plate. For a slot break the jet causes both thrust forces and forces on the remaining metal joint. The thrust forces are the same for both break types. For the outlet of saturated pressurized water with large outlet diameters they can be sufficiently described by an impulse balance at the outlet, assuming homogeneous flow distribution.

The conditions for annular flow must be taken into account for small outlet diameters. The method of considering the force effect by means of the thrust factor, already introduced in literature is especially suitable to show the influence of parameters on the forces. Beside the value of the pipe resistance coefficient, these are the stagnation pressure, the outlet diameter and therefore the flow regime. The radial forces appearing in slot break geometries reach their maximum value when the inlet cross sections are equal to the break outlet cross section. The radial forces compared with the axial thrust forces show different results for the geometries investigated. For tests on geometry I the radial force is always smaller than twice the axial thrust force. When diverting two jets by 90 ^{o} without loss it is possible to reach maximal $R_L / (2 R_Q) = 1$ for the ratio of radial force to the sum of the axial thrust forces. For tests on geometry II, however, the radial forces assume values that can be higher than twice the axial thrust force, $R_L / (2 R_Q) > 1$.

This is due to the fact that - in case of outlet cross section and inlet cross sections being equal - at geometry I only low pressure due to the expansion in all directions perpendicular to the pipe bottom becomes effective. In the case of geometry II a considerable additional force (due to the pressure which at A/D = 0,61 is larger than the critical pressure) exists on a surface, which for this kind of break simulation can become larger than the remaining metal joint.

Next Steps

Work has been completed on this project.

Relation with Other Projects

RS 16 Investigation of the Phenomena Involved in the Depressurisation of Water-Cooled Reactors

RS 50 Investigation of the Phenomena Occurring within a Multicompartment Containment after Rupture of the Primary Cooling Circuit in Water-Cooled Reactors

Reference Documents/Degree of Availability

W. Kastner, R. Eichler, K. Riedle
Experimentelle Untersuchungen zu Kräften kritischer Zweiphasenstrahlen bei Quer- und Längsrissen von Rohrleitungen
Abschlußbericht zum Förderungsvorhaben BMFT RS 93,
KWU-Erlangen, (Oktober 1975)

Company Confidential

Classification: 7.1	
Title 1 (Original Language): Sicherheitsexperimente im Kernkraftwerk Marviken, Schweden (RS 33 - I.1.4., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GKSS, Geesthacht
Title 2 (english): Joint Reactor Safety Experiments in the Power Station of Marviken, Sweden	**Project Leader:** Franke
Initiated (Date): December 1971 **Completed (Date):** March 1975 **Status:** finished **Last Updating (Date):** December 1975	

1. General Aim

The Marviken experiments are carried out to investigate the behaviour of BWR containment with pressure-suppression system after a loss of coolant accident. The results are used to check analytical models for the design of containments and pressure-suppression systems.

The following institutions and organisations are participating in the program:

- Aktiebolaget Atomenergi, Sweden
- Atomenergikommissionen, Denmark
- Ministry of Trade and Industry, Finnland
- Institut for Atomenergi, Norway
- Gesellschaft für Kernenergieverwertung in Schiffbau und Schiffahrt mbH, Federal Republic of Germany.

In March 1973 the United States Atomic Energy Commission (US AEC), in May 1974 the Japan Atomic Energy Research Institute (JAERI) and in June 1974 the Comissarat à l'Energie Atomique (CEA) joint the project.

2. Particular Objectives

There were 20 blowdown experiments scheduled, 16 of which have been preliminarily defined in regard to the initial parameters.

The experiments were carried out under operational conditions, for which the reactor originally had been laid out (50 bar, 264 ^oK).Steam and feed water pipe ruptures with fracture areas up to 710 or 280 cm^2 respectively, were scheduled. For the investigation of the pool behaviour its initial temperature was varied between 25 and 60 ^oC.

3. Experimental Facilities and Program

See Report IRS-F 24 (1974) p. 136

4. Project Status

4.1 Progress up to Date

The preparation of the containment for the tests and the conduction of preliminary experiments to test the facility, including the instrumentation and the data recording system, were carried out in the phase 1.12.1971 - 31.9.1972. In the range 1.10.1972 - 14.5.1973 16 blow-downs were performed. After termination of the experimental phase the evaluation and reporting of the results were started. Till to the end of March 1975 all reports were completed and the project terminated.

4.2 Essential Results

4.2.1. Containment reponse to loss of coolant accident

4.2.1.1. Basis for the computer code testing

It may first be stated that the experiments have provided information which can be used for the verification of computer codes for calculating the pressure build-up and temperature rise in a pressure-suppression containment, which was actually the main objective fo the series of blowdown experiments.
Dry containment analytical models may be tested against the short-term pressure transients in the drywell compartment, before the onset of venting to the wetwell compartment.

4.2.1.2. The project computations

The limited testing of computer codes for containment response analysis whithin the project has shown that these can reproduce the containment pressure and temperature conditions fairly well. The current opinion that computer codes give a conservative prediction of the containment pressure response was further strengthened, although it is apparent that nonconservative features exist in the analytical models. Consequently, there are some points in the models which can be improved so as to yield more realistic calculations and thereby more reliable determinations of the safety margins.

4.2.1.3. The Tests as demonstrations

The sixteen simulated blowdown tests, carried out under different conditions, have repeatedly demonstrated in full scale the ability of a water pool based pressure-suppression containment to receive large quantities of mass and energy at varying rates, condense the steam and control the containment pressurization, as intended. The tests have, in that sense, confirmed the applicability of pressure-suppression systems,as previously stated on the basis of e.g. the small-scale Humboldt Bay and Bodega Bay tests.

The occurrence of minor damage, which in a real accident situation could have been a potential risk, indicated the importance of carefully ensuring the integrity of the containment and vital safety systems; in this context it is necessary to point out that the assessment and licensing of the Marviken containment systems had never been completed.

4.2.1.4. The presence of pressure oscillations

A specific phenomenon demonstrated in these tests was the presence of pressure oscillations of significant amplitudes, not only in the wetwell water pool but also in the vent pipe system and in the drywell compartment.

Since the oscillations impose additional loadings on structures in the containment, it seems necessary to gain some insight into their nature or cause. This does not seem possible, however, on the basis of the limited information from this series of tests; the data acquisition system was not designed for the recording of pressure oscillations.

4.2.2. Jodine and Xenon experiments

4.2.2.1. Leak testing of the containment structures

The absolute method and a radioactive tracer isotope method using ^{133}Xe were applied in two leakage rate tests under static conditions. From the investigation it may be concluded that:

The leakage of ^{133}Xe was somewhat higher than that of dry air, the difference being 30 - 50 %.
The absolute method is to be preferred for a proof leakage rate test. This test is carried out infrequently, so that the long time required for the test is not a serious drawback. The isotope method however, is very promising, and might well be applied for intermittent controls of containment tightness as an alternative to the absolute method.
The application of the isotope method is limited to cases where a vessel can be completely surrounded by a limited volume. It must be possible to take representative samples of this volume, preferably by ventilation and subsequent sampling of the contained air.

4.2.2.2. Containment leakage during accident conditions

The leakage rate in runs with elemental iodine is apparently c. 500 times smaller than in runs with methyl iodide. The reason for this is that iodine is removed from the air, probably by deposition on the containment walls and washout. Another contributing reason may be that leakage paths for elemental iodine are blocked after blowdown. A small fraction of the iodine is converted into organic iodine compounds and this fraction may be responsible for the leakage.

The leakage rate of methyl iodide is 2 - 3 times lower than that of xenon if the activities are injected before or during the blowdown. This difference ist, in part, accounted for by removal of the iodine from the air space as a result of retentions in the wetwell pool. Spray cooling also removes some methyl iodide.

The leakage rate of xenon determined under static conditions was about five times higher after all the blowdown runs than before the runs. The reason for this may be an increased leakage through the structure and/or the conduits through the walls. New conduits were introduced during the experimental period. No tendency was found towards an increasing leakage rate as the number of experiments performed increased.

The leakage rate under blowdown conditions was higher than that determined under static conditions both before and after all blowdown runs, the leakage rate of xenon increased with time in parallel with a decrease in the total pressure in the containment. The reason for the increase in the leakage rate is not clear. It may be connected with the effects of condensation and evaporation of a water film in the leakage paths as steam and heat penetrate into the containment walls, and/or a decreased tightness of the containment structure. A timedependent distribution of xenon in the drywell atmosphere might also contribute to an increasing leakage.

4.2.3. Component Tests

The component tests were carried out in order to find out how the different components stood up to the atmosphere under blowdown conditions Most of the components were left inside the PS containment between the blowdowns, and for some days before and several days after each blowdown they were exposed to an atmosphere with a high humidity and an increased temperature.

In addition to certain components which were selected and installed for testing, some of the components originally installed were also included in these component tests. These tests involved the registration of signals from measuring devices and the operation of valve actuators during the blowdowns.

The following groups of components were tested:
Electrical equipment(Measuring devices, Switches, Installation material, Electric motos, Electric valve actuators)
Mechanical equipment (Pneumatic valve actuators, Hydraulic valve actuators, Heat insulation)
Building materials (Surfacers and paints for concrete).

The aim of the tests was not to obtain quantitative results. The tests were principally of the go/no-go type, and the investigations and data collecting were carried out accordingly. In some cases it was possible to obtain quantitative information from the reported results; however, the intention was not to evaluate in the sense that the cause of a malfunction would be discussed in detail. For more information about the results, reference should be made to the complete test reports.

5. Next Steps

In order to get more detailed information about the pressure oscillations, a new project with specially instrumentated experiments is founded (RS 33a).

6. Relation with other Projects

RS 50: Investigation of the phenomena occuring with a multicompartment containment after rupture of the primary cooling circuit in water-cooled reactors.
Batelle Institute, Frankfurt, 1971 till 1976.

RS 33a: Joint Reactor Safety Experiments in the Power Station of Marviken, Sweden Part 2 - Oscillationexperiments.
Gesellschaft für Kernenergieverwertung in Schiffbau und Schifffahrt mbH, Geesthacht, 1975 till 1977.

7. Reference Documents

1. Reactor Safety Experiments in the PS-Containment of the Marviken Power Station, H.G. Thorén et al., AB Atomenergi, Studswik/Sweden July 1971. (written in english)

2. Annual Report A 72 IRS-F-12 written in english
   Annual Report A 73 IRS-F-18 written in english
   Annual Report A 74 IRS-F-24 written in english

3. Intermediate Report December 1971 till June 1973 IRS-F-16 written in german

4. Full-Scale Containment Experiments performed in the Marviken Power Station, H.G. Thorén et al., AB Atomenergi, Studsvik/Sweden 1973. (written in english)

5. About 15 MXA-Reports from the project, table 1, list of MXA-reports, in IRS-F-24 p. 140-141. (written in english)

8. Degree of Availability

References 1 - 4 are free available. Reference 5 ist available only to the contracting parties and free available in about 1 year.

Classification: 7.1	
Title 1 (Original Language): Sicherheitsexperimente im Kernkraftwerk Marviken, Schweden. Teil 2 - Schwingungsuntersuchungen (RS 33 A - I.1.4., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GKSS, Geesthacht
Title 2 (english): Joint Reactor Safety Experiments in the Power Station of Marviken, Sweden. Part II - Oscillation Experiments	**Project Leader:** W. Franke
Initiated (Date): February 1975 **Status:** continuing	**Completed (Date):** March 1977 **Last Updating (Date):** December 1975

1. General Aim

In February 1975 a second agreement for full scale containment experiments in the Marviken power plant was signed between Denmark, Norway, Germany, Finland, Sweden, France, Japan and USA. The containment response tests are mainly carried out to investigate the pressure oscillation behaviour of the containment with pressure-suppression system after a loss of coolant accident and to make measurements of parameters thought to be important for the understanding of these phenomena. After the preparation phase (1.2.75 - 30.1.76) the Blowdowns shall be carried out during the experimental phase (1.2.76 - 30.9.76). The evaluation and reporting stage shall be started on 1.10.1976.

2. Particular Objectives

There were 8 Blowdown experiments (Nr. 17 till 24) scheduled, all of them have been preliminarily defined in regard to the initial parameters.

The experiments were carried out under operational conditions, for which the reactor originally had been laid out (50 bar, 264 ^{o}K). Pipe ruptures with fracture areas up to 900 cm^2 and Flowrates up to 2000 kg/sec were scheduled. For the investigation of the pool behaviour the submergence depth (0,5 - 2,8 m) and ventarea (4 - 2 m^2) is varied.

3.1 Experimental Facilities

The experimental facility is the decommissioned reactor of Marviken, Sweden. This facility has been designed and constructed as a 140 MWe direct circuit boiling heavy water reactor. Soon after completion it had been decommissioned and offers the opportunity to conduct full scale safety experiments.

3.2 Research Program

The experimental program i.e. the choice of testparameters, is done due to the following reasons:

Blowdown 17. This is a close simulation of Blowdown 10 in the earlier series which had indicated significant pressure oscillations. The test should reproduce those conditions and thereby provide a data base for instrumentation check-out.

Blowdown 18. This is similar to Blowdown 17 but with a reduced vent flow area. The purpose is to get an increased vent mass flux. The blocking of the vent pipes should be arranged so as to provide a simple symmetrical and compact pattern. The arrangements will be compatible with the new pool geometry in Blowdown 21.

Blowdown 19. This test will utilize prepurging of the drywell air in order to investigate the influence on oscillations of noncondensibles in the vent flow.

Blowdown 20. This test will be run with a reduced submergence depth associated with a reduced pool water mass. A significant change of submergence depth is wanted and for geometrical reasons can only be obtained by reducing the pool water volume. The lower temperature indicated at the start of the prepurging is meant to compensate for the decreased pool mass.

Blowdown 21. This test will be carried out with the pool volume reduced to about three quarter of the normal volume. The reduction will be achieved by introducing an internal structure forming a new pool around the unblocked vent pipes and thus reducing the pool cross section. The smaller pool mass will result in an increased temperature rise. The reduced pool mass may also have an effect on the pressure oscillations.

Blowdown 22. This test should show the effect of air on pressure oscillations as compared to Blowdown 21. The dynamic loads on thestructures above the pool can be compard to the results from Blowdowns 17 and 18.

Blowdowns 23 and 24. These tests intended to show effects on the pressure oscillations due to changes in vent system geometry. The specific changes should be determined on the basis of further theoretical studies.

4. Project Status

4.1. Progress up to Date

The preparation of the containment for the tests and the conduction of preliminary experiments to test the facility, including the instrumentation and the data recording system, were carried out in the phase 1.2.1975 - 31.12.1975.

4.2. Essential Results

As to the end of the reported period no experiments were carried out at this place the instrumentation installed (during the year 1975) shall be described.

The main part of measurements consists of the pressure, differential pressure and temperature measurements. The semistatic measurements of these quantities will be recorded by a VARIAN 620/L process computer on digital tape while the so-called dynamic measurements will be recorded by a Pulse Code Modulation (PCM)-system. In addition, the pressure in the pressure vessel, drywell and wetwell and the differential pressure in the pressure vessel - one channel for each - are recorded by pen-recorders. These latter measurements are primarily for controlling purposes in the control room.

The water levels in the wetwell pool and vent pipes are measured with level probes of spark plug type. The signals from these measurements are recorded by PCM-system. In measuring the water level in the pressure vessel a special type of level probes are used and the recording is made on a light beam recorder.

The impact load in the wetwell will be investigated by using nine accelerometers and three strain gauges all of which are recorded on PCM-system.

In the wetwell the behaviour of the water is observed via two TV-cameras. At the same time the sound in the wetwell is recorded by a microphone. Another microphone is placed near the room 122 (break room) in the drywell.

The flow related measurements in the break pipe and the downcomers will be performed using the radio tracer method and the infrared-absorption-technique.

5. Next Steps

It is expected to start the experimental phase in February 1976 and carry out all 8 experiments till end of September 1976.

6. Relation with other Projects

RS 50: Investigation of the phenomena occuring with a multicompartment containment after rupture of the primary cooling circuit in water-cooled reactors.
Battelle Institute, Frankfurt, 1971 till 1977.

7. Reference Documents

Commentary on the proposal for The Marviken II containment response tests, AB-Atomenergi Studsvik 1974.

8. Degree of Availability

The above mentioned reference is available only to the contracting parties.

	$p_{R,0}$ /bar/			ϑ_{Pool} /K/			$\dot{m}/A$ /kg m^{-2} s^{-1}/		
Parameter	2,00	2,25	2,50	30	40	50	75	107	125
ν /Hz/	4,44	4,48	4,50	4,50	4,35	4,32	4,25	4,50	4,71
$2\,\Delta p_1$ /bar/	0,35	0,30	0,25	0,25	0,20	0,15	0,30	0,25	0,10
$2\,\Delta p_2$ /bar/	0,80	0,60	0,45	0,45	0,30	0,25	0,50	0,45	0,20
$\overline{R}$ /m/	0,39	0,39	0,40	0,40	0,41	0,42	0,34	0,40	0,41
$2\Delta R$ /m/	0,23	0,21	0,11	0,11	0,09	0,08	0,19	0,11	0,06

Table 1: Parameter variations for the MARVIKEN geometry.

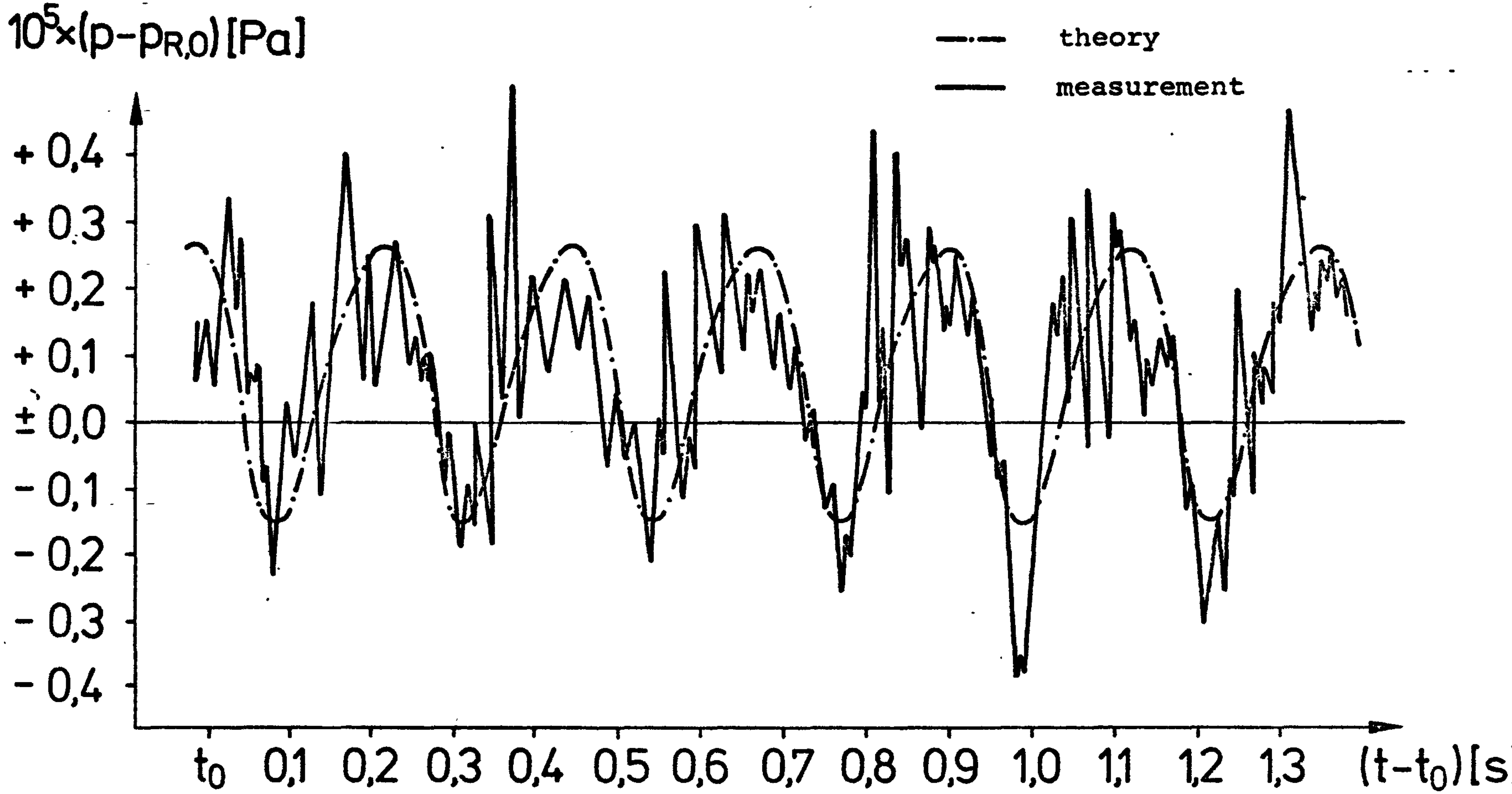

Fig. 1: Comparison between the theoretical and experimental pressure transients in the wetwell during Marviken-13 blowdown.

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Classification: 7.1	
Title 1 (Original Language): Entwicklung und Verifizierung von Rechenprogrammen zur Beschreibung von Containment-Problemen (ATT 085 A - I.1.4!, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (english): Development and Verification of Codes Simulating Containment Problems	Project Leader: Dr. H. Karwat
Initiated (Date): Completed (Date):	
Status: continuing Last Updating (Date): December 1975	

1. General Aim

General Aim of this work is the application of existing codes to verify results of relevant experiments and to improve existing codes which simulate effects associated with the pressurization of containment systems and its long time behaviour with respect to pressure, temperature and hydrogen distribution of the loss-of-coolant accident.

2. Particular Objectives

During the reporting period the work concentrated on pre- and recalculations of containment experiments carried out at the Battelle-Institute in Frankfurt, the improvement of codes simulating pressure suppression system effects and on a code simulating hydrogen distribution within containment systems resulting from radiolytic decomposition of water during the emergency core cooling process.

3. Experimental Facilities and Program

Experimental work carried out in connection with the pre- and recalculation mentioned above is reported under the headline "Investigation of the phenomena occuring within a multi-compartment containment after a rupture of the primary cooling circuit in water-cooled reactors" (RS 50) by the Battelle-Institute, Frankfurt.

4. Project Status

4.1 Progress to Date

An improved version of the containment code ZOCO has been generated which has more flexibility with respect to the simulation of heat transfer processes between the fluid and the containment structure. The necessity for the improvement was given by deviations observed in the first series of the recalculation of blowdown-experiments which have a surface-to-volume ratio of approximately 2.4 compared to the ratio of a typical DWR of 0.4. The new version ZOCO VI is described in detail in /1/.

Work concerning the simulation of separate effects of pressure suppression systems resulted in the generation of a code describing harmonic oscillations associated with the condensation of steam in water pools. These oscillations were observed in the Marviken experiments (RS 33) as well as during GKM-experiments (RS 78a). The code KSWING has been successfully applied to verify these oscillations with respect to its amplitudes and frequencies. The model is based on a selective description of heat conduction- and condensation processes at the surface of a steam bubble.

KSWING may be considered as an important part of a new comprehensive model still under development which simulates in detail a complete pressure suppression process.

In connection with the experimental program carried out in the Battelle-Institute Frankfurt (RS 50) the analytical verification of the experimental results has made progress.

Six tests have been subject to intensive recalculations using the code ZOCO VI. The comparison between analysis and experiment allowed some preliminary conclusions with respect to the applicability of codes for the prediction of pressure differences, the neccessary improvement in measurement technique in future experimental programs and with respect to future code development work.

A first version of a code simulating hydrogen distribution within a containment system has been finished and is now in use to perform some parametric studies for the determination of important parameters influencing the calculated results. In connection with this work a proposal has been elaborated for a relevant experiment to back-up the analytical work.

4.2 Essential Results

The analytical verification of the containment experiments showed the capability of the existing codes to predict the fluiddynamic behaviour of full pressure containment systems under accident conditions. Heat transfer processes are of importance in specific to describe experiments with large surface-to-volume ratios. With respect to the prediction of pressure differences the adopted procedure to study all possible locations of primary system ruptures in large containment systems in terms of generated pressure differences probably exclude large design errors. It seems, however, desirable to improve the analytical technique which describes the fluiddynamic processes of containment pressurizations by switching from multi-node point model technique to one- or three-dimensional codes. This may improve the possibility to predict pressure differences also in those areas where no energy release from the primary or secondary system has to be assumed for the containment design calculations.

Improvement in measurement technique is highly desirable in specific in the field of two-phase flow transient measurements and local density measurements. Up to now the only reliable fluiddynamic information useful for the verification of codes is expected from pressure and temperature measurements.

A careful selection of flow resistance and water entrainment parameters is required in the analysis to guarantee conservative results of containment design calculations. A specific experimental program is going to be envisaged to study the possible range of water entrainment and flow resistance values for typical containment flowpath orifices.

KSWING studies performed for Marviken blowdown experiments showed the importance of the absolute pressure, coolant temperature and mass flow density for the behaviour of condensing steam bubbles surrounded by cold water. Fig. 1 shows as an example the good agreement between frequency and amplitudes between experiment and analysis. Table 1 gives quantitative results of the reference case corresponding to Fig. 1.

5. Next Steps

With respect to the development of improved codes work on a comprehensive code describing pressure suppression system effects will continue within the next year. For codes describing full pressure containment systems an improvement is expected with respect to pressure wave propagation simulation by additional development work. The simulation of hydrogen distribution is going to be subject to experimental verification as mentioned above.

6. Relations with Other Projects

Strong relations exist between our code-development and -verification program and the experimental facilities operated by the Battelle-Institute in Frankfurt (RS 50), the Marviken II experiments and the HDR program.

7. Reference Documents

/1/
G. Mansfeld
ZOCO VI - Ein Rechenprogramm zur Berechnung von zeitlichen und örtlichen Druckverteilungen in Volldrucksicherheitsbehältern wassergekühlter Reaktoren. Programmbeschreibung
MRR-P-14, Dezember 1974

/2/
G. Mansfeld
Instationäre Druck-, Druckdifferenz- und Temperaturverläufe in vielfach unterteilten Volldrucksicherheitsbehältern. Vergleich zwischen Theorie und RS 50 - Experiment C1
MRR 149, Mai 1975

/3/
P. Schally, W. Schweiger
Versuch zur modellmäßigen Erfassung von Kondensationsschwingungen.
Interner Bericht
MRR-I-45, Mai 1975

/4/
S. Herzog
LTOLHZ - Ein Programm zum Überspielen von Daten, die in PICT-konformer Weise erstellt wurden. Programmbeschreibung. Interner Bericht
MRR-I-46, Mai 1975

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Behandlung der digitalisierten Meßergebnisse im Rahmen der Experimentalversuche RS 16 und RS 50. Interner Bericht
MRR-I-47, Juni 1975

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H. Bauer, C. Stellmach
Nachrechnungen zu den Hauptversuchen C2, C3 und C5 des Forschungsvorhabens RS 50 "Druckverteilung im Containment". Interner Bericht
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H. Karwat
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8. Degree of Availability

Documents are available through
Laboratorium für Reaktorregelung und Anlagensicherung
D-8046 Garching
Federal Republic of Germany

Reports MRR-I are confidential and therefore normally not available.

Classification: 7.1	
Title 1 (Original Language): Auslegung, Vorausberechnung und Auswertung der HDR-Blowdown-Experimente zur dynamischen Belastung und Beanspruchung von Reaktordruckbehältereinbauten (PNS 4221-I.1.4, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Design and Pre-calculation of the HDR-Blowdown-Experiments on Dynamic Loading, Stresses and Deformation of Reactor Vessel Internals	Project Leader: Dr.Krieg Dr.Schlechtendahl
Initiated (Date): :tober 1974 — Completed (Date): 1979	
Status: continuing — Last Updating (Date): December 1975	

1. General aim

Up to date methods and codes are available to design LWR-reactors in such a way, that a blowdown-accident due to failure of one of the primary coolant circuits can be safely contained. However, in the case of such a blowdown it is not known so far, up to which degree the load carrying potential of the plant will actually be utilized. Hence, on present knowledge a further development to probabilistic safety analysis is not yet possible. To investigate the safety margins in detail more realistic methods and codes verified by appropriate experiments rather than the well known conservative models must be developed.

2. Particular objectives

In particular to assess the proper function of both the shutdown-system and the emergency-cooling-system after blowdown, the dynamic stresses and deformations of the vessel internals must be investigated. To get realistic results the phenomenon of fluid-structure interaction, i.e. the feedback of structural flexibility upon the imposed pressure field has to be included. That means, in experimental investigations the thickness of relevant structural members should be relatively small as to allow for appropriate flexibilities, while with respect to theoretical investigations the development of computer codes for coupled fluid and structural dynamics is mandatory.

3.1 Experimental facilities

Most important for investigations as described above are experiments which allow for an adequate instrumentation without creating scaling problems. In fact, the (out-of-service-)HDR-reactor, which is being changed to an almost full-scale model of a PWR-reactor, is an appropriate experimental facility. A PWR-typical temperature distribution of the water within the reactor vessel can be provided by a special loop system with fossil heating. The blowdown is initiated by a bursting membrane at the end of a so-called burst-nozzle, which simulates the entrance nozzle of the primary cooling circuit. According to existing knowledge the maximum stresses and deformations will occur either within the first phase which is dominated by propagating expansion waves, or within the second phase which is dominated by inertia (acceleration). Hence, only this time region must be simulated properly by the experiment. Since the flow resistance mostly influence the blowdown for later time regions, a detailed simulation of the flow resistances within the core was of minor importance. Consequently the vessel internals are only represented by the core barrel carrying a mass which is to simulate the core mass. Fuel element dummies will not be used in the first series of tests.

3.2 Research program

Several blowdown tests under varying thermodynamic conditions (different temperatur differences between inside and outside the core barrel) are planned. The experimental results will consist of pressures, pressure differences, temperatures, core barrel strains, displacements and accelerations for representative locations all over the outer surface of the core barrel. All these signals as well as the flow rate in the blowdown nozzle will be measured as a function of time. The experimental results will be compared with results obtained from computer models. Some of these computations will be done before and others after the particular test run. Depending on agreement between experiment and theory the computer codes will be confirmed, or else, the physical model or the mathematical procedure in the codes will be improved.

Some of the advanced computer programs for coupled fluid structural dynamics in the case of a reactor blowdown are now under development. Details can be found in the annual report for project PNS 4223.

4. Project status - Progress to date - Essential results

As a basis of the HDR blowdown testprogram a reference test was defined, which resembles most the blowdown in a typical PWR (failure in the cold line). Starting from this reference test the thermodynamic conditions for the other tests were changed to such an extent that - as far as possible and foreseeable - the following effects can be studied seperately: The pressure drop from the initiating conditions to saturation conditions, the transient mass flow in the blowdown nozzle, or the size of the annular space in which early boiling must be anticipated.

Care was taken to realize an upper end core barrel support which can be idealized by a fixed support. At the same time the upper end core barrel flange had to have a minimum flexibility to allow for a well defined pre-stressing. Therefore, flange torsion assessments as well as finite element analyses for the flange cross section had to be carried out. The key point of core barrel design was the core barrel wall thickness which should be as small as possible. First, the transient load was obtained from calculations of expansion wave propagations within annular space. These were done by LRA and by IRS. Then a detailed transient finite element stress analysis was carried out for the core barrel as a cylindrical shell with bending resistance, including the upper and the lower flange. It was found that for a wall thickness of 23 mm the stresses will be always within the elastic region.

To simulate the temperature difference between inside and outside core barrel before blowdown, it was planned to replace the hot water within annular space by colder water, which enters at the upper end of the annular space. Recently it had been turned out that this method will lead to an extremly instable temperature distribution. However, for proper test interpretation the exact temperature distribution within the reactor vessel before blowdown has to be known. Therefore another method with the colder water entering the annular space from the lower plenum has been proposed. In this way it seems to be possible to establish a stable temperature difference between inside and outside core barrel of at least 48°C.

5. Next steps

Since in the design methods used up to now the load reducing effect of coupled fluid structural dynamics could not be included, it is likely, that only a part of the elastic load carrying potential of the core barrel will be utilized. For optimal test interpretation the full utilization of the elastic potential of the core barrel is desirable. Consequently work is on the way to find out appropriate modifications for increasing the blowdown load. With respect to the establishment of the fluid temperature distribution before starting of the test runs, a decision has to be made, whether or not the heating loop concept will be changed.

6. Relation with other projects

This research program is based on the RS16 program, where blowdowns of a vessel of about the same height but a diameter of only about 0.8 m are investigated. Furthermore, as mentioned before, parallel to this experimental code verification program, blowdown codes with coupled fluid structural dynamics will be developed under project PNS 4223.

7. Reference documents

/¯1_7 R. Krieg: Zur Geschwindigkeitsmessung von Anlaufströmungen mit Hilfe des drag-body-Verfahrens. To be published in Brennstoff-Wärme-Kraft, 1976

/¯2_7 R. Krieg: Die wichtigsten Überlegungen zur konstruktiven Gestaltung und Durchführung der DWR-Blowdown-Versuche am HDR-Reactor. (1975)

/¯3_7 R. Krieg, B. Laursen: Auslegung des Kernmantels und der Kernmanteleinspannung für die DWR-Blowdown-Versuche am HDR-Reaktor. (1975/1976)

8. Degree of availability

/¯1_7 Unrestricted distribution.

/¯2_7 ... /¯3_7 Restricted distribution.

Classification: 7.1		
Title 1 (Original Language): Meßtechnische Erfassung und Auswertung des dynamischen Verhaltens der Versuchseinbauten im Reaktordruckbehälter des HDR im Rahmen der HDR-Blowdown-Versuche (PNS 4222 - I.1.4., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Experimental Data Acquisition and Processing of the Dynamic Behavior of the Pressure Vessel Test Internals under the HDR-Blowdown-Experiments		Project Leader: R.A. Müller
Initiated (Date): 1.1.1975	Completed (Date): 1978	
Status: continuing	Last Updating (Date): December 1975	

1. General Aim

Methods and code are presently available which allow LWR reactor design in such a way that a blowdown accident due to failure of one of the primary coolant circuits can be safely contained. However, in the case of such a blowdown the degree is not yet known up to which the load carrying potential of the plant will actually be utilized. To investigate the safety margins in detail, the conservative models must be replaced by more realistic methods and codes which will have to be verified by appropriate experiments.

2. Particular Objectives

The dynamic stresses and deformations of the vessel internals must be investigated in particular. To get realistic results, the phenomenon of fluid-structure interaction has to be included, i.e. the feedback of structural flexibility upon the pressure field imposed.

Consequently, in appropriate experiments the variables of state in the fluid as well as the dynamic reactions of the loaded structures must be recorded simultaneously by measurement technology. It is the objective of this task to measure the structure response (strains, accelerations, displacements) in the planned blowdown tests to be performed in the HDR facility and to evaluate the experimental data. Experience gathered in other similar projects have shown that the instrumentation used must satisfy very stringent quality requirements if an excessive number of failures and a negative influence on the experimental results, respectively, is to be avoided. Thus, the aim of the task includes developing an appropriate instrumentation for large-scale experiments, testing this instrumentation and making it available for the scheduled experiments.

3.1 Experimental Facilities

Experiments allowing to provide an adequate instrumentation without creating scaling problems are most important the investigations described above. Therefore, the (ouf-of-service-) HDR-reactor, which had been converted into an almost full-scale model of a PWR-reactor, is used as an appropriate experimental facility. Details can be found in the annual report of the PNS 4221 project. In the laboratory an autoclave system is used for the development and testing of the instrumentation.

4. Project Status - Progress to Date

A proposal was worked out for instrumentation of the core barrel and after minor modifications included in the test specification for the HDR Blowdown-Experiments. It was agreed that part of the necessary instrumentation will be provided by Battelle (temperature, pressure and differential pressure transducers) and part by IRE (strain, acceleration and displacement transducers). An autoclave test device was conceived to test the instrumentation to be supplied by IRE. It will be possible with this device to expose the prototype transducers to realistic or even tightened up conditions of operation. Investigation started of the measurement techniques,which are best suited for the present case of application and a first instrumentation test program is being prepared.

5. Next Steps

The test autoclave system will be ordered, installed and subjected to test operation. After delivery of the prototype transducers the instrumentation tests will be carried out. These tests will pursue a double objective: (1) determination of the mechanical and electric operating reliability under the given test conditions, and (2) determination of the accuracy of measurement under the given test conditions.

6. Relation with Other Projects

The data on the structure response under blowdown conditions, which were found in the experiment and subsequently processed, will be made available to the PNS 4223 project for future use.

7. Reference Documents

PNS-Arbeitsbericht Nr. 47/75 (august 1975)

8. Degree of Availability

Restricted distribution

Classification: 7.1	
Title 1 (Original Language): Weiterentwicklung und Verifizierung gekoppelter fluid- und strukturdynamischer Codes zur Analyse der dynamischen Spannungen und Verformungen von RDB-Einbauten bei Kühlmittelverlustunfällen in LWR (PNS 4223 - I.1.4, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Development and Verification of Coupled Fluid-Structural-Dynamic Codes for Stress and Deformation Analysis of Reactor Vessel Internals under Blowdown Loading	Project Leader: Dr. Krieg, Dr. Schlechtendahl
Initiated (Date): Oct. 1974 — Completed (Date): 1979	
Status: continuing — Last Updating (Date): Dec. 1975	

1. General aim

Up to date methods and codes are available to design LWR-reactors in such a way, that a blowdown-accident due to failure of one of the primary coolant circuits can be safely contained. However, in the case of such a blowdown it is not known so far, up to which degree the load carrying potential of the plant will actually be utilized. Hence, on present knowledge a further development to probabilistic safety analysis is not yet possible. To investigate the safety margins in detail more realistic methods and codes verified by appropriate experiments rather than the well known conservative models must be developed.

2. Particular objectives

In particular to assess the proper function of both the shutdown-system and the emergency-cooling-system after blowdown, the dynamic stresses and deformations of the vessel internals must be investigated. To get realistic results the phenomenon of fluid-structure interaction, i.e. the feedback of structural flexibility upon the imposed pressure field has to be included. That means, in experimental investigations the thickness of relevant structural members should be relatively small as to allow for appropriate flexibilities, while with respect to theoretical investigations the development of computer codes for coupled fluid and structural dynamics is mandatory.

3.1 Experimental facilities

Most important for investigations described before are experiments which allow for an adequate instrumentation without creating scaling problems. Therefore the (out-of-service-)HDR-reactor, which has been changed to an almost full-scale model of a PWR-reactor, is used as an appropriate experimental facility. Details can be found in the annual report for project PNS 4221.

Furthermore a series of small-scale experiments for investigation of special blowdown phenomena are going to be planned.

3.2 Research program

The experimental results will be compared with computer results for stress and deformation analysis of reactor vessel internals under blowdown loading. Appropriate computer codes which take account for the effect of coupled fluid and structural dynamics are under development. For analysis of the fluid dynamics within the annular space between reactor vessel and core barrel the finite difference technique will be used. Basically these codes will be two-dimensional, but having special features to allow for dynamic variations of the thickness within the third dimension. The so-called singularity method is under discussion as an alternative method especially for the fluid region within the core barrel. For analysis of dynamic stresses and deformations of the core barrel care must be taken to obtain reasonable computer times which allow several time steps with iterations in the case of fluid structural coupling. Therefore, a method will be used which is based on analytical solutions of a shell with dynamic loading.

4. Project status - Progress to date - Essential results

The YAQUI program developed in Los Alamos (LASL) for calculation of two-dimensional transient flows was used as a basis for the fluid part. It was adapted to PL/1 and made available as a REGENT subsystem. Trial computations had been successfull. Also special features for non-rectangular boundaries and variable thickness of the flow channels are built in. In addition, top down design was started for another fluid dynamic code FLUST01 which is more flexible than YAQUI for complex geometries. Also progress was made on the method of singularity which is in discussion for analysis of the fluid region within the core barrel. A shell model (static, linear-elastic, without bending resistance) was formulated and verified. It was found that bending resistance must be taken into account. Therefore, another shell model was formulated which included bending resistance and inertia effects. Comparisions between this model and STRUDL/DYNAL computations yielded agreement with respect to natural frequencies and modes of vibrations. The development of a computer program on this basis is under way.

5. Next steps

The most difficult problem is the coupling of the models for fluid and structural dynamics. Depending on the effects of coupling upon stability significant modifications of the particular codes may be necessary.

6. Relations with other projects

This research program is based on the RS16 program, where blowdowns of a vessel with a relatively small diameter are investigated. Furthermore, as mentioned before, parallel to this theoretical program, full scale blowdown experiments will run on the HDR-reactor under project PNS 4221.

7. Reference documents

/ 1 / R. Krieg: Ein halbanalytisches Verfahren zur Spannungs- und Verformungsanalyse für eine linear elastische Kreiszylinder-Schale unter nicht-rotationssymmetrischer (spiegelsymmetrischer) dynamischer Belastung normal zur Schalenfläche.

8. Degree of availability

Restricted distribution.

Classification 7.1

Title 1 Internal Missile Effects	**Country** UK
Title 2	**Sponsor** UKAEA **Organisation** SRD, Culcheth
Initiated 1974 **Completed** **Status** Continuing **Last updating**	**Project leaders** D. L. Hunt

1 General Aim

To enable the possible missile effect to be predicted following an assumed circuit or vessel rupture, and thus to design containment.

2 Particular Objectives

For the assumed case of a ruptured pressure vessel, to predict the velocity which may be attained by a large fragment.

3. Experimental Facilities and Programme

A tube (circa 20 cm. diameter) is pressurised to LWR conditions, then the end is removed by bursting a diaphragm. The acceleration is observed.

4 Project Status

The "steam rocket" experiment has been built to test SRD calculational methods. The flight of the pipe is measured over 15 ft. mainly by fast photography (2,000 frames/sec.) against a grid, but acceleration over the first 6" is by a photo-electric method. Pressure variation in flight will be measured using tansducers with flying leads.

The first test took place in mid-October, with further tests up to the end of March, 1975.

PROJECT TITLE : Development of a code for transient analysis	CLASSIFICATION 7.1
SPONSORING COUNTRY : ITALY	ORGANISATION : NIRA-Cirene
DATE INITIATED : 1974 DATE COMPLETED : end of 1976	PROJECT LEADER : NIRA

Description :

Scope of the work is the development of the TILT code for the analysis of the transient following a LOCA in a pressure tube heavy water reactor. The work is performed in the framework of thecontract awarded from CNEN to NIRA for the design of the CIRENE prototype Nuclear Island.

Reference documents

NIRA Report
"Il Programma di calcolo TILT NIRA" - 01300 - RT - 19

Classification: 7.2	
Title 1 (Original Language): Versuche zum Wärmeübergang bei Eiskondensation (RS 67 - I.1.4., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: TU München
Title 2 (english): Experiments for Heat Transfer at Ice-Condensation	Project Leader: Prof. Dr. Grigull
Initiated (Date): .3.1973 — Completed (Date): 29.2.1976 Status: continuing — Last Updating (Date): December 1975	

1. General aim

A leak in the primary loop of a reactor results in a pressure increase in the reactor containment due to outflow of water-steam mixture. If the steam can be condensed e.g. on an ice surface the pressure increase is nearly avoided and therefore the danger of contamination following after a containment failure minimized.

2. Particular objectives

...is research program will investigate the heat- and mass transfer after the condensation of steam at vertical ice-surface and establish analytic means for calculating ice-condensers.

3. Experimental facilities and program

See 2nd Annual Report RS 67 - I.1.4, A 74

4. Project status

4.1 Progress to date

The following work has been completed

- construction and test of the thermopiles
- test runs with steam flowing upwards
- improvement of the hard- and software of the data acquisition system.

A thermopile consists of 14 thermocuples in series with a junction distance of 10 mm and is used for the measurement of the melting rate in 18 points.

The test run showed that the measurements of temperature, total pressure and dynamic pressure of the steam as well as the melting rate of the ice in the ice channel are reliable.

Filmshooting of the waterdrops in the steamflow is also practicable.

4.2 Essential results

The steam velocity in the ice channel is dependent on the steam flow rate and alters along and across the ice surface as well as with the distance from it. The steam rips the liquid film of molten ice and condensed steam.

If the steam velocity is high ($\sim$ 100 m/s), then the water drops drift in the steamflow. If it is low ($\sim$ 40 m/s), then they may fall downwards into regions with higher velocity and are carried up again. The two Phase flow is very turbulent.

The melting rate at high velocity is about 5 mm/s, at low velocity about 2 mm/s.

5. Next steps

- further test runs with the steam flowing upwards
- test runs with the steam flowing downwards

6. Relation with other projects

- see Annual report IRS - F - 24

7. Reference documents

- Quarterly Reports (German) in the series "IRS-Forschungsberichte" Oct. 1974 - Sept. 1975: IRS - F - 23, 25, 26, 27
- Annual Report (English) in the series: "IRS-Forschungsberichte" Dec. 1973 - Dec. 1974: IRS - F - 24
- Progress Report (German) "Versuche zum Wärmeübergang bei Eiskondensation" March 1973 - July 1975

8. Degree of availability

The reports in the series "IRS-Forschungsberichte" are available from the Institut für Reaktorsicherheit, Köln. The progress report is available from the Technische Universität München, Lehrstuhl A für Thermodynamik.

Classification: 7.2	
Title 1 (Original Language): Theoretische Arbeiten zum Druckabbausystem (Kondensation IV, Teil 1) (RS 78 B - I.1.4, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Frankfurt
Title 2 (english): Theoretical Investigations of the Pressure Suppression System (Condensation IV, Part 1)	Project Leader: Dr. Sobottka
Initiated (Date): 15. 10. 74 Status: Continuing	Completed (Date): March 76 Last Updating (Date): 31. 12. 75

General Aim and Particular Objectives

Evaluation of theoretical models to describe special events in the pressure suppression system (unsymmetric water swell and condensation oscillation) during a loss of coolant accident. To use existing experimental material for the model development, a special data evaluation for the problems mentioned must be executed.

Project Status/Progress to Date

Unsymmetric Water Swell:

The reason for the unsymmetric water swell was investigated. The air/vapour mixture was calculated on different places in the upper ring room of the pressure chamber, which was parted theoretically into several smaller rooms with homogenous air/vapour mixture. Concerning the airstream into the water of the condensation chamber, the unsymmetric water swell was calculated without lateral flow of water.

The air transport through the water was determined with a new model for the bubble ascension.

In order to improve the model with respect to the experimental results, the radial extension of the air bubbles was taken into consideration. For the calculations a two-dimensional model was used.

Condensation Oscillations

As the ancient model, describing the condensation oscillations only based on a pure vapour supply, the model was extended to mixture of vapour and not condensing gases. Besides this the model for the heat transfer on the surface of a bubble was improved. After this, stationary oscillations and collapsing can be calculated, which were observed during the experimental tests.

The model KSWING IV was refined and prepared for routine calculations. With various parameters the Marviken and GKM tests were reexamined.

In order to describe the pressure extension in a fluid of a certain geometry during collapsing of one or several bubbles a monocell-modell was developed, which attached every bubble to a certain fluid room or a certain effective mass. Coupled with a single mass oscillator, considering elasticity and eigenfrequency of the containment walls, the pressure in the containment can be calculated on every point.

Project Status/Essential Results

Unsymmetric Water Swell:

A parameter study showed, that the new model with the refined bubble model gives lower oscillations of the water surface. A comparison with experimental data showed, that the ascension velocity of the bubbles in the initial phase is probably higher than expected recently.

Condensation Oscillations

Tests with the new model showed that the results were in good agreement with the results obtained from the condensation oscillation.

The recalculation of the Marviken and GKM tests gave a smaller bubble diameter at higher frequences compared with the radial symmetric energy transport formula. After the introduction of a new boundary layer concept the results were in good agreement with the experimental data.

With the monocell-oscillation model the measured pressure course of different containments were confirmed, results for other geometries were obtained.

Next Steps

Unsymmetric Water Swell:

Evaluation of the boundary conditions

Condensations Oscillations:

Continued work on the model KSWING IV. The monocell modell wil be refined, using a three dimensional potential.

Reference with Other Projects

see RS 78-A 74

Reference Documents/Degree of Availability

No reports available.

Classification: 7.2	
Title 1 (Original Language): Dynamische Beanspruchung von LWR-Druckabbausystemen (PNS 4211 - I.1.4., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Dynamic Load of LWR Pressure Suppression Systems	Project Leader: R.A. Müller
Initiated (Date): January 1972 Status: continuing Completed (Date): December 1976 Last Updating (Date): December 1975	

General Aim

The pressure suppression systems used in boiling water reactor facilities are exposed to dynamic loads during specific conditions of operation and in a loss-of-coolant accident, the characteristics and intensity of which is not yet sufficiently known in all cases. The aim consists in enlarging knowledge of the extended steady-state condensation process so that the dynamic loads applied to the components can be determined. The implementation of this task is governed by the availability of suitable test facilities.

Particular Objectives

Evaluation of the intensity and characteristics of pressure pulsations taking place during condensation in the water pool (essential parameters: mass flow density, pipe diameter, number of pipes, water temperature, pressure level and air content of the steam).

Influence of the drywell on the condensation process.

Stresses and dynamic behavior of containment structures under the influence of condensation loads.

Project Status

Progress to Date

In the containment experiments performed in the Marviken Nuclear Power Station in 1972 and 1973 higher pressure oscillations were observed to occur in the pressure suppression system, above all in tests Nos. 10 and 13, which seemed to indicate possible system resonances. To explore these problems, the natural frequencies of the Marviken plant were calculated. A model was used for the calculations, which takes into account coupling of the three most important single systems,namely blow-down channel, header and vent pipe. The results are strongly dependet on the velocity of sound which in turn is determined mainly by the steam moisture.

For realistic values of the velocity of sound the first three natural frequencies calculated by this model come close to the measured values to be derived from the test data. Since also the excitating frequency from the water pool, which is caused by condensation, lies within the range of the first and second natural frequencies, conditions could actually have been passed during blowdown under which the steam and water volume fraction, respectively, in the drywell has generated such a sound velocity and hence natural frequency close to the frequency range of excitation that resonance took place. This would explain the greater pressure amplitudes observed.

The development work in the laboratory scale of the infrared measurement technique was terminated and an offer submitted to the Marviken II Project. After the order for one measuring position had been received, the measuring equipment was conceived, designed and manufactured. The supporting frames and the supply lines were installed in Marviken while the optical source and receiver components as well as the electric and electronic supply units were pressembled and tested in the workshop. This development work was carried out together with Institut für Thermische Strömungsmaschinenen (ITS) of Karlsruhe University.

As a supplement to the tests with the pressure relief nozzles performed in the Brunsbüttel Nuclear Power Station in October 1974 further tests were made in June 1975 with a vent pipe of 600 mm Ø. The measured values recorded by IRE in the two test series were evaluated. In all tests standing pressure waves having low amplitudes were generated in the annular water pool in addition to the local events taking place.

A transient calculation was performed in the studies of the dynamic behavior of the containment structure. For this purpose, a pressure field in terms of time and space was interpolated from one test of the first Brunsbüttel test series applied as a load to the model of calculation. The displacement with respect to time of interesting points of the containment was determined. The actual displacements were assayed by double integration of the measured accelerations and thus comparable with the calculated results. The agreement is not yet satisfactory.

Within a new series of condensation tests involving single and multipipe arrangements and carried out by KWU in Karlstein (previously Großwelzheim) measurements were performed in a total of 11 tests. These measurements related to the pressure development with respect to time at 15 points of measurement arranged on three coordinates at right angles to each other. The origin of the coordinates was the expected bubble center. The evaluation first preferably concerned the tests having a mass flow density of 16 kg/m^2 sec since this value had been used to investigate all the configurations (pipe diameter, number of pipes).

Moreover, first parameter calculations were started within the period of reporting, using a computer program which describes the thermohydraulic events of the condensation process. In this program vent pipes operating in parallel are treated and significant coupling mechanisms on the steam and water sides are taken into account.

Essential Results

The computations are well as the observations made during the Marviken I tests support the assumption that events resembling resonances might taken place in pressure suppression systems under specific conditions of geometry and operation, which might lead to pressure pulsation amplification.

According to the results of laboratory tests use of the infrared absorption measurement technique is promissory for determination of the moisture content of a two-phase flow in blowdown tests and for determination of the velocity of the liquid phase.

In toroidal water pools of pressure suppression systems standing pressure waves having low amplitudes are generated in addition to the local events caused by the individual vent pipes.

The high pressure peaks generated during condensation of air-free steam and superimposed to the usual pressure pulsations with smaller amplitudes have a very short base width of about 0,15 msec. Therefore, their significance with respect to current structural components is generally low.

Next Steps

Final assembly work, commissioning and use in future tests of the infrared measurement equipment delivered to Marviken.

The calculations will be continued relative to the dynamic behavior of the containment structure. Analysis of further test results is to yield the reasons of the as yet unsatisfactory agreement between the calculated and experimental displacements so that an improvement of the calculation model can possible by achieved.

The parameter calculations concerning the condensation at one and several parallel pipes are being continued, partly by use of improved programs.

Reference documents

Report KFK 2195 (1975)
Report KFK 2262 (1975)

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Vorläufige Resultate der Untersuchungen des Druckpulsationsfeldes in der Wasservorlage der Kondensationskammer der entsprechenden dynamischen Reaktion des Sicherheitsbehälters während der Entlastungsventilversuche im Kernkraftwerk Brunsbüttel.
Reaktortagung, Nürnberg, 8. - 11. April 1975,
Deutsches Atomforum e.V., Kerntechnische Ges. im Dt. Atomforum e.V., Leopoldshafen 1975

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Investigations of the Fluctuating Pressure Field in the Suppression Pool of the Marviken Containment during Blowdown,
3. Internat. Conference on Structural Mechanics in Reactor Technology, London, September 1- 5, 1975

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R.A. Müller, E.G. Schlechtendahl,
Dynamische Belastungen und Strukturverhalten von Reaktorsicherheitsbehältern mit Druckabbausystemen,
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B. Göller, B. Laursen, E.G. Schlechtendahl,
Results of Calculation of the Dynamic Behavior of Pressure Suppression System during Blowdown
Extreme Load Conditions and Limit Analysis Procedures for Structural Reactor Safeguards and Containment Structures, Berlin, Sept. 8 - 11, 1975

Degree of Availability

Unrestricted Distribution

PROJECT TITLE : HEAT TRANSFER IN VAPOUR SUPPRESSION SYSTEM	CLASSIFICATION : 7.2
SPONSORING COUNTRY : ITALY	ORGANISATION : CNEN
DATE INITIATED : JUNE 73 DATE COMPLETED : DECEMBER 76	PROJECT LEADER : G.E. FARELLO

Description :

The vapour suppression system is one of the important safety components of the L.W.R.
After the emergency due to coolant loss, begins blow-down into water tank from the pressure vessel in order to prevent the release of radioactive materials. It is possible to investigate the behaviour of steam jets in subcooled water with an experimental apparatus which allows a photografic evidence of the basic phenomena in the vapour suppression containment.

The research is performed in a small experimental loop which utilizes high speed cinematography and schlieren techniques. Heat transfer (vapour to liquid) coefficients and thermodynamic instabilities are the main objects of the research.

PROJECT TITLE :	CLASSIFICATION
SOPRE 1 - Research on the behaviour of pressure suppression containment system after LOCA	7.2
SPONSORING COUNTRY : ITALY	ORGANISATION : UNIVERSITY OF PISA
DATE INITIATED : 1974 (actual phase) DATE COMPLETED : 1976 (actual phase)	PROJECT LEADER : M. MAZZINI

Description :

Research program:

Experimental investigation of pressure and temperature transients within a model of MARK II pressure suppression containment system. The effects of blow-down flow rate, inlet area and flow rate distribution among the vent pipes, are examined.

Assess of capability of computer codes, like CONTEMPT-PS, to fit experimental data.

In the next future it is possible that MARK III type systems will be studied.

Facilities:

SOPRE 1 apparatus, which is a 1:13 scale model of MARK II system. Steam is supplied by a pressure vessel; its main features are: design pressure 100 Kg/cm^2, design temperature 310 °C, capacity 100 litres, blow-down area variable up to 6 cm of equivalent diameter.

Reference documents:

1. B. GUERRINI, M. MAZZINI

L'apparecchiatura sperimentale SOPRE 1 per ricerche sui sistemi di contenimento a soppressione di pressione.

To be issued.

Related projects:

1.1.1 - 7.2(CNEN)

Remark:

The research is performed on C.N.R. contract. Financial support by C.N.E.N. and ENEL is hoped for the future.

CLASSIFICATION

7.3

TITLE 1 CORROSION EN AMBIANCES NATURELLES ET ARTIFICIELLES	COUNTRY FRANCE
	SPONSOR E.D.F.
	ORGANIZATION E.D.F.
TITLE 2 CORROSION OF MATERIALS INSIDE OF THE CONTAINMENT	Project Leader E.D.F./DER/TEGAEE
Initiated février 1975 Completed 1976	Scientists MM. BUREAU BERGE
Status Last updating : 20.01.75	

I - GENERAL AIM

II - PARTICULAR OBJECTIVES

Détermination de la quantité d'hydrogène produite lors de l'aspersion dans l'enceinte de confinement d'une chaudière à eau sous-pression d'une solution d'acide borique et de soude après une rupture de tuyauterie primaire.

III - EXPERIMENTAL FACILITIES AND PROGRAMME

Etude de la corrosion de l'alliage léger AG 5 et des zincs de galvanisation Z6 et Z9. Influence de la température du pH, du mode de contact métal-solution et de la composition des alliages légers.

IV - PROJECT STATUS

Début des essais en 1975.

V - NEXT STEPS

- essais en immersion,
- essais en aspersion.

...

VI - RELATION WITH OTHER PROJECTS

Néant.

VII - REFERENCE DOCUMENTS

Aspersion de l'enceinte de confinement d'une chaudière à eau sous-pression. Aspects chimiques. Note E.D.F./SEPTEN (restreinte E.D.F.)

IX - DEGREE OF AVAILABILITY

E.D.F.

PROJECT TITLE : Control of Hydrogen Concentration in Containment Following a LOCA	CLASSIFICATION 7.3
SPONSORING COUNTRY : ITALY	ORGANISATION : C.AM.E.N.-S.Piero a Grado (PISA) UNIVERSITY OF PISA
DATE INITIATED : 1975 DATE COMPLETED : 1977	PROJECT LEADER : S. LANZA (University) S. MANFREDINI (CAMEN) R. MIRANDOLA (Univers.)

Description:

The aim of the research program is to study hydrogen transfer inside a containment with the final goal of evaluating the capability of proposed devices to keep hydrogen concentration in reactor containments below the flammability point in post-LOCA conditions.

Guidelines of the program are:

- theoretical evaluation of the processes able to cause hydrogen transfer
- experimental study of hydrogen transfer inside the PSICO 10 vessel to approach the BWR post-LOCA conditions
- control of hydrogen concentration.

The PSICO 10 facility: $\sim$100 m^3 vessel is its main component; features are design pressure 5 ATA, design temperature 150 °C.

This program will be sponsored by CNEN with collaboration of the CAMEN.

References documents

1. GUERRINI B. et alii
PSICO 10: a facility for testing in the field of containment technology for nuclear plants.
Ingegneria Nucleare - Marzo-Aprile 1970.

2. GUERRINI B. et alii
L'apparecchiatura sperimentale PSICO 10.
Notiziario CNEN - Ottobre 1972.

RCN	CLASSIFICATION: 7.4
TITLE: Advies inzake periodieke controle van insluitvaten op lekkages.	COUNTRY: NETHERLANDS SPONSOR: Ministry of Social Affairs ORGANIZATION: Reactor Centrum Nederland
TITLE (ENGLISH LANGUAGE): Advice on periodic supervision on leakage tightness of containments.	PROJECTLEADER: H.J. van Grol
INITIATED: June 1974 — COMPLETED: end 1976 STATUS: progressing — LAST UPDATING: february 1976	SCIENTISTS: J.W.H. van den Bergh H. Pruimboom

1. General aim

To assemble data on the measurements of leak rate on containment vessels in order to advise the authorities concerning periodic supervision of leakage performances as compared to design leak rates.

2. Particular objectives

The programme consists of:
- making an inventory of the generally used methods of leakage rates measurements including an evaluation resulting in a general purpose method
- specification of main and secondary variables when measuring leakage rates
- performing an analysis of extrapolation modes in order to establish the safe lower limit of overpressure in relation to design pressure when performing periodic leak rate measurements
- listing experiences obtaines elsewhere when measuring leakage rates.

3. Experimental facilities

Not applicable.

4. Project status

Literature study proceding. About 50% of the questionnaires sent to different power reactor stations have been returned and will be evaluated.

5. Next steps: -

6. Relation with other projects: -

7. Reference documents: -

8. Degree of availability

The reports will be written in the English language. They will be submitted to the Ministry of Social Affairs, Postbus 69, Voorburg.

8. INSTRUMENTATION, CONTROL AND COMPUTERISED PROTECTION

Classification: 8.	
Title 1 (Original Language): Untersuchungen über die Zuverlässigkeit von Druck- und Differenzdruck-Meßumformern unter GaU-Bedingungen (RS 110 - I.1.4, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Investigation of the Behaviour of Pressure and Differential Pressure Transducers under the Influence of an MCA	**Project Leader:** K. Riemann
Initiated (Date): Oct. 1973 — **Completed (Date):** Dec. 1975	
Status: Applied for prolongation — **Last Updating (Date):** 31. 12. 1975	

General Aim and Particular Objectives

Testing of pressure and differential pressure transducers suitable for operation at temperatures up to 150 °C and pressures of 5 bar.

Experimental Facilities and Program

In order to increase reliability under operation and MCA the primary pressure and differential pressure transducers shall be arranged within the safety shell.
Until now data of the reliability of suitable temperature and pressure transducers are only available from the producers. Before they are installed in the reactor, it is indispensable to test "wet" transducers being on the market under simulated MCA-conditions (ϑ= 150 °C, p = 5 bar). If the transducers do not satisfy this conditions it is intended to weld in automatic tube rupture fuses in the pulse pipes. Several types are to be tested and eventually improved.

Project Status/Progress to Date

Tests with pressure and differential pressure transducers of various producers were conducted. Ranges on test were 0 - 210 bar and 0 - 10 bar.

<u>Project Status/Essential Results</u>

Transducers of one producer have not satisfied the simulated MCA conditions. The instruments were destroyed by moisture or showed rather big measuring errors (up to 11 %). The tests showed that casting of the electric components does not secure operation under MCA conditions.

Transducers of another producer worked however quite well under simulated MCA conditions (N_2-H_2O vapour-atmosphere, of 150 °C and 5,8 bar overpressure). After cooling and adjusting the transducers reached the specified measuring accuracy. No destruction was found on the instrumentation and electronic components after inspection. Another MCA test was conducted at 180 °C for half an hour; the transducers operated without difficulties.

The transducers had an error of + 0,2 % to + 1,1 %, the differential transducers of + 0,7 % to + 2,5 %.

<u>Next Steps</u>

The equipment of the transducers, which have failed, will be improved (pressure tight boxes). After this, the MCA tests will be repeated.
Another producer has offered transducers for a test under MCA conditions.

<u>Relation with Other Projects</u>

see RS 110 - A 74

<u>Reference Documents/Degree of Availability</u>

No reports available.

Classification: 8.	
Title 1 (Original Language): Einsatz von Prozeßrechnern in Leistungsreaktoren zur Verbesserung der Betriebssicherheit (ATT 085 A - II.4.4., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (english): The Use of Process Computers in Power Reactors for Improvement of Operation Safety	Project Leader: Dr. H. Hoermann
Initiated (Date): Completed (Date):	
Status: continuing Last Updating (Date): December 1975	

1. General Aim

(a)

Improvement of the operational safety through the use of computerized inspection and plant supervision techniques.

(b)

Development of methods and guidelines for the construction of safe process computer systems and for the verification of desired safety or reliability margins.

2. Particular Objectives

(a)

Development of a general concept for computerized inspection of engineered safety systems, in order to improve failure detection and localization with respect to time conditions and overall performance. Investigation of the capability of new methods for on-line analysis of plant disturbances. Development of methods to implement new concepts for incipient failure detection using process computers.

(b)

Investigation of hardware and software structures for safety related computer applications in nuclear power plants. Reliability and availability evaluation of such systems in comparison with hardwired solutions. Development of guidelines for the design, construction and implementation of computer systems with safety related applications; providing methods for the verification of their reliability or their correct performance. Application of the research results and experiences to the computer-based fuel rod protection system for the German LMFBR.

3. Experimental Facilities and Program

(b)

For the tests performed on a computerized reactor protection system for BWRs an AEG 60-10 computer has been installed at the Laboratory.

4. Project status

4.1 Progress to Date

(a)

Using the results of the first simulation experiments the developed inspection routine for engineered safety systems has been refined and improvements concerning the hybrid model have been implemented. During the subsequently performed simulations runs stochastic failures of the mechanical components, as well as instrumentation faults, have been generated with the following conditions:

- An equal failure probability was assumed for all components and the associated instrumentation.
- The maximal number of simultaneously generated faults was limited to three, allowing the mixed occurance of component failures and instrumentation faults.

- The probability for the generation of multiple fault disturbance patterns was based on the binomial distribution.

Such simulation experiments have been carried out substituting different modeling and measuring inaccuracies to derive the achievable performance of the disturbance analysis during the inspection dependent upon these parameters /1/.

For the used reference system (simplified core flooding system operating in the recirculation mode) a useful and tested concept for the computer controlled inspection is now available.

b)

Investigations concerning the hardware related reliability and availability of redundant m-out-of-n computer systems with repair lead to evaluation formulas, which, up to now, had not been presented in the relevant literature with the necessary degree of accuracy. In addition, the influence of individual failure detection rates for different failure modes on the overall availability figures of a triple-modulary redundant system has been derived. Based on this theoretical background recommendations for the construction of such systems and for appropriate failure detecting measures could be given.

he reliability assessment work directed to the KWU computerized protection system for BWRs lead to improvements concerning the completeness of failure detection and the availability of the process interface unit (AD converter). The hybrid computer test of this system comprised more than 10^5 test cases. Based on that work a reliability study comparing the computerized protection system with a hardwired (dynamic solid state) system has been elaborated as far as unambiguously safety increasing actions are concerned.

Using the gained experiences and further theoretical work a lot of guidelines have been formulated for the software development directed to the computer-based fuel rod protection system of the LMFBR /4/.

4.2 Essential Results

(a)

The verification of the normal operating characteristics of a system as well as the detection of system disturbances can be carried out easily and with sufficient reliability by checking a predefined set of relevant system variables or parameters against their nominal values.

To achieve sufficient reliability and performance of the disturbance analysis, a system decomposition is undertaken and the analysis is carried out in two steps. In the first step the disturbed and undisturbed subsystems are separated. Even if complex disturbance patterns occur, for which a prime cause analysis would fail, the subsystems check still has a high reliability.

As the consequences of basic faults only can be predicted qualitatively, the prime cause analysis, which is the second step, is to be restricted to single failures in each subsystem. This, in principle, leads to limitations of the inspection performance. The loss of performance can be kept small, if the subsystems are made small enougl by extensive system decomposition. For the basic fault analysis component faults and instrumentation faults are handled separately. Thus misinterpretations are very unlikely.

(b)

The derived relations between the availability of computer systems with repair and the failure detection rates and measures allow their specification dependent on the predefined reliability margins. Vice versa, the hardware related safety performance may be ascertained for differently structured systems with given failure detection features.

Preliminary results indicate that with regard to unambiguously safety increasing protection measures the probability for a dangerous failure of a seriously constructed computerized system may be kept on a reasonable low level but will not achieve the values offered by advanced hardwired solutions.

5. Next Steps

(a)

The method developed will be applied to an emergency core cooling system of a PWR, since for this system considerable work on systems analysis has already been carried out. It is intended to verify the concept of independent basic faults and to estimate the amount of instrumentation equipment required before the method will be proved in a nuclear power plant.

(b)

The comparative reliability assessment study for reactor protection system will be continued with the consideration of measures which have no unambiguously safety increasing effect.

he work on guidelines for safe system design and verification methods will be continued and applied during the advisory co-operation for the computer-based fuel rod protection system for the LMFBR and further comparable applications.

6. Relation with Other Projects

The project is related to the following activities:

- Project Nuclear Safety (GfK).
- Project Fast Breeder Reactor (GfK, IA).
- Project Process Control with DP-Systems (GfK).
- Halden Reactor Project (OECD).

7. Reference Documents

/1/
W. Hawickhorst
A General Approach to Computerized Inspection of Engineered Safety Systems
IAEA-Symposium on Reliability of Nuclear Power Plant, Innsbruck, Austria, April 1975, IAEA-SM-195/19

/2/
R. Schwaiger
Simulation des dynamischen Verhaltens einer Rechnerlogik
Diplomarbeit am LRA, Januar 1975

/3/
K. Wenzl, W. Ehrenberger
Rückübersetzung von Speicherabzügen der AEG 60-10
Datenverarbeitung AEG-Telefunken, 1974, Heft 2/3, published April 1975

/4/
U. Voges, W. Ehrenberger
Vorschläge zu Programmierrichtlinien für ein Reaktorschutzsystem
Report KFK-Ext 13/72 - 2, Mai 1975

/5/
W. Ehrenberger, E. Hofer, H. Hoermann
Probability Considerations Concerning the Test of the Correct Performance of a Process Computer by Means of a Second Computer
Paper presented at the IFAC-Congress, Boston, Mass., USA, August 1975

/6/
H. Hoermann, E. Nieckau
Novel Features in Instrumentation and Automation for the Control of Nuclear Power Stations
Kerntechnik 9/10, vol. 17, 1975

/7/
Quarterly Reports in the Series IRS-Forschungsberichte

8. Degree of Availability

Documents are available through
Laboratorium für Reaktorregelung und Anlagensicherung
D-8046 Garching
Federal Republic of Germany

9. Additional Information

Concerning the computer safety aspects members of the LRA participated in a working group of the VDI/VDE (Document-VDI/VDE 3553: Failure detection and location in process computer systems) and in the TC 'Safety and Security' of the Purdue Europe Workshop on Industrial Computer Applications.

9. OTHER SAFEGUARDS

Classification: 9	
Title 1 (Original Language): Untersuchungsprogramm zur Erprobung einer Berstsicherung für Reaktorkomponenten (RS 104 - I.1.7, Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** KWU, Erlangen
Title 2 (english): Investigation Program for the Testing of a Fracture Safety Device Protection System for Reactor Components	**Project Leader:** Dr. Kopp
Initiated (Date): ebr. 1973 **Status:** Continuing	**Completed (Date):** 31. 12. 1977 **Last Updating (Date):** December 1975

General Aim

Under this project, burst tests are to be carried out on burst-protected pipe cross-sections of various dimensions in order to prove the effectiveness and functional worthiness of a burst protection structure. The break behaviour of shielded main coolant pipes, phenomena subsequent to a fracture and the loading of the burst protection device system are to be determined.

The project is being used simultaneously for the technical estimation of phenomena and loading within the scope of the research project "Fracture Safety Device for Reactor Pressure Vessels".

Experimental Facilities and Program

The investigation program encompasses the following works:

1. Preliminary Tests on Burst-Protection Structure Components
   The material behaviour of insulation concrete subjected to steam and/or water jet loads as well as to step and impulse loads is being investigated. In addition the optimal construction and arrangement of the elements of the fracture safety device is to be determined.

2. Preliminary Tests on Pipe Elements
   The carrying-out of fracture-mechanic tests on Ø 355 x 10 pipes made of 15Mo3 to determine the critical crack size and crack growth. These tests will provide the necessary crack lengths in the test pipes for the burst experiments.

3. Preliminary Tests on Pipe Elements being Protected by a Burst Protection Structure
   Four pipes Ø 355.5 x 19.7, Material ASTM-A 533, Size B (20 MnMoNi 55), which are enclosed by a burst protection structure, are to be caused to burst at a temperature of 300 °C, at a pressure of about 175 bar. The pipes are provided with sealed, wall-penetrating longitudinal slits, the length of which was determined in the preliminary tests (Point 2).

   Pressure rise in the burst protection structure, its loading and the fracture behaviour are to be measured. In addition the break behaviour of pipe elements, being protected by a burst protection structure, is to be investigated for various clearance between the pipe and burst protection structure.

4. Primary Experiments for Testing the Burst Protection Structure
   For 5 pipes Ø 872.2 x 36.1, Material ASTM-A 533, Size B (20 MnMoNi 55), burst experiments are to be carried out at a temperature of 300 °C and pressures around 175 bar. The pipes have sealed, wall-penetrating longitudinal slits and are enclosed by a burst protection structure. Results from the preliminary tests (Point 3) are to be taken into account. The break behaviour of the pipes inside the burst protection structure as well as its loading and its components during a fracture are to be investigated.

Progress State / Progress to Date

Interatom has completed one pipe element for the preliminary tests; it was returned to the producer in order to test the sealing.

Classification: 9

Title 1 (Original Language): Untersuchungsprogramm zur Erprobung einer Berstsicherung für Reaktorkomponenten (RS 104 - I.1.7, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Investigation Program for the Testing of a Fracture Safety Device Protection System for Reactor Components	Project Leader: Dr. Kopp
Initiated (Date): Febr. 1973 Status: Continuing Completed (Date): 31. 12. 1977 Last Updating (Date): December 1975	

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   The material behaviour of insulation concrete subjected to steam and/or water jet loads as well as to step and impulse loads is being investigated. In addition the optimal construction and arrangement of the elements of the fracture safety device is to be determined.

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For 5 pipes Ø 872.2 x 36.1, Material ASTM-A 533, Size B (20 MnMoNi 55), burst experiments are to be carried out at a temperature of 300 °C and pressures around 175 bar. The pipes have sealed, wall-penetrating longitudinal slits and are enclosed by a burst protection structure. Results from the preliminary tests (Point 3) are to be taken into account. The break behaviour of the pipes inside the burst protection structure as well as its loading and its components during a fracture are to be investigated.

Progress State / Progress to Date

Interatom has completed one pipe element for the preliminary tests; it was returned to the producer in order to test the sealing.

The two residual pipes have got sealed, wall-penetration longitudinal slits and are delivered to Interatom for crack growth by pressure oscillation.

With two test pipes burst tests were performed, but bursting was not reached.

Project Status / Essential Results

During oscillation of pipe No. 3 after 5000 cycles leaks occurred which stopped the procedure.

During the burst test No. 1 the water temperature and pressure were forced up to 300 °C and 250 bar. As no bursting occurred, the test had to be stopped because the instrumentation was in danger to be destroyed. Now the notch in the wall will be enlarged before another test is started.

During the burst test No. 2 with a wall penetrating longitudinal slit the temperature and pressure were also forced up to 300 °C and 250 bar, when a little leak was visualized. However bursting was not reached.

Next Steps

Evaluation of the results.
Burst-tests on preliminary pipes.
Construction of the pipes for the main experiments.

Relation with Other Projects

RS 108 Fracture Safety Device for the Primary Circuit

RS 157 Investigation of the Heat-Insulting Cooling System under Dynamic Load Conditions with a View to Fracture Safety Device Protection for Reator Pressure Vessels

Reference Documents/Degree of Availability

No reports available

Classification: 9	
Title 1 (Original Language): Berstsicherheit für Primärkreislauf (RS 108 - I.1.7, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Fracture Safety Device for the Primary Circuit	Project Leader: Dr. Dorner
Initiated (Date): ug. 1973 Completed (Date): April 1976 Status: Continuing Last Updating (Date): December 1975	

General Aim and Particular Objectives

The concept of a fracture safety device for the primary circuit has to be developed, which offers changes for a better protection after MCA. This is relieved by components with simple geometry and small surfaces. Under this project investigations are carried out to change the present design in order to protect the containment after a pressure vessel rupture without reducing access from the outside for inservice inspection and without reducing reliability.

Experimental Facilities and Research Program

The investigation program encompasses the following work:

1. Design of a fracture safety device system for the pressure vessel
2. Design of a fracture safety device system for steam generator
3. Design of a fracture safety device for the pressurizer
4. Design of a fracture safety device for the main coolant pumps
5. Design of a fracture safety device for the primary circuit (piping)

Project Status/Progress to Date

Model tests on semi-axial pumps were completed. Analytical tests on the vibration performance of steam generator primary chamber components were made. Stress analysis of the steam generator primary components as well as determination of the eigen-frequencies are being conducted using the computer program STARDYN. Geometries of the semi-axial and axial pump components were reproduced for use in the program.

Burst protection work on the tubing which consists of single interlocking rings has been finalized for the time being.

An order to Messrs. Krupp was being prepared covering the development of a missile shield cylinder for steam generator burst protection. The cylinder will preferably be fabricated as multilayer construction. A change was made in the missile shield so that the multilayer rings were welded to one cylinder. Thus the cylinder can also absorb axial loads. Consequently, the axial rods can be neglected. Further advantage of this design is a simplified closure head of the missile shield achieved by means of shear wedges.

Major part of the work concerning construction and calculation of the hydraulic cylinder for tightening the steam generator to the RPV burst protection has been completed. Detailed calculations on the lower bottom plate of the steam generator missile shield resulted in stress values which were too high. It is assumed that this will require an increase of the height of the bottom plate from 1.20 m to 1.40 m. Centering of the steam generator with the bottom plate was prepared, which positions the steam generator by three legs welded to the bottom of the primary chamber, absorbing the forces and moments which might occur.

Calculations have been performed on the upper locking device of the RPV burst protection. The dimensions of the supporting flange have been optimized. Calculations showed that slight relative movements might occur between the two construction units due to the temperature differences in the supporting flange and RPV top during start-up and shut down. The RPV support construction with round springs, i. e. necked-down bolts has been optimized.

Since elastomer bearings seem to be extremely useful for the steam generator assembly, the following tests have been performed on full size bearings:

- pressure tests
- shear test
- shear break test
- pressure test with eccentric vertical stress
- temperature test
- accompanying tests (e. g. Shore-hardness)

In anticipation of the test results it can be said that the tests proved the suitability of the elastomer bearings.

Project Status/Essential Results

Use of the round spring rods for RPV support construction results in sufficient design safety factors. Like the turbine vane anchorage, the U-shaped springs, having a rectangular cross-section (100 x 30 mm), will be tightened to the spring legs by a lugged support. A lateral lift-up of the springs will be eliminated by a cross-piece attached laterally to the lugger. This construction, however, requires an immense amount of work and a very costly and exact manufacturing process.

The primary chamber of the steam generator will be provided with a thicker forged ring and attached nozzles (as have been used before). The main steam nozzle was moved up to the upper center and the steam generator cut down for the length of the inside steam guide tube, which is no longer needed.

The results of the tests performed on the elastomer bearings prove the suitability of the type bearing for steam generator assembly; they also indicate, however, that influence of temperature has to be examined in more detail. Construction and calculation of the hydraulic cylinder for tightening the steam generator to the RPV burst protection has been finalized.

Next Steps

Stress calculation of the lower bottom (height: 1.40 m) of steam generator missile shield construction. Armored elastomer bearings are preferred with bridge construction and temperatures of about -20 to +30 °C. Since in the discussed construction temperatures are involved which will be about 100° higher, the influence of temperature has to be investigated in more detail. First of all one has to find out whether or not Chloropren-caouchuc, which is usually used for bearings for bridge constructions, will be the optimum elastomer for our purposes. It can be seen from the literature that different types of elastomer e. g. Silicone-caouchuc or caouchuc fluoride should used with increased temperatures.

The components of the steam generator primary chambers will be calculated with the STARDYN program.

Detailed tests on the reactor components and burst protections will be continued.

Relation with Other Projects

see RS 104

Reference Documents / Degree of Availability

No reports available.

Classification: 9	
Title 1 (Original Language): Untersuchung des Verhaltens des Wärmedämm-Kühl-Systems (WKS) der Berstsicherung (BS) für RDB bei dynamischer Belastung (RS 157 - I.1.7., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** Dyckerhoff & Widmann AG, München
Title 2 (english): Investigation of the Heat-Insulting Cooling-System under Dynamic Load Conditions with a View to Fracture Safety Device Protection for Reactor Pressure Vessels	**Project Leader:** O. Seidl Dr. Bittner
Initiated (Date): 5.3.1975 **Status:** continuing	**Completed (Date):** 31.12.1975 **Last Updating (Date):** December 1975

1. General aim

In order to establish absolutely reliable characteristic data for the behavior of reactor pressure vessels during the fracturing process, a series of dynamic loading experiments was carried out.

2. Particular objectives

The experiments involve the heat-insulating cooling system. For this purpose a test set-up using lightweight insulation concrete (Bn 550) both with and without a special filling material (spherical steatit, Ø 20 mm) was constructed in model form on a scale of 1:1 and subsequently subjected to various dynamic loadings. The test Specimens were 500 and 260 mm in diameter, the test length 700 and 400 mm. Gaps of 20, 12, 6 and 0 mm were fixed in order to allow free acceleration paths.

3.1 Experimental facilities

It was possible to obtain time-deformation functions on the basis of this test project.

3.2 Research program

The load was applied dynamically by means of a system of spring-coupled masses, consisting of steel plates and 58 m-long DYWIDAG-prestressing bars (quality 110/135).

Since no experimental values for the dynamic behavior of concrete with this type of filling material were available, it was first necessary to establish limit values. These limit values were obtained by varying the specific unit pressures between the material causing the impact (steel plates) and the material sustaining the impact (lightweight insulation concrete). The specific unit pressures provided for in the test program were approximately 175 bar at the load application point, i.e. between the steel and the concrete, and 40 and 175 bar at the load removal point, i.e. from the filling material to the support structure. The loads applied ranged from 100 to 500 Mp.

The loads were applied dynamically by means of the prestressing bars and a hinged nut.

The test values were recorded with the aid of electronic measuring equipment.

The strains in the prestressed bars were determined by means of strain gages.

Carrier frequency and direct-current bridges were used for the measurements.

The dynamic test values were stored with a magnetic tape recorder (frequency range 0 - 15 kHz) and subsequently recorded via an ultra-violet printer (frequency range 0 - 13 kHz).

The load application time was 1 - 2 ms. The times required for a maximum of elastic and plastic deformations to occur range from 5 to 10 ms.

For each dynamic loading process with the associated accelerations there are at least three test recordings suitable for evaluation purposes.

4. Project status

The tests were carried out according to plan. The course of the impact through the test material was determined integrally between the load application point and the transmission of the load to the support structure with the aid of time and deformation measurements.

The recordings, in the form of ultra-violet diagrams, (time-deformation diagrams), provide exact information on the dynamic loading and unloading processes as a function of time.

The acceleration rates of the impact plate can be read off these diagrams at each point in the recordings by examining the variation in the tangent curve.

The paths of the prestressing bar ends - and consequently the movements of the impact plate as a function of time - can be determined by step-by-step integration of the measured values along the prestressing bar. The time-deformation function at the load application point is thus obtained. The measured values at the measuring cylinder indicate the load-time function.

The load-deformation function for the complete test is obtained by combining these two functions.

The final report contains a summary of the test values, including a series of detailed diagrams.

5. Next steps

It is our intention to extend the range of experiments to include the acceleration and deformation processes within the test specimens. To this end the test program is to be extended by incorporating a number of experiments which will be of considerable importance for calculation purposes. New improved electronic instrumentation will be used. Whereas previous experiments were invariably one-dimensional, plans have been made to expand the test set-up to include two-dimensional loading. Further proposals are being investigated and adopted at the present time.

6. Relation with other projects

The tests described in this report proved characteristic data for the fracture calculations on reactor pressure vessels which are currently being carried out by the Kraftwerk Union, Erlangen.

7. Reference documents

Forschungsprogramm Reaktorsicherheit Abschlußbericht Förderungsvorhaben BMFT RS 53, Voruntersuchungen zum Programm Berstsicherheit für Reaktordruckbehälter KWU, Reaktortechnik Erlangen Februar 1974.

Berstfolgeschutz aus Spannbeton für Reaktor-Druck-Behälter von Druckwasserreaktoren 600 MWe, Forschungs- und Entwicklungsauftrag BMFT RS 108 - KWU BS 006D, Bericht 1975, von Fa. Dyckerhoff & Widmann AG, München.

Dynamische Belastungsversuche und Bestimmung des Stoßfaktors für die Spannglieder des Kernkraftwerkes der BASF, Dyckerhoff & Widmann AG, München 1974.

Abschlußbericht Nr. 1 "Spezielle Materialmodelle für PISCES-Berechnungen, KWU, Jan. 1975.

8. Degree of availability

Some of the tests are being recalculated by KWU Erlangen using the PISCES program in order to compare the accuracy of the PISCES calculation for the fracturing of reactors.

Classification: 9	
Title 1 (Original Language): Funktionsversuch zum Schwungradabfall, 2. Versuch (RS 168 - II.1.6., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** KWU, Erlangen
Title 2 (english): Functional Test of an Automatic Fly-Wheel Detach at High Pump Rotation Speed	**Project Leader:** H. Schäfer
Initiated (Date): 20. 6. 75 **Completed (Date):** 31. 10. 75 **Status:** Completed **Last Updating (Date):** 31. 12. 75	

General Aim/Particular Objectives

In a functional test the detach process of a fly-wheel of a primary coolant pump was investigated in order to verify the precalculated rotation velocity where the fly-wheel should detach and to get informations of the detach-procedure and the slow down behaviour of the fly-wheel.

Experimental/Facilities/Research Program

A test equipment was built up by KWU, which imitates the original in all essential parts. As drive, a motor of a fly-wheel bunker was used.

Project Status/Progress to Date

The test equipment was built in a fly-wheel-bunker. The T.V. and measuring lines were installed. A test run was carried out and the measuring values as well as the T.V. pictures were recorded till the standstill of the fly-wheel. The results were evaluated and reportet to the IRS.

Project Status/Essential Results

The fly-wheel was detatched at the precalculated rotation velocity and was slowed down safety in the hold up equipment.
The analysis of the experimental data showed good agreement with the theoretical investigations.

Next Steps

Work on this project has been completed.

Relation with Other Projects

No relations with other projects.

Reference Documents/Degree of Availability

No reports available.

Classification 9.1, 9.2, 9.3	
Title 1 Rekenmodel voor de simulatie van transiënten in kokendwaterreactoren	Country The Netherlands Organization KEMA
Title 2 Computercode for the simulation of transients in boiling water reactors	Projectleader R.M. van Kuijk
Initiated Completed 1977 Status Progressing	Scientist P. Kloeg

1. General aim

Calculations of the dynamic response of BWR's in the case of transients and small accidents.

2. Particular objectives

Study of the control systems, capacity of safety valves, influenc influence of setpoints, heat transfer in the core, fuel temperatures, scram action.

3. Experimental facilities and programme

Computer code REBOR.

4. Project status

Operational for Dodewaard BWR.

5. Next steps

Other types of BWR's.
Comparisans with experiments in Dodewaard.

6. Relation with other projects

Not applicable.

7. Reference documents

Internal KEMA reports.

8. Degree of availability

Free on basis of exchange with other programmes.

10. CORE AND PRIMARY CIRCUIT IN STEADY STATE CONDITIONS

PROJECT TITLE : Statistical analysis of random signals for safety problems	CLASSIFICATION 10 - 14 - 8
SPONSORING COUNTRY : Italy	ORGANISATION : CNEN
DATE INITIATED : 1966 DATE COMPLETED : in progress	PROJECT LEADER : A. Federico

Description : Concern the development of methods for acquisition and elaboration of data coming from nuclear reactors and experimental loops.
These methods give the possibilities to study reactor physics, thermohydraulic and mechanical effects in normal and accidental conditions. (Possible application: early failure detection).

Facilities

- Transducers, amplifiers, filters, magnetic recorders for data acquisition.
- 1 hybrid computer EAI 8945
  1 hybrid computer EAI PACER 700
- Software for statistical elaboration running on the hybrid computer mentioned.

Reference documents

1) L.Cimorelli - A.Federico
Applications of spectra analysis tecniques to examine natural and superimposed neutronic flux fluctuations in a nuclear power reactor
Rapp. CNEN - IN(69)3 - Marzo 1969

2) A.Federico - S.Taglienti
Frequency and time-domain systems for statistical signals elaboration developed in CNEN laboratories-IAEA specialist meeting on Analysis of Measurements to Diagnose Potential Failures.
Roma, Aprile 10-11, 1972

<table>
<tr><td colspan="3">Classification: 10.1</td></tr>
<tr><td colspan="2">Title 1 (Original Language):
Anwendung statistischer Analysenverfahren in Leistungsreaktoren mit dem sicherheitstechnischen Ziel der Früherkennung von Schäden
(RS 68 A - II.4.1., Jahresbericht A 75)</td><td>COUNTRY:
BRD
SPONSOR:
BMFT
ORGANIZATION:
TU Hannover</td></tr>
<tr><td colspan="2">Title 2 (english):
Application of Statistical Analysis Methods in Power Reactors under the Safety Oriented Aspects of Early Fault Detection</td><td rowspan="2">Project Leader:
Prof. Dr.-Ing. Stegemann</td></tr>
<tr><td>Initiated (Date):
1.5.1972
Status:
continuing</td><td>Completed (Date):
28.2.1977
Last Updating (Date):
December 1975</td></tr>
</table>

1. General aim

The aim of these investigations is the development of an early fault detection system for power reactors, based on the analysis of the stochastic neutron flux fluctuations during operation and correlation with other available signals of relevant reactor parameters.

2. Particular objectives

Noise measurements in both Boiling- and Pressurized Water Reactors have to be performed in order to gain extensive experiences about space-dependent reactivity effects caused by pressure-, vibration- or coolant density feedback.
Since in many cases a suitable in-core instrumentation is not available, special effort had to be concentrated on the development of a new and reliable measuring system, using "self-powered" detectors. The application of these improved devices allows neutron flux distribution measurements with high spatial resolution and additionally reactor noise measurements without disturbance of the normal plant operation.

3.1 Experimental facilities

The interesting reactor operation parameters can be measured either by means of the existing instrumentation (pressure and temperature sensors, in-core and out-core fission chambers ...) or from additionally inserted sensors, like self-powered detectors, in-core gamma chambers and vibration sensors. After suppression of the stationary component, the fluctuating part of the signals will be amplified and filtered before analysing "on-line", using a digital correlator and spectrum display, or recording on analog magnetic tape for later processing via ADC and a digital computer. The latter method had been preferred because of the advantage to evaluate the stored data later on under different aspects.

3.2 Research program

During the report period the research program consisted of:

(a) Continuation of the measurements at the PWR Obrigheim (KWO), concerning axial neutron flux distribution and local neutron noise by means of several self-powered neutron detectors as well as pressure-, temperature and coolant flow fluctuations.

(b) Production of new self-powered detectors for the instrumentation of three in-core assemblies for the 7th cycle of the KWO power plant.

(c) Investigations in the field of self-powered detectors, with special regard to the improvement of the neutron-to-gamma signal ratio.

(d) Systematic investigations of the space dependent neutron flux fluctuations in the boiling water reactor Lingen (KWL), using different types of SPN-detectors, which could be inserted in a tube of the TIP-system and moved at any axial position in the core.

(e) Computer analysis of the recorded stochastic signals and evaluation with respect to the identification of noise sources which might be relevant for safe reactor operation.

4. Project status

4.1 Progress to date

At the PWR Obrigheim continuous measurements of the stationary axial neutron flux distribution and reactivity fluctuations have been performed during the first months of 1975 by the help of six cobalt-SPN-detectors, positioned one above the other in a central core position. Three new assemblies, each with six detectors of the Cobalt, Erbium and Hafnium-type, have been manufactured and built in into the reactor during the refueling period of the plant in July. From these

in-core sensors stationary and fluctuating signals were available and recorded weekly on magnetic tape up to now.
Since the time for computation of the characteristic fluctions in time and frequency domain and evaluation with respect to nuclear and thermohydraulic parameters is rather large compared with the measuring time, only some selected records, taken during any changed reactor operation conditions, could be analysed.
Systematic measurements of flux distributions and space dependent noise phenomena have been performed in the boiling water reactor Lingen (KWL) in October 75. A special arrangement of two Erbium- and one Hafnium-SPN-detectors had been prepared at the Institut für Kerntechnik for insertion into a dry thimble of the TIP-System. The three detectors could be moved through the core and additionally one against the other during reactor operation. Measurements and tape records from a large number of axial core positions have been analysed "off-line" with the digital computer at the institute.

4.2 Essential results

Characteristic functions like auto and cross power spectra and coherences have been obtained for a pressurized water reactor from three fuel cycles, different fuel elements (Pu and U) and several core positions. Furtheron some experience was gained about the influence of the power level on shape and structure of the noise spectra and the space dependent noise amplitudes.
The large amount of BWR in-core data, gained by the help of the three-detector-construction, permits the determination of important nuclear and thermohydraulic parameters with high local resolution and good accuracy.
From the different neutron spectral sensitivities of Erbium and Hafnium detectors, for example, the absolute thermal and fast neutron flux profiles could be calculated.
The mean local steam bubble velocity has been determined from the very beginning of bubble generation at about 45 cm from the core bottom over the whole height with a spatial resolution of 8 per cent and an accuracy of about ± 3 per cent. Comparison between the measured bubble velocity profile at a given flux profile and a parametric calculation yields the mean thermal power of the adjacent bundles. Furtheron it is possible, by introducing a recently developed theoretical model, to determine the mean local steam void content

from the space dependent noise amplitudes.

5. Next steps

Measurements at the KWO power plant will be continued during this one and the next fuel cycle with special remark to changes in the stationary signal behaviour and frequency spectra as function of different reactor operation conditions and long-time effects. Additional effort will be concentrated on the detection of subcooled boiling, which is presumed to occur in the upper region of high loaded fuel elements in PWR's.

With respect to boiling water reactors, further experiments are planned at different plants, using partly the available in-core instrumentation and special self-built SPN-detectors with improved sensitivities to permit even higher spatial resolution.

Data acquisition will be done under several aspects:

- increase of general information particularly from BWR-cores.
- measurements of steam bubble velocity profiles with better resolution and higher accuracy including comparison with theoretical values.
- investigation of the spatial detection range of the applied in-core sensors.
- further development and application of methods for the determination of local steam void content and thermal power of the fuel bundles.
- acceleration of the analysis and evaluation by development of an "on-line"-system for continuous surveillance of the dynamic behaviour of relevant nuclear and thermohydraulic parameters and consequently of a safe reactor operation.

6. Relation with other projects

see IRS - F - 24

7. Reference documents

Quarterly reports V 75/1, V75/2, V 75/3 (german)

Zwischenbericht zum Förderungsvorhaben RS 68A, Jan. 1976 (german)

8. Degree of availability

Documents are available through

Institut für Kerntechnik

Elbestr. 38 A

D-3000 Hannover 21

Classification: 10.1	
Title 1 (Original Language): Früherkennung von Fehlern und Schäden (ATT 085 A - II.4.1., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMI **ORGANIZATION:** LRA, Garching
Title 2 (english): Incipient Failure Detection	**Project Leader:** Dr. W. Bastl D. Wach
Initiated (Date): **Completed (Date):** **Status:** continuing **Last Updating (Date):** December 1975	

1. General Aim

The general aim of this research work is the development of measuring methods which permit the on-line monitoring of the mechanical status of main components in the primary system of a power reactor plant. Special attention is paid to the fact that already existent signals can be used for this purpose and that the need of additional sensors is kept to a minimum.

2. Particular Objectives

The following objectives are envisaged:

- On-line monitoring of the vibrational behaviour of reactor vessel internals.
- Detection and identification of malfunctions inside the reactor vessel.
- Detection and location of loose parts in the primary system.

3. Experimental Facilities and Program

Not relevant.

4. Project Status

4.1 Progress to Date

To realize on-line monitoring systems surveying the vibrative behaviour of reactor internals and diagnosing other malfunctions (loose parts, blockages) one has to use indirect measuring methods due to inaccessibility and safety reasons in a power reactor. This means that during power operation only external vibration and acceleration signals measured at the pressure vessel wall, neutron noise inside and outside the core and dynamic pressure signals are used as information carriers. The main problems of the indirect measuring methods are the identification and separation of signal components being representative for the vibrations of the individual reactor internals. To solve these problems theoretical models are necessary describing the coupled multi-mass system of the mechanical structures as well as the composition and transfer pathes of the signal sources in the available physical parameters. Identification methods are applicated for optimal estimations of model parameters. The verification of the mechanical vibration models is gained by extensive vibration measurements during hot functional tests without and with the core in reactor proto-types, performed in collaboration with the industry during the pre-operational test phase. At the same time correlations of component vibrations measured inside and outside the reactor vessel are determined empirically. The measured data are used as a reference state of the reactor for the on-line monitoring.

Based on experiences from investigations in the reactors Obrigheim and Stade (operational measurements), in Stade and Biblis-A (pre-operational measurements) a proto-type vibration monitoring system has been planned and installed in the Neckarwestheim PWR. With this system the advantages of the on-line monitoring will be demonstrated. Besides we shall be able to gather further operational experiences and knowledge about perturbations which are due to normal operational handlings. First measurements of the reference state of the reactor could be performed during the first hot functional tests in the end of the report period.

Special attention has been paid to improvement of theoretical models developed so far. The double pendulum model of the system core barrel/ pressure vessel was extended by considering more degrees of freedoms (3-dimensions and additional components) and an unsymmetric distribution of the spring constant of the pressure vessel supportings.

Similar investigations have been performed on the sodium cooled test reactor KNK-I. To interprete the measured vibration signals, beam-mode and shell-mode vibration models have been developed. In spite of the sophisticated vibration behaviour of the different thin-walled concentric tanks of this reactor good agreement could be obtained between theoretical and experimental results.

A further activity is directed to the detection and triangulation of loose parts inside the primary system of the reactors. Accelerometers mounted immediately on the vessel wall surface and working in the acoustic frequency range were found to be appropriate for this purpose.

4.2 Essential Results

The fixed installation of a proto-type vibration and a loose part monitoring system in the Neckarwestheim reactor is based on successful results obtained on different PWRs investigated within this project in the last years. A survey of the most important results is given in the publications /1 - 7/. Summarizing we can state:

- All investigated signals (noise signals of incore and out-of-core ionisation chambers, dynamic pressure signals, vibration signals and accelerometers both attached to the vessel surface) contain signal components being applicable for surveillance purposes.
- Horizontal core barrel movements can be monitored by neutron noise of out-of-core chambers as well as by seismic vibration gauges at the cover flange.
- Pressure signals can be used as a good measure of the driving forces which are required as the input signals in the theoretical models. In addition they contain information about internal vibrations being the response of the structure to the fluid.

- Characteristic bursts in the accelerometer signals can be related to loose parts inside the primary system. Using a binary modification of the signals the delay times between the bursts are calculable by means of normal hard-ware correlator. Then the determination of the locus of loose parts is possible using a pre-calculated normalized triangulation nomograph.

5. Next Steps

The proto-type vibration monitoring system will be available in the near future. Long time behaviour, normal operational influences and signal variations during the first fuel cycle will be investigated systematically (e.g. variations of the neutron noise due to reactivity coefficient changes). Studies are planned for the on-line processing of the data and deriving of alarm criteria indicating an abnormal reactor state or malfunctions inside the primary system. It will be investigated wether methods of pattern recognition and learning codes for the reference functions can be involved to solve these problems. The experience in internal vibration monitoring as gained at PWRs will be applicated also to BWRs. In this sense at the Brunsbüttel BWR measurements during pre-operational tests and during the first operational period will be performed. The installed external accelerometers are also used for loose parts detection during the whole starting period of the plant.

The work performed on the KNK facility will be continued with the aim to develop diagnostic systems for the SNR fast breader reactor.

6. Relation with Other Projects

See annual report 1974.

7. Reference Documents

/1/
D. Wach
Fortschritte bei der Entwicklung von Meßmethoden zur Früherkennung von Schäden in Leistungsreaktoren
Reaktortagung 1975, Nürnberg, April 1975

/2/
D. Wach
Korrelationsanalyse in der Reaktormeßtechnik
ATM, Blatt V 8238-5, Mai 1975, S. 85-90

/3/
V. Bauernfeind, L. Kasprzytzki
Untersuchung der Varianz und der spektralen Zusammensetzung diverser Schwingungssignale. Zwischenergebnisse aus der laufenden Auswertung Biblis A. Interner Bericht
MRR-I-41, Mai 1975

/4/
V. Bauernfeind, B. Olma
Untersuchung zum Schwingungsverhalten des Reaktors KNK. KNK-Versuchsprogramm. Abschlußbericht KNK-I
Gesellschaft für Kernforschung mbH. Karlsruhe, Versuchsanlagen, Bericht in Druck.

/5/
E. Sädtler
State Identification and Parameter Estimation of the Pressure Vessel Vibrations of the BIBLIS-PWR
9th Informal Noise Meeting at EIR Würenlingen, Semptember 1975

/6/
V. Bauernfeind, B.J. Olma
Experimental and Theoretical Vibration Analysis by Noise Measurements of a Liquid Sodium Cooled Reactor
3rd International Conference on Structural Mechanics in Reactor Technology, London, September 1975 (not printed)

/7/
W. Bastl, D. Wach
On-load Vibration and Noise Monitoring in Sodium Cooled and Light Water Reactors
2nd Power Plant Dynamics Control and Testing Symposium, Knoxville/ Tennessee, USA, September 1975, Paper No. 11

8. Degree of Availability

Documents are available through
Laboratorium für Reaktorregelung und Anlagensicherung
D-8046 Garching
Federal Republic of Germany

Reports MRR-I and MRR-V are confidential and therefore normally not available.

Classification 10.2

Title 1 Statisk Reaktorfysik	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 Development of calculation methods for static and quasistatic reactor physics in light water reactors	Project leader Hans Neltrup
Initiated 1968 Completed Status Last updating progressing	Scientists C.F. Højerup G.K. Kristiansen L. Mortensen H. Neltrup T. Petersen

1. General aim

Provision of static and quasistatic reactor physics information relevant for safety assesment of light water reactors.

2. Particular objective

Reliable methods for calculating -
Power shape as well within fuel boxes as overall for the reactor as function of burn up.
Reactivity as function of burn up,
Reactivity control and reactivity budgets.
Reactivity coefficients.
Stability.

3. Experimental facilities and programmes

4. Project status

1. Progress to date

a) Multi-group cross section generation from UKNDL completed

b) Multi-group $G \leq 76$, collision probability methods for fuel rods completed and coupled to multi-group diffusion theory calculations completed.

c) Few group 2D and 3D diffusion theory calculation codes, using finite difference techniques, flux synthesis and nodal theory for the reactor·core completed.

d) 3D flux synthesis coupled to hydraulic channel calculation for BWR including burn up completed.

2. Essentiel Results

Reactor physics code system verified in calculation on Yankee Rowe, Connecticut Yankee and Dresden.

5. Next Steps

1. Complete 3D flux synthesis with multichannel hydraulics for PWR (in progress)
2. Test the system further against BWR measurements - to the extent they become available.

6. Relation with other projects

Provision of cross-sections and parameters for dynamics projects.

7. Reference Documents

1. A.M. Hvidtfeldt Larsen, H. Larsen and T. Petersen. Calculation on a Boiling Water Reactor as a Test of the Risø Reactor Code Complex. Risø Report No. 268, 1972.
2. Torben Petersen. Aspects of Prediction of the Performance of a Boiling Water Reactor. Risø Report No. 289, 1973.
3. H. Neltrup and Per B. Suhr. Survey Calculation on the Hadam Neck (Connecticut Yankee) Power Plant as a Test of the Risø Reactor Physics Code System. Risø Report No. 298, 1973.

7. Degree of availability

Partly available, partly available on exchange basis.

<table>
<tr><td colspan="3">Classification: 10.2</td></tr>
<tr><td colspan="2">Title 1 (Original Language):
Dynamik großer leichtwassergekühlter Reaktoren
(ATT 085 A - II.1.2., Jahresbericht A 75)</td><td>COUNTRY:
BRD
SPONSOR:
BMI
ORGANIZATION:
LRA, Garching</td></tr>
<tr><td colspan="2">Title 2 (english):
Dynamics of Large Water Reactors</td><td rowspan="2">Project Leader:
Prof. Dr. Birkhofer
Dr. Frisch
Dr. Werner</td></tr>
<tr><td>Initiated (Date):

Status:
continuing</td><td>Completed (Date):

Last Updating (Date):
December 1975</td></tr>
</table>

1. General Aim

The general aim of this project is the development of reactor and plant dynamic models and computer programs, and reactor and plant transient analysis with respect to safety problems in large water reactors.

2. Particular Objectives

One main objective of this research project is the development of three-dimensional models of large reactor cores in order to analyze local effects during power transients. This requires the development of new computational methods in order to solve the problems with sufficient accuracy within the limits of computer memory space and computation time.

Another objective is the analysis of reactor plant transients (system analysis, analysis of anticipated transients with and without scram), requiring complex models of the entire plant as well as core models and safety systems.

3. Experimental Facilities and Program

The performance of experiments is not part of the project, however, the results of experiments and especially of light water reactor start-up tests are used for model verification purposes.

4. Project Status

The code development was concentrated on five types of models:

TYOBOX A three-dimensional code calculating space dependent neutron flux, power, temperature and coolant density in boiling water reactor cores. The program uses a coarse mesh method for solving the neutron kinetics equations. The hydrodynamic model is taken from the code TYOFUX.

QUABOX/ CUBBOX A series of programs for space-time-kinetics simulation using a new coarse mesh method for space discretisation and an implicit matrix decomposition time integration method.

ALMOS A series of plant models for BWRs with one-dimensional solution of the hydrodynamic equations and optional a point kinetics model or a space dependent solution of the neutron kinetics equations. The model includes all important plant components and the safety system.

ALMOD A plant model for PWRs similiar to the BWR model ALMOS.

KEMOL A core model for light water reactors (BWRs and PWRs) with one-dimensional hydrodynamics and optional zero- or one-dimensional neutron kinetics. For dynamic hot channel calculations the model contains several parallel coolant channels.

4.1 Progress to Date

Program development and testing for TYOBOX is completed. Program description is published /1/.

For the QUABOX/CUBBOX models further improvement in computation time for the dynamic solution was obtained by a more efficient time integration method. The development of the program QUABOX was completed and the program is in use at several institutions in Germany and the US. Literature on the method used in QUABOX is available /3/.

There is further progress in the development of the program CUBBOX (cubic approximation of the local flux distribution within a box). It has been verified that CUBBOX permits larger box sizes than QUABOX, especially near and in the reflector.

The BWR plant model ALMOS had already been completed in 1974. Only minor changes, mainly concerning more convenient handling of the program have been done recently. ALMOS was in use frequently (see 4.2).

For the PWR model ALMOD the program development of the single components (core, piping, pressurizer, steam generator) is essentially finished. Test runs with single components have been carried out.

The one-dimensional light water reactor core model KEMOL, derived from the ALMOS code and extended to pressurized water reactor conditions, has been finished. Test runs included ATWS-calculations for a PWR.

4.2 Essential Results

Generally, it has been found, that with xy-mesh sizes corresponding to the fuel element structure of modern LWR's (15 - 25 cm) and 20 - 30 cm in the axial (z)-direction, quadratic approximations (QUABOX) yield accuracies comparable to FD-calculations using a 5 cm mesh, and cubic approximations (CUBBOX) yield accuracies comparable to FD-calculations using a 2 - 3 cm mesh. Per space point and time step the QUABOX version requires about 1.5 times the computing time of FD-codes, and the CUBBOX version about twice the computing time of FD-codes. Thus, for static solutions, which require about the same number of time steps (or iterations) a computing time reduction (relative to the above indicated FD-calculations) by a factor 40 - 300 is reached for 3d-cases. For time-dependent cases, the computing time reduction is even greater: Since the temporal discretization error always depends on inverse powers of the spatial mesh size, the described method generally permits larger time steps at comparable accuracy. Thus, for example, a 3d-quarter core simulation of a rod ejection accident in a large BWR using adiabatic heatup and doppler feedback model and following the transient from cold reactor initial state over first maximum, first minimum, second maximum to about three times the peak time requires less than 15 min cpu time on a CDC 7600 or IBM 360/195.

With the BWR plant model ALMOS numerous applications were carried out such as:

- Comparison of point kinetic and one-dimensional calculations for a feedwater disturbance, showing the importance of space-dependent models.
- Comparison with start-up test measurements as part of the model verification.
- A complete ATWS study.
- Extended sensitively studies in order to analyze the feedback mechanism during pressure transients.

Results are published in /2, 3/.

5. Next Steps

Coupling of the QUABOX/CUBBOX program with the hydrodynamic part (modified COBRA III/c) of the MIT developed MEKIN code will be the first step toward a 3d-core model. In parallel action an advanced fluiddynamic model will be developed which will replace the fluid-dynamics used in COBRA III/c at a later time.

With the plant model ALMOS further sensitivity studies and comparisons with other models will be carried out. In addition the model will be used for the investigation of other component failures than the fast shut-down system. After the completion of the PWR plant model ALMOD, further test calculations such as comparison with start-up tests at the power plant Biblis A will be carried out. Furthermore the model will be applied in ATWS analysis.

First steps toward 3-D plant models are planned, because for certain transients, such as a steam line rupture at a PWR, a 3-dimensional simulation of a neutron kinetics is necessary within the plant model.

6. Relation with Other Projects

Investigations with similar objectives with respect to core simulation are carried out at KWU (RS 26, RS 178).

Results of experimental projects are used for code development and verifications (RS 64, RB 455).

7. Reference Documents

/1/
S. Langenbuch, W. Maurer, A. Schmidt
TYOBOX-STATIONÄR - Dreidimensionales SWR-Kernmodell mit Brennstoff- und Moderatorrückwirkung. Programmbeschreibung
MRR-P-19, November 1975

/2/
P. Peternell
ATWS - Rechnungen für KKB mit ALMOS. Interner Bericht
MRR-I-57, Dezember 1975

/3/
Quarterly reports in the series IRS-Forschungsberichte

8. Degree of Availability

Documents are available through
Laboratorium für Reaktorregelung und Anlagensicherung
D-8046 Garching
Federal Republic of Germany

Reports MRR-I are confidential and therefore normally not available.

Classification 10.3

Title 1	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 SDS, a thermohydraulic subchannel programme for steady-state analysis of water-reactors	Project leader A. Olsen
Initiated: May 1970 Completed: Autumn 1974 Status: Code made, verification continues. Last updating: Autumn 1974	Scientists: F. Cortzen V.S. Pejtersen

1. General aim

To provide a general subchannel code, for the analysis of thermal-hydraulic, steady-state conditions in water-reactors.

2. Particular objectives

The objective is to make a calculation model (code) which can predict, based on sound physical models, local void, mass flow, enthalpy and burn-out margin for safety evaluation and design.

3. Experimental facilities and programme

The experimental programme has been executed concurrently with the code-development to supply data on local phenomenas in fuel-elements with respect to thermo-hydraulics.

Experiments were made on a 7-rod and a 9-rod BWR geometry, as well as on (concentric and excentric) annulus geometries.

4. Project status

1. Progress to data

A FORTRAN 4 code, exists, capable of solving, on a subchannel basis, steady-state thermal hydraulic problems, for BWR's and PWR's. Specifically predicting void- and mass-flow-distributions on a fuel-box or core-basis. (No limit on number of subchannels,so far up to 400 has been tested).

2. Essential Results

On the modelling side a substantial improvement have been made with respect to the formulation of the subchannel equations, physically, mathematically and codewise. This in turn has made it possible to simulate, on a sound basis, hitherto not approachable experiments, such as flow-blockages.

5. Next step

Physical models and correlations must be verified in respect to new and further experimental data.

6. Relation with other projects

The basic subchannel project was a joint venture of three Scandinavian organizations: Danish AEC, AB Atomenergi, Sweden, and Institutt for Atomenergi, Norway. The nordic co-operative project was completed 1973, and the present Danish project is an extension of the work.

7. Reference documents

Z. Rouhani: Review of Momentum balance in Subchannel Geometry
Eur. Two-phase flow meeting (ETPFM), Brussels 1973

J. Bosio, O. Imset: Two-Phase flow investigations in a 7-rod bundle.
ANS European Meeting,Karlsruhe, October 1973.

8. Degree of availability

Not available.

Classification: 10.3	
Title 1 (Original Language): Mischungseffekte bei parallel durchströmten Kanälen (RS 81 - II.1.3., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: EURATOM, Ispra
Title 2 (english): Mixing Effects in Parallel Channels with Water Two-Phase Flow	**Project Leader:** H. Herkenrath, W. Hufschmidt
Initiated (Date): 1.1.1975 — **Completed (Date):** 31.12.1977	
Status: continuing — **Last Updating (Date):** December 1975	

1. General aim
The investigation of the interaction and mixing effects between adjacent subchannels of a fuel rod bundle cooled by boiling water is aimed at providing experimental data on the local distribution of mass-flow, enthalpy and void-fraction so as to achieve a better thermohydraulic description and more precise DNB margin for a BWR core. Furthermore, studies in steady-state conditions are a necessary preliminary to studies of the mixing process in transient conditions such as those occurring during power transients or blow-down. Therefore, the first phase of the programme will be devoted to steady-state and the transient conditions will be studied later.

2. Particular objectives
The mixing programme was started in the frame of collaboration contracts between CNEN (Rome) and BMFT (Bonn) on the one side and Euratom on the other. These contracts involved the studies of the boiling mixing phenomena using two different test sections:
A two-channel test section for fundamental studies (pressure up to 150 bars; high-pressure water loop PRIL) and a 16-rod cluster test section with BWR geometry (pressure 70 bars; high-pressure water loop BOWAL). Later, the BOWAL studies will be extended to a pressure of 160 bar using PWR geometries. After first operational tests with the two-channel test section in the PRIL loop at the beginning of 1974, this part of the programme was interrupted and effort was concentrated on various preparations for the 16-rod bundle test in BOWAL.

2.1. Experimental facility

The high-pressure water loop BOWAL, to be used for the boiling mixing experiments with the 16 rod cluster test section, has been described in detail in / 1 /. Meanwhile, the total power available for the test section has been increased to 3.6 MW. The loop modifications needed for the tests are mainly concerned with the special measuring facilities required to monitor the mixing process in the test section. The latter, which is being constructed by CISE-Milan, has 16 rods of 15.0 mm OD on a pitch of 19.3 mm assembled in a square array of 80.3 mm and having about 3 m heated length. To measure the enthalpy exchange between the different subchannels at the outlet of the test section, 5 subchannels will be sampled simultaneously using a sophisticated flow splitting device with appropriate taps for the measurement of the static pressure at each subchannel exit. The enthalpy of the steam-water mixture at the outlet of each sampled subchannel will be obtained from the measured heat-balances of different condensors. During sampling, the difference in static pressure between similar subchannels will be set to zero (isokinetic method) by means of sensitive regulation valves in the pipes carrying the single phase (water) flow beyond the condensors. The subchannel mass-flow rate will also be measured in these pipes by means of diaphragms. For stability reasons the condensators will be cooled on the secondary side with high-pressure water at lower temperature and a supplementary heat-exchanger has been installed. From the measured heat-balances of the different condensors, the outlet quality of the steam-water mixture in the subchannels can be calculated. As this calculation involves considerable error when the steam quality is small, it is planned to install a device for measuring the void-fraction, and hence the steam quality, at the outlet of the test section. In the case of the transient conditions to be studied later, (the BOWAL loop is capable of partial blowdown) the calorimetric method for the determination of the outlet enthalpy will fail because of the transit time delays between the test section and the condensors. In this case special voidmeters become indispensable.
One system which has been proved suitable for the determination of the void-fraction in two-phase flows is the absorption method, using X-rays or Gamma-rays. Theoretical comparisons of these two methodshave shown that, in the present case, X-ray voidmeters are more appropriate. Detailed calculations led to a X-ray tube with an energy of 30 KeV (power 600 W). This should be able to determine the void-fraction over the whole range with a precision better than 5% and a time constant of about 10 ms.

3.2. Research programme

Experimental investigation of subchannel interactions in two-phase flow conditions (mixing).
The original program provided the following studies:
- Steady state measurements of the subchannel interaction in two-phase flow conditions with a two-channel test section of different geometries for fundamental studies and with a 16-rod bundle test-section in BWR-geometry.
- Investigation of the mixing phenomena in transient conditions for partial depressurization with the above mentioned test-sections.

Due to personnel reasons in the JRC Ispra the fundamental studies with the two-channel test-section have to be postponed and a new program has been set up concentrating on the investigations with 16-rod clusters. The fundamental studies will be resumed only in case of necessity for the interpretation of the test results with the bundle experiments.
The actual program covers
- steady state measurement of mixing effect with a 16-rod cluster in BWR-geometries (70 bar)
- mixing studies in transient conditions with 16-rod clusters in BWR- (70 bar) and PWR- geometries (160 bar).

4. Project status

4.1. Progress to date

The whole activity in this field has been concentrated on the modification of the high-pressure water loop BOWAL in three main parts:
- increase of power supply from 2.4 to 3.6 MW
- installation of special loop elements for the mixing studies (condensers, heat exchangers, valves, diaphragms etc)
- adjustment of the measurement device (temperatures, pressures, mass-flows and void-fractions etc)

The first two items have been completed during 1975. The third item will be terminated in the early beginning of 1976.

4.2. Essential results

The theoretical work carried out has been concentrated on the development of the necessary measurement techniques, especially that for the void-fraction where rather little experience existed. The result of this study (published in / 2 /) has shown that in the present case a X-ray void-meter is appropriate. During 1975 two X-ray devices have been tested on an optical bank and different troubles could be removed.

5. Next steps
   - Completion of the whole installation of loop BOWAL,
   - Calibration of the X-ray void meters on the optical bank as well as under real operational conditions in a calibration loop (PRIL),
   - Pretests of the whole components of loop BOWAL for the mixing studies,
   - Steady state measurements with a 16-rod cluster test section in BWR geometry (70 bar)
   - Development and construction of the 16-rod bundle in PWR geometry (160 bars) together with CISE, Milan.

6. Relation with other projects
   RS 109 Experimental investigation of the influence of PWR loops on blow-down.

7. Reference documents

/ 1 / Herkenrath H., Mörk-Mörkenstein P.
The pressurized and boiling water loop of the technology division at Ispra - EUR 4683 e (1971) (English)

/ 2 / Herzberger P., Hufschmidt W., Wind F.
Die Verwendung von Gamma- oder Röntgenstrahlen zur Messung des Dampfgehaltes in Zweiphasenströmungen
EUR 5415 (1975) (German)

8. Degree of availability
   Reports mentioned in 7 are not classified and are on sale at the office for official publications of the European Communities 37, rue Glesener, Luxembourg.

Classification 10.3

Title 1 Control of Dryout in Boiling Clusters	**Country** UK
Title 2	**Sponsor** UKAEA **Organisation** AEA Winfrith
Initiated 1966 **Completed** **Status** Continuing **Last updating**	**Project leaders** J. Obertelli

1 General Aim

To control the conditions for a boiling-type reactor so that dryout is sufficientl unlikely to occur; this implies a knowledge of the consequences of dry-out.

2 Particular Objectives

To observe dryout in an out-of-pile rig under conditions representative of the reactor.

3 Experimental Facilities

The 9 MW electrically heated rig capable of taking a full-sized bundle.

4 Project Status

Dryout has been studied as a function of the bundle geometry, flow etc.

5. Next Steps

Further measurements to optimise power distribution and bundle geometry, and the effect of a continuously-running spray.

Reference Documents

Internal documents.

Obertelli and Owen. AEEW-M935 (Unclassified)

	Classification 10.3
Title 1 Burnout experiments in BWR round tubes	**Country** Italy **Sponsor** CNEN **Organisation** CNEN-CISE
Title 2	
Initiated 1.1.75 **Completed** 30/9/75 **Status** planning **Last updating** Dec. 74	**Project leaders** V.Marinelli

1. General aim

In order to optimize the burnout correlation for a rod bundle it is necessary to have accurate reference to rounde tube data. Afterwords the rod bundle and mixing effects can be taken into account.

2. Particular objectives

Accurate round tube CHF correlation.

3. Experimental facility and program

A set of 300 point data will be obtained in the CISE "Ieti-I" plant on two tubular sections 600 cm long and having diameters of 1.26 and 1.7 cm at pressures of 70,50,30 bars and specific mass flow rate between 12 and 200 $gr/cm^2 sec$.

4. Project status

1) Progress to date The installation of the test sections has been initiated and test data matrix has been decided.

2) Essential results none

5. Next steps

First preliminary experiments.

6. Relation with other projects

The experimental results and the correlation will serve as basis for the burnout experiments in BWR fuel elements.

7. Reference documents

None

8. Budget

35 M/it. / Personnel: 40MM

PROJECT TITLE : Burnout experiments in BWR fuel elements	CLASSIFICATION 10.3
SPONSORING COUNTRY : Italy	ORGANISATION : CNEN-CISE
DATE INITIATED : 1972 DATE COMPLETED : May 1976	PROJECT LEADER : V. Marinelli (CNEN) F. Lucchini, G.Caspari (CISE)

Description :

1. General aim

Development of burnout correlations based on the critical quality-boiling length concept, utilizing both the bundle parameters and the subchannel parameters.

2. Particular objectives

Measurements of CHF data on BWR 16-rod test sections, 12 ft long, having different radial subaxial flux distributions.

3. Experimental facilities and programme

The experiments are done in the CISE IETI-III loop of 8 MW, in the following range of parameters: pressure 70 ata, specific flow rate G=12-200 g/cm^2 sec.

4. Project status

4.1. Progress to date

The measurements have been completed obtaining data from 6 different test sections.

4.2 Essential results

Critical quality-boiling length parameter correlate well the data on a bundle parameters basis and a new calculation procedure, named ACHAB method, has been developed to predict the CHF margin in terms of power ratio.

./.

5. Next steps :

The analysis of the data will be completed . It has been recognized the necessity of obtaining new round tube CHF data in order to optimize the rod-bundle CHF correlation. This new CHF tests will be done on round tubes having the same heated diameters as the rod-bundle subchannels.

6. Relation with other projects

CHF round tube experiments will provide data to be used in the theoretical developments of this programme.

7. Reference documents

G.P. GASTONI et al. "Experiments on the Influence of Radial Heat Flux Distribution on Critical Heat Flux in BWR Bundle Geometry" - Reactor Heat Transfer Meeting of Karlsruhe - October 9-11 1973.

V. MARINELLI at al. "Dryout Experiments in a 16-rod BWR Geometry with Six Different Radial Test Flux Distributions" - 1975 Heat Transfer Conference - S. Francisco , Cal. - August 11-13 1975

8. Degree of availability

Open

9. Budget

115 M Lit. Personnel 100 MM.

Classification

10.3 (1.1.2.)

Title 1 Basic studies of two phase mixing in fuel cluster geometries	Country: JRC Sponsor: CEC Organization: JRC ISPRA Establishment
Initiated : 1973 Completed : 1977 Status : progressing Last updating : March 1975	Project leader: H. Herkenrath

1.) General aim

Basic studies of thermohydraulic mixing between the subchannels of a boiling fuel cluster.

2.) Particular objectives

To examine boiling mixing (steady state and transient conditions) in a 16 rod cluster (BOWAL loop). Validation of cluster thermo-hydraulics codes. Calibration of different void meters for steady state and transient conditions (PRIL loop).

3.) Experimental facilities and programme

(a) PRIL loop

This loop consists of :
- controllable forced circulation;
- partial blowdown (~ 20%) facility for the transient tests;
- pressurizer, condensor, subcooler.

The characteristics of PRIL are :

Maximum power	: 600 KW
Maximum pressure	: 200 bar
Maximum temperature	: 365°C
Special Instrumentation	: different void meters

(b) BOWAL loop

This loop consists of :

- controllable forced circulation;
- partial blowdown (20%) facility for the transient tests;
- pressurizer, condensor, subcooler and preheater (500 KW).

The characteristics of BOWAL are :

Maximum power	:	3.6 MW
Maximum pressure	:	250 bar
Maximum temperature	:	365°C
Special Instrumentation	:	distribution of temperature and power along a number of the rods; pressure distribution along the test section. Outlet enthalpy distribution among the 6 basic subchannels of the cluster.

4.) Projects status

1. Progress to date : The modification of the BOWAL loop is underway and will be finished in 1975. The 16 rod cluster test section (PELCOS-CISE) has been mounted in the loop.

2. Essential results : To be reported when available.

5) Next steps : See item 4 above

6) Relation with other projects : -

7) Reference documents : JRC 1974 safety programme progress report.

8) Degree of availability : Freely available

9) Budget : The expected total investment from the CEC is 260 000 UA which includes the modification of the loop (BOWAL) and the power unit, and the running costs. CNEN and BMFT are contributing 330 000 UA in the frame of collaboration contracts.

10) Personnel : 7 men/year

11) Additional Information : -

Classification: 10.4	
Title 1 (Original Language): Vorhaben Körperschallmessungen an Reaktordruckbehältern und Primärkreislauf von Kernkraftwerken (RS 97 A - II. 3.5, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: ALLIANZ GmbH
Title 2 (english): Solid-Borne Sound Measuring on Reactor Pressure Vessels and on Primary Circuit of the Cooling System	Project Leader: Raible
Initiated (Date): April 1975 — Completed (Date): April 1977	
Status: continuing — Last Updating (Date): December 1975	

1. General aim

The object of this project is the detection of impending damage to the core components and in the primary circuit of the cooling system at an early stage. An essential feature of this measuring system is that the transducers are located outside of the reactor pressure vessel and the remaining cooling system, and that information can thus be obtained on the behaviour of the core components and the primary circuit during operating, with simultaneous monitoring thereof.

2. Particular objectives

The particular objectives and main emphasis of the work are:

a) Study of parameters on an experimental loop for a Leak Detection and Monitoring System based on acoustic emission.

b) Development of an enlarged Loose Parts Monitoring System (LPMS), with which loose and loosened parts can be detected and located.

3.1 Experimental facilities

For the study of parameters, the Kraftwerk Union Erlangen has made available an experimental loop, which makes it possible to simulate leaks under specific reactor conditions. The solid-borne sound noises are recorded on magnetic tape and then analysed off-line with the

aid of auto and cross correlation. If the propagation velocity of the noise in the material is known, the location of the leak can be determined by cross correlation.
For testing the loose parts monitoring system, the burst signals of loose parts from the Stade Power Station recorded in 1973 are used. As, in this case, the locations of both the transducers and the loose parts are known, an algorithm for the triangulation can be found.

3.2 Research program

The research program for the study of parameters for the leak monitoring system embraces a total of thirty experiments:
Variation of the following parameters is planned:

Three types of leaks:	leaking flange, leaking valve, crack
Two leakage rates:	24o l/h, 24 l/h
Three enthalpies:	steam, water/steam, water
Underground noise:	yes/no

Investigations will be carried out to determine in which manner different types of leaks are discernible in solid-borne sound (noise analysis, correlation, power spectra, delay-time measuring).

4. Project status

4.1 Progress to date

In cooperation with the Kraftwerk Union Erlangen and Battelle-Institut Frankfurt, the research program was elaborated. The Kraftwerk Union has been commisioned to prepare the experimental loop for measuring and to set-up and produce the different types of leakage.

4.2 Essential results

Essential results are not yet to hand.

5. Next steps

It is planned to conduct the experiments in two stages. The first measurements will commence in April 1976. The results of the subsequent evaluations will then decide further procedure.

6. Relation with other projects

Battelle-Institut Frankfurt:

"Pre-Experiments for Leak Monitoring by means of Acoustic Emission Analysis".

7. Reference documents

None

8. Degree of availability

None

<table>
<tr><td colspan="3">Classification: 10.4</td></tr>
<tr><td colspan="2">Title 1 (Original Language):
Modellmäßige Behandlung der Schwingungen von Kern-einbauten in Leichtwasserreaktoren
(RS 0066 A - II.4.1., Jahresbericht A 75)</td><td>COUNTRY:
BRD
SPONSOR:
BMFT
ORGANIZATION:
University Erlangen-Nürnbg.</td></tr>
<tr><td colspan="2">Title 2 (english):
Model Calculation of Vibrating Core Components in Light Water Reactors</td><td rowspan="2">Project Leader:</td></tr>
<tr><td>Initiated (Date):
March 1972
Status:
continuing</td><td>Completed (Date):
December 1977
Last Updating (Date):
December 1975</td></tr>
</table>

I. General aim.

Apparatus and plants using large quantities of fluids are often damaged by flow-incluced vibrations. It is not possible by theory to gain information concerning the best way of laying out a mechanical structure to reduce or eliminate vibrations. Under real conditions the flow pattern itself and the geometry defined by the mechanical structure are very complicated. It therefore remains for the experimental engineer to explain the relationship between the movement of the mechanical structure and the excitation caused by the flow. For a systematic investigation in the planning stage experiments making use of models can also be carried out complementary to measurements taken at the plant itself. Model technology is justified theoretically in mechanics of similarity.

II. Particular objectives

Under investigation are the flow-induced vibrations of the core components in the reactor pressure vessel of light-water-reactors caused by the flow of the main coolant. The measurements are taken on a pressure vessel model (scale 1:10) of normal construction (4-loop-plant). Figure 1 gives a diagrammatic representation of the investigation-program.

Using only the hydrodynamic analogy of the flow, the structure of the model should be rigid, because it doesn't correspond in its elastic properties to the actual installation. There are three dimensionless numbers enabling a transfer of measurements taken at the model to the actual reactor-pressure-vessel. An additional requirement is the observance - in a similar way - of the hydrodynamic boundary conditions. In this model it is possible to investigate fluid-dynamic and fluid-resonant feedback - when ever they exist in the hydrodynamic system.

If one takes into accout not only the hydrodynamics but also the kinetics of the core components one obtains a hydroelastic similar model. This model allows the investigation of fluid-elastic feedback between structure and fluid. The feedback mechanism is important since it will excite the core components to vibrate periodically. The difficulty of this model is that six dimensionsles numbers and the boundary conditions of the flow and the structure must be fulfilled. As the flow - induced excitation of the structure is of stochastic nature - whenever there exist no dominant feedback mechanisms - one needs statistical procedures for the analysis of the measured signals.

III.1. Experimental facilites

A.) Experiments conducted on the hydrodynamically similar model
A geometric scale-model of the pressure vessel with core components is constructed. The experiments are conducted in open circuit with air as flow medium. The points at which measurements are taken are where the flow medium enters and leaves the model, in the ring gap between the pressure vessel and the core barrel, on the lower core support, on the simulated fuel elements and on the control rod guide assemblies. Dependent on various parameters the pressure fluctuations on the core components are measured by means of pressure transducers.

B.) Experiments conducted on the hydroelastically similar model
This model corresponds not just in its hydrodynamic similarity but even in its elastic features to the actual installation. The experiments conducted in this model are carried out in closed circulation with water as the flow medium.

Figure 2 shows the instrumentation of the hydro-elastic model. Three main groups of variables will be measured:

1.) vibrations, i.e. displacements and accelerations, of the structure
2.) deformations of the structure due to the vibrations and
3.) pulsations of hydrodynamic pressure, which induce the structure to vibrate.

III.2. Research program

The representation of the measured data shall be made in a dimensionless, semi-empirical mathematical model. To do this the input of the system - flow velocity and external pressure pulsations - must be correlated with the relevant output of the system - movement of the structure. The main goal is to get a criterion for a safe design of the core components for a pressured-water-reactor of this type. Therefore it is of interest to get theoretical realtionships, which are representative for the above-mentioned input-output correlation. To asshure the validity of these results it is necessary to compare the measured signals of the model with those of the original reactor.

IV. Progress status

IV.1. Progress to date

a) completion of the signal pick-up and the data processing equipment including the software for the analysis of stochastic signals
b) completion of the core assembly and experiments with the hydrodynamic model. The flow medium being air
c) planing of the hydro-elastic model; preparing of drawings and start of production of components of this model
d) analysis of original data from a nuclear power station
e) implementation and testing of computer programs to calculate vibrations - free and induced - of the structure by lumped-mass and finite element techniques.

IV.2. Essential results

The experiments with the hydrodynamic model will be concluded at the end of 1976 and subsequently published. Figure 3 shows signals of pressure pulsations, which were measured in different positions in the ring gap between core barrel and pressure vessel. By transfering the data thus obtained to the actual design - according to he similarity numbers - one obtains pressure fluctuations of about 0,25 bar. Figure 4 shows a general plan of the hydro-elastic model. The design data are

main coolant flow	4 x 180 m^3/h of water
temperature range	30 upto 120 ^oC
pressure	16 bar
total head of the pumps	70 m H_2O
electrical power	4 x 45 kW
max. flow of cooling water	4 x 4 m^3/h

The copy of the reactor-pressel-vessel with core components as a hydro-elastic model is very complicated and laborious. Experiments will start in the midst of 1976.

V. Next steps

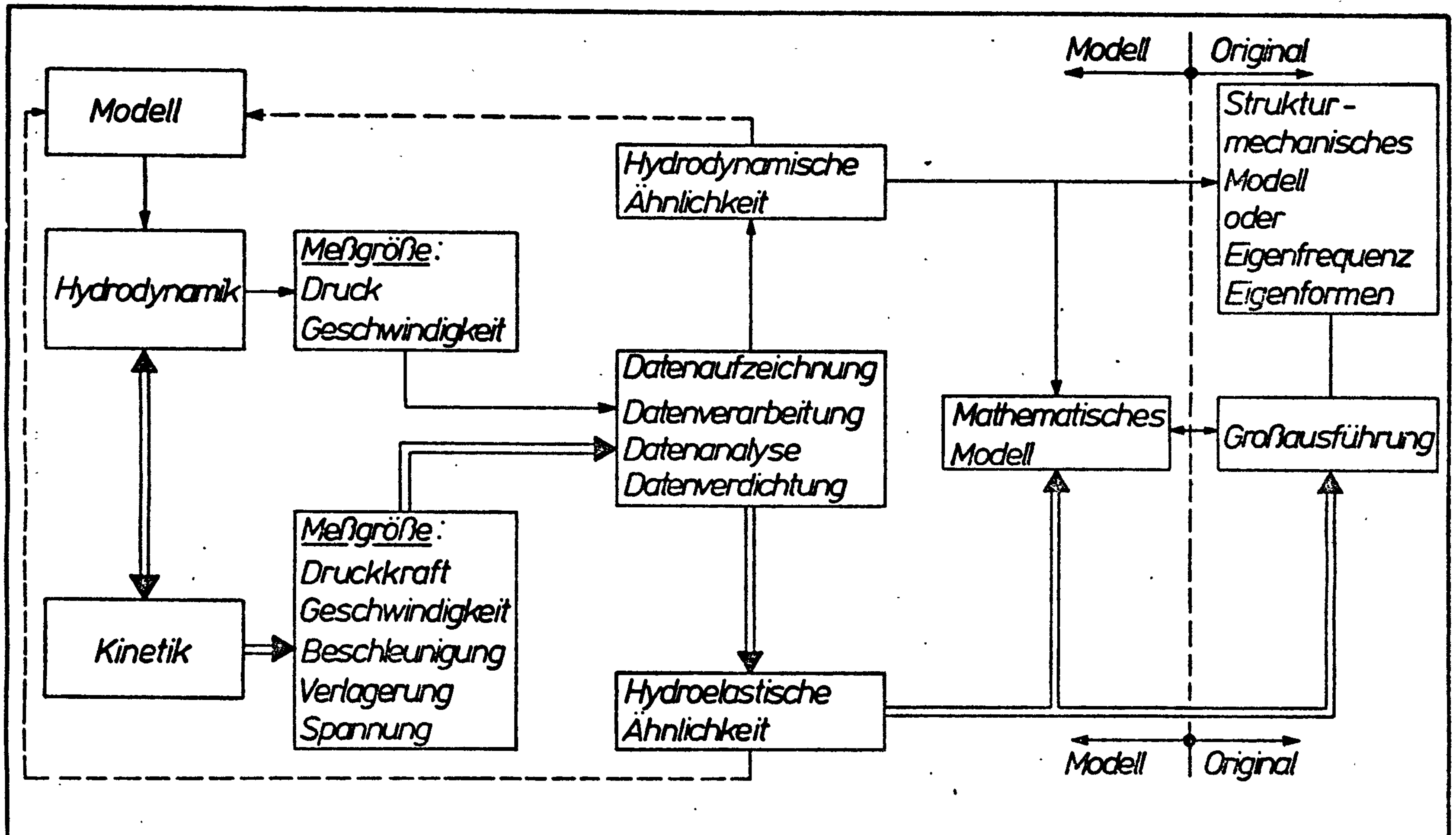
See illustration 5

VI. Relation with other projects

See this report for 1974

VII. Reference documents

Annual reports for the IRS (German)



UEN AA	Übersicht: Forschungsvorhaben RS 66 Modellmäßige Behandlung d. Schwingungen v. Kerneinbauten i. Druckwasserreaktoren	RS 66

Abb. 1

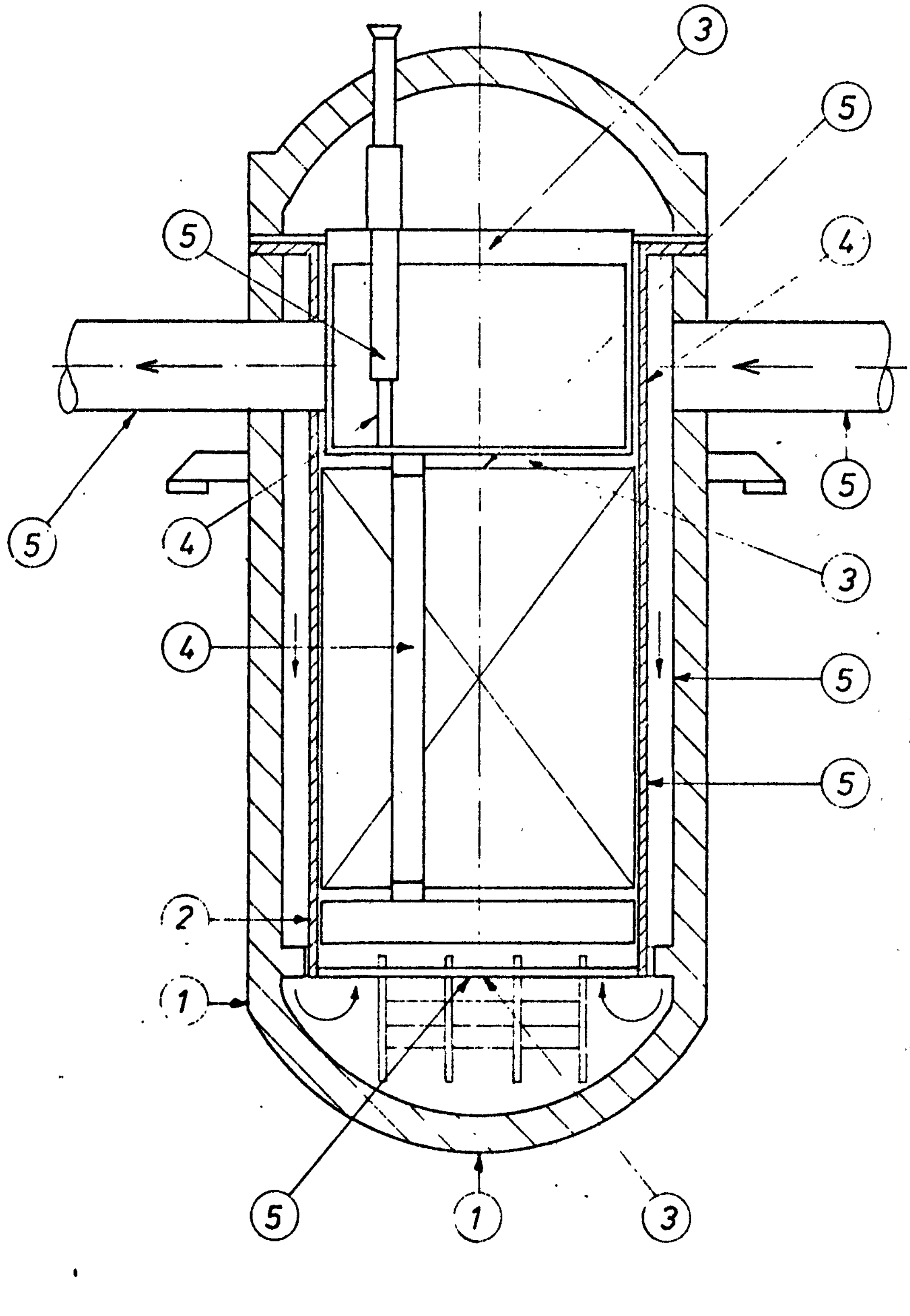
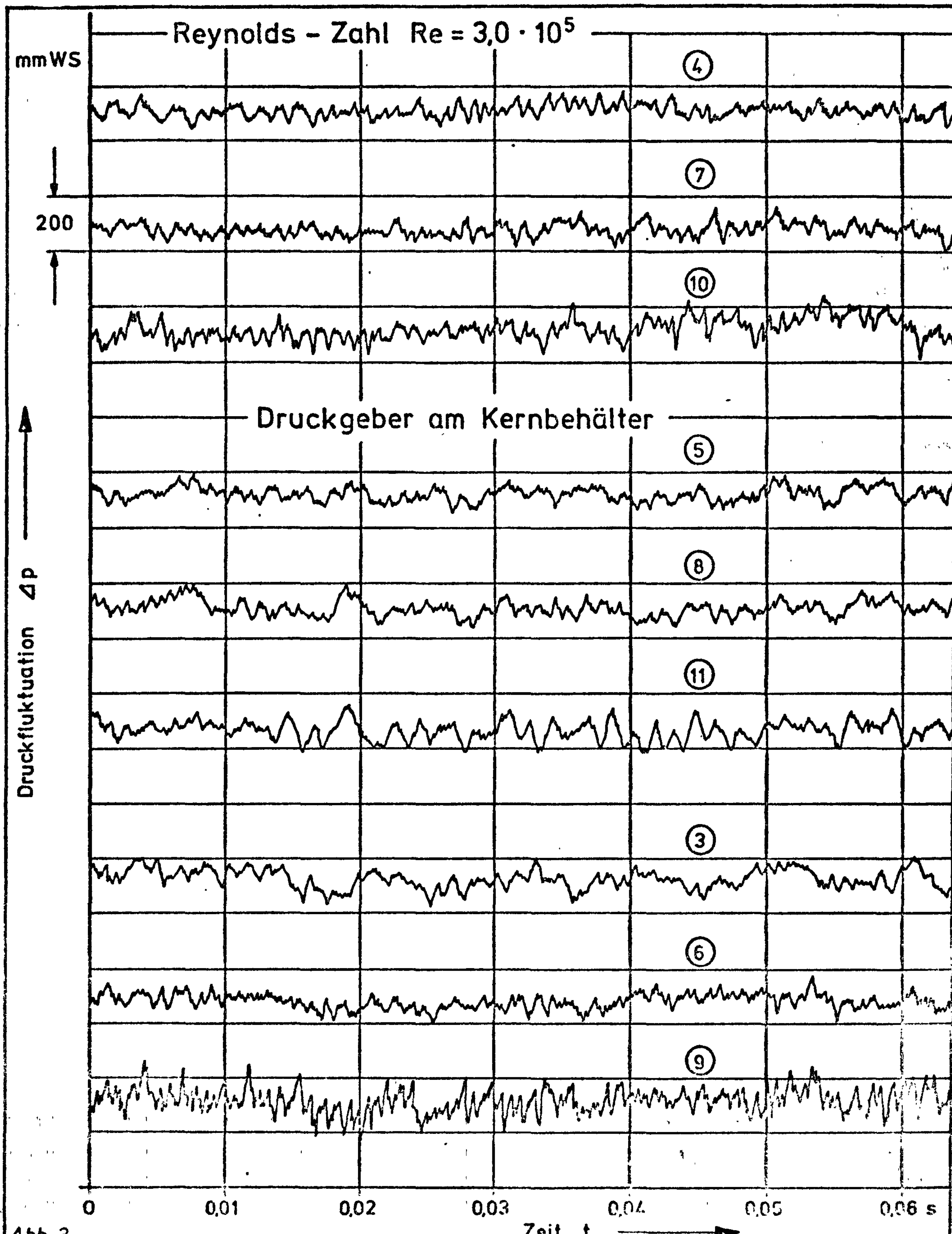


Abb. 2: Meßstellen im Druckgefäß


Reynolds - Zahl Re = 3,0 · 10^5
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Druckgeber am Kernbehälter
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Abb. 3

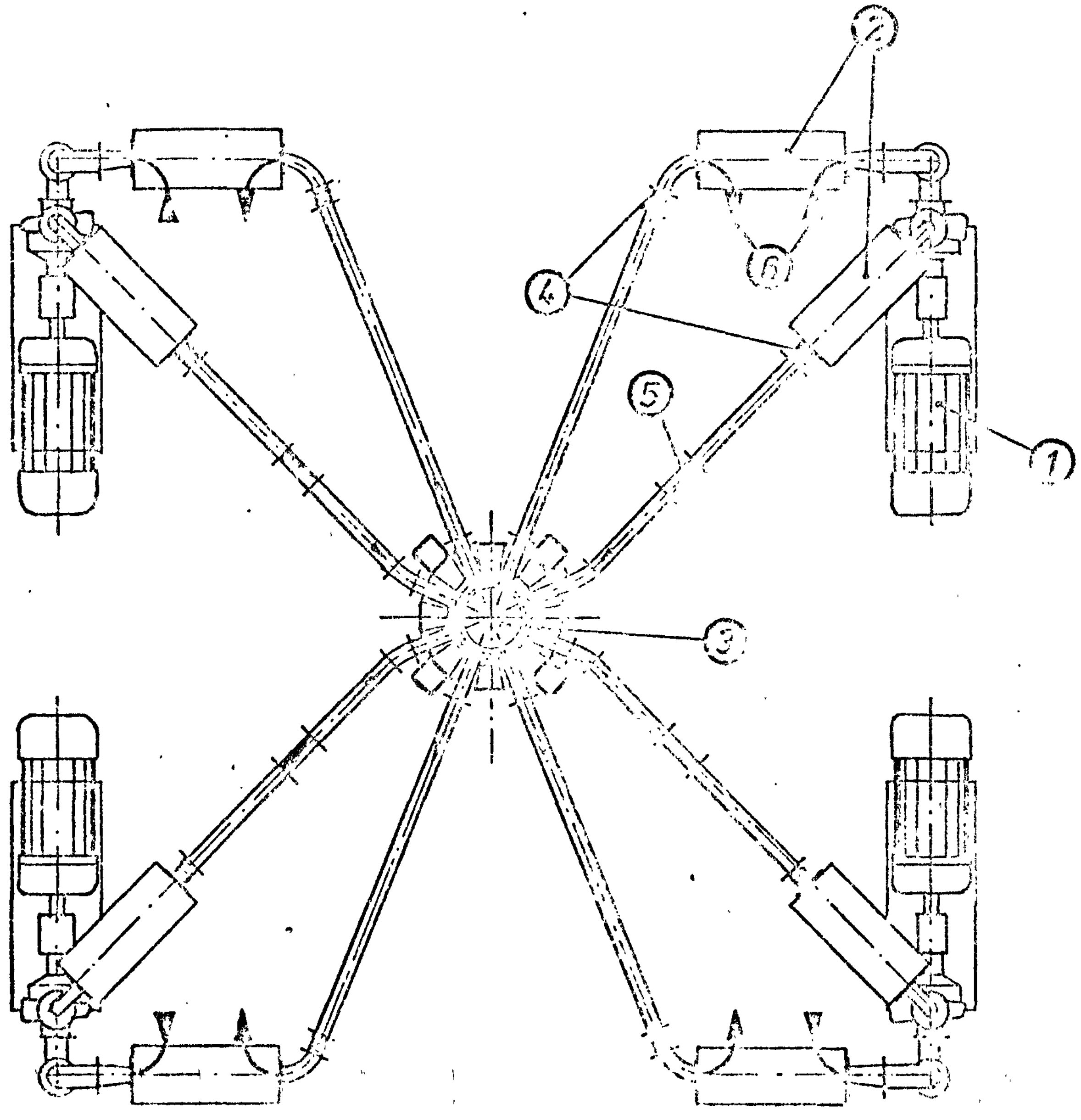
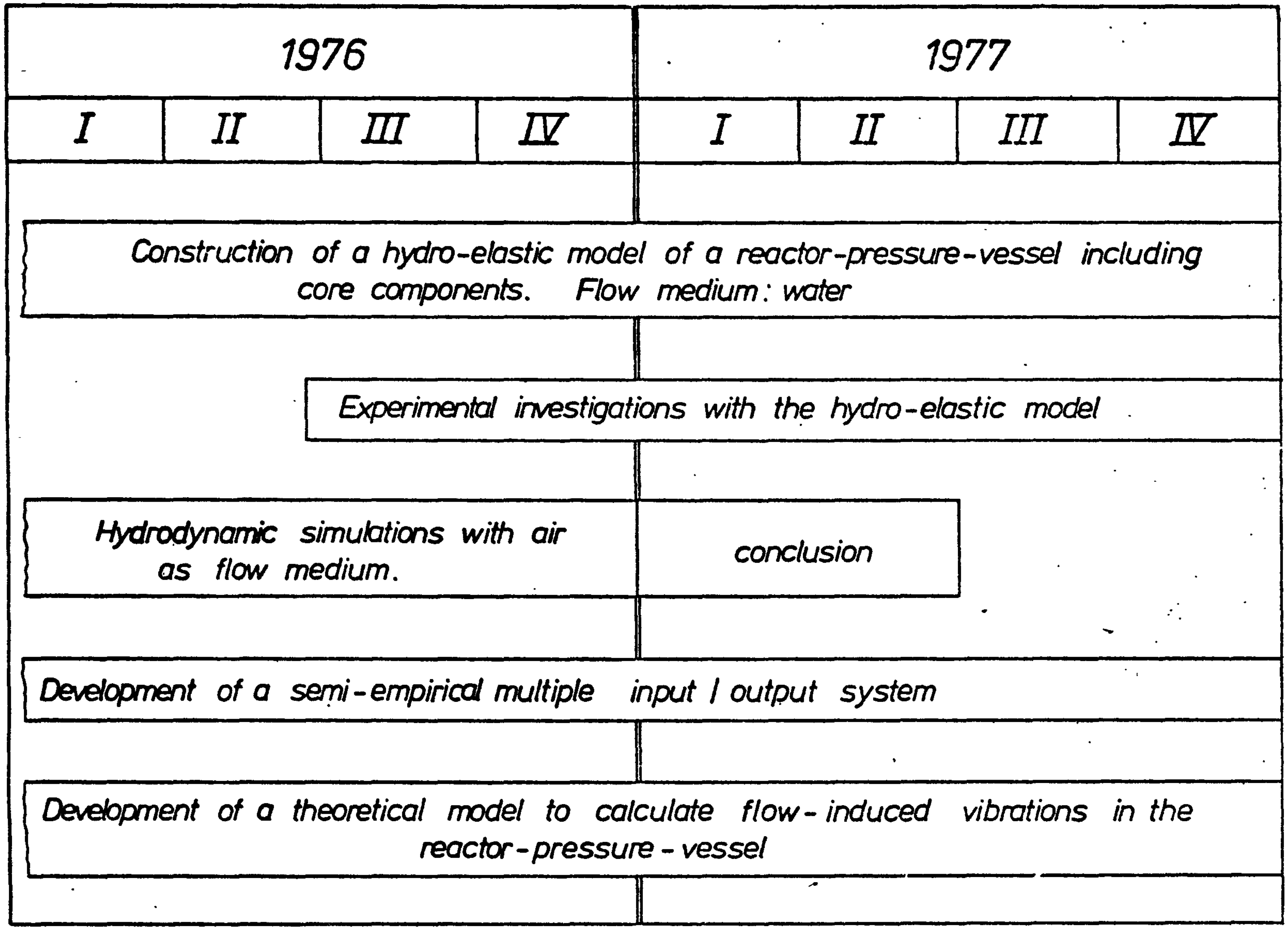


Abb.4: Primärkreislauf - Modell eines Druckwasser - Reaktors



Schedule for the research project RS 66

Abb. 5

PROJECT TITLE : Vibration Behaviour of Core Components	CLASSIFICATION 10.4
SPONSORING COUNTRY : ITALY	ORGANISATION : UNIVERSITY OF PISA
DATE INITIATED : 1973 DATE COMPLETED : 1976	PROJECT LEADER : E. MANFREDI

Description :

Research program:

It has been devised a research program aimed to study, either by experiments and through a theoretical approach, the vibration behaviour of core components when subjected to fluid flow. The fuel element vibration behaviour is the first objective of this research.

The theoretical study will be developed mainly through the use of the finite elements method. An experimental research will be carried out by means of a test facility designed for this scope.

Facilities:

- Water test loop which is able to carry out experiment with test sections up to 3 meters lenght and 200 mm outer diameter with flow velocities up to 10 m/sec
- vibration transducers and flow measurement instruments
- signal processing instrumentation
- computers IBM 370 of CNUCE and IBM 1130 for the analysis of test data and for the development of theoretical studies.

Reference documents:

1. C. CARMIGNANI, E. MANFREDI, S. REALE
   Influence of assembly configurations on the flow induced vibrations of a BWR fuel element.
   3rd SMIRT Conference - London 1975.

The program has been sponsored by C.N.R. and C.N.E.N.

Classification 10.4	
Title 1 Ruisanalyse van de reactor en het primaire circuit	**Country** The Netherlands **Organization** KEMA
Tittle 2 Reactor and primary circuit noise analysis	**Scientists** J. Hoekstra J. v.d. Veer E. Türkçan

1. General aim

On load surveillance of vital systems.

2. Particular objectives

Dynamic behaviour of a BWR.

3. Experimental facilities

Dodewaard nuclear power plant.

4. Project status

Still in progress.

5. Next steps

Not applicable.

6. Relation with other projects

Same investigation at Borssele nuclear power plant by E. Türkçan.

7. Reference documents

None.

8. Degree of availability

Through the organizations KEMA and RCN.

11. MATERIALS AND MECHANICAL PROBLEMS IN NORMAL AND ACCIDENT CONDITIONS

Classification 11.1

Title 1 Mixed oxide fuel element behaviour under irradiation. Title 2	Country : Belgium Sponsor : BELGONUCLEAIRE Organisation : BELGONUCLEAIRE-CEN/SCK
Initiated : 1972 Completed : - Status : in progress Last updating : -	Project leader : Mr. H. Bairiot

1. General Aim

Study of mixed oxide fuel element behaviour under irradiation.
Post-irradiation examination.

2. Particular Objectives

a) Fuel densification study and evaluation of the qualitative and quantitative effects (e.g. neutronic point-of-view, physical mechanisms, attempt to deduce empirical correlations for predetermination calculations);

b) Fuel behaviour under accidental condition ;

c) Power ramp capabilities ;

d) Surveillance of statistical behaviour of Pu fuels in BWR and PWR power plants.

3. Experimental facilities and programme

- In-pile (BR-2 reactor) and out-of-pile tests for mixed oxide fuels. Neutron radiography.
- VENUS critical facility for study of local power peaking effects with simulated axial gaps.
- Fuel assemblies irradiated in BR-3, SENA, DODEWAARD, GARIGLIANO, etc ...

4. Project status

Progress to date : VENUS tests, some BR-2 irradiations.
Essential results : Power peaking effects.

5. Next steps : -

6. Relation with other projects : -

7. Reference documents : -

8. Degree of availability : Proprietary.

Classification: 11.1

Title 1 Ramp Testing of UO_2-Zr Fuel	COUNTRY: Denmark SPONSOR: DAEC, Risø ORGANIZATION: DAEC, Risø
Title 2 Ramp Testing of UO_2-Zr Fuel	Project Leader: P. Knudsen
Initiated: 1972 Completed: Status: Progressing Last Updating:	Scientists:

1. General Aim

To examine the performance of UO_2-Zr fuel pins during overpower ramps.

2. Particular Objectives

To submit BWR- and PWR-type UO_2-Zr fuel pins of medium-to-high burnup to power increases mainly simulating nominally normal operating conditions in power reactors.

3. Experimental Facilities and Programme

DR 3 Reactor at Risø with associated loops.

4. Project Status

4.1. Progress to date: BWR- and PWR-type fuel pins have been tested, with burnups up to 35,000 MWD/t UO_2.

4.2. Fuel pin failures indicate the existence of limiting combinations of design and operating conditions.

5. Next Steps

Continue tests to support and extend present experience.

6. Relation with Other Projects

Results will be used for verification of fuel model calculations.

7. Reference Documents

(a) P. Knudsen, H.H.Hagen, J. Stiff, Atomwirtschaft 3 (1974) 135-136.
(b) P. Knudsen, K. Bryndum, Trans. ANS Winter Meeting 1974, p. 140.

8. Degree of Availability

Classification: 11.1	
Title 1 Fuel Modelling	COUNTRY: Denmark SPONSOR: DAEC, Risø ORGANIZATION: DAEC, Risø
Title 2 Fuel Modelling	**Project Leader:** N. Kjær-Pedersen
Initiated: 1972 **Completed:** **Status:** Progressing **Last Updating:**	**Scientists:**

1. General Aim

To develop a computer model complex for fuel pin performance evaluation under nonaccident operating conditions.

2. Particular Objectives

A new version, WAFER-2, of the WAFER model is under development. Differences from WAFER-1 are: a) An improved finite difference treatment of the pellet with the aim of obtaining a more realistic fuel stress distribution and cracking pattern, and b) programmatic changes to minimize accumulation of error.

A simplified (disc) model, HOTCAKE, covering several fuel pin cross-sections is being built for faster analysis and improved fission gas release evaluation.

3. Experimental Facilities and Programme

Danish ramp testing programme.

4. Project Status

4.1. Progress to date: WAFER-1 in normal production status.

5. Next Steps

Completion of WAFER-2 and HOTCAKE.

6. Relation with Other Projects

Will utilize data from Danish overpower test programme for verification.

7. Reference Documents

P. Knudsen and N. Kjær-Pedersen: Performance Analysis of PWR Power Ramp Tests. ASME Winter Annual Meeting 1975, Houston, Tx. (75-WA/HT-68).

N. Kjær-Pedersen: Mathematical Description of WAFER-1, A three-Dimensional Code for LWR Fuel Performance Analysis. Nucl. Eng. and Des. 35 (1975) 387-398.

8. Degree of Availability

Not generally available.

Title 1 (original language) SURETE ELEMENT COMBUSTIBLE	COUNTRY France SPONSOR ORGANIZATION C.E.A. DSN
Title 2 (english)	Project leader C.RINGOT Scientists : H. VIDAL
Initiated (date) 1972 Completed : (date) Status : progressing Last updating (date)	

1. OBJECTIF PRINCIPAL

L'intégrité de la gaine des éléments combustibles qui constitue la première barrière aux produits de fission doit être assurée pendant toute la vie. Cela signifie qu'il convient d'éviter toute rupture à caractère systématique et que le comportement d'un élément rompu ne conduise pas à une évolution catastrophique Pendant un accident, tel que la dépressurisation du circuit primaire, l'élément combustible doit se comporter de telle sorte que le maintien de la fonction sûreté de la centrale soit assuré.

2. OBJECTIF PARTICULIER

Le programme des études est directement rattaché à l'objectif principal.

3. SITUATION

Les principales études sont :

a) comportement d'éléments rompus en pile-taux de relâchement des produits de fission d'éléments rompus.

Test d'irradiation BOUFFON dans le réacteur SILOE à GRENOBLE

1/12/74 - 12/75

Détermination de la durée de vie possible en pile d'un élément présentant une rupture primaire.

Les premiers résultats font apparaître un dégagement accru des produits de fission avec des éléments combustibles rompus.

b) comportement des gaines à haute température pendant le LOCA. Programme EDGAR

Cette étude constitue un support indispensable des essais en réacteur-appelés PHEBUS.

Le but est d'obtenir une courbe réelle des déformations diamétrales de la gaine en fonction du temps. Les résultats sont directement appliqués à l'élaboration d'un code.

- essais en laboratoire et en pile
- planning 1/12/75 - fin 76.

.../...

c) critère de rupture des éléments combustibles des réacteurs bouillants pendant des transitoires accidentels :

planning	- boucle en pile	
	(- étude de la boucle	1975
	(réalisation de la boucle	1975
	(expériences	78-79

d) comportement des éléments combustibles des réacteurs bouillants dans des conditions normales.

Détermination d'une courbe limite de puissance admissible avant rupture en fonction du taux de combustion : "STRIP" Test

programme en collaboration avec la Suède

- réacteur R2 à STUDVICK
- planning Juillet 75 - 80

Classification: 11.1	
Title 1 (Original Language): Untersuchungen zur chemischen Wechselwirkung Brennstoff/Zircaloy-Hülle (PNS 4235.3 - I.1.3., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Investigation of the Chemical Interaction between Fuel and Cladding	**Project Leader:** P. Hofmann
Initiated (Date): 1973 — **Completed (Date):** 1978/79	
Status: continuing — **Last Updating (Date):** December 1975	

1. General Aim

Determination of the extent of internal corrosion and of the influence of chemical interactions with the oxide fuel and the fission products on the mechanical properties of Zry-cladding tubes at temperatures up to 1200°C. It is the aim of these experiments to determine the deformation behavior and the failure mechanisms of cladding tubes. Essentially the limiting values of parameters governing cladding expansion and crack formation of cladding tubes will have to be determined quantitatively.

2. Particular Objectives

Influence of the O/U-ratio (oxygen potential) of UO_2 and of the reactivity of the fission products (Cs, I, Te) on the mechanical properties of Zry-cladding tubes at high temperatures and large differential pressures. Determination of the burst temperature and burst pressure of the cladding tubes as a function of the oxygen and fission products present. Determination of the strain rate ($\dot{\varepsilon}$) of tube specimens.

3. Experimental Facilities

Furnace for tube burst experiments under a cover gas at high temperatures.

4. Project Status

4.1 Progress to date

In 1975 theoretical estimates have been made relative to the maximum fission product concentrations and stoichiometric shift in oxide fuel as a function of burnup. Subsequently, the influence of the O/U-ratio of UO_2 and the reactivity of the fission products Cs, I and Te on the strain and rupture behavior of Zry-4-tubes has been determined experimentally at small differential pressures up to temperatures of 1200°C.

Along with the experiments on short tube specimen experiments have been conducted with Zry-crucible specimens up to 1300°C in order to study the chemical interactions between the fission product doped UO_2 and Zry.

4.2 Essential Results

The experiments, performed at temperature transients of 2-15°C/s showed, that in the presence of oxide fuel ($O/U > 2.00$) distinctly greater plastic strains of the cladding tubes occur compared to argon-filled reference specimens. Part of the Zry-tubes behave like superplastic materials with circumferential strains up to about 110%.

The effect of the Cs, I and Te fission products on the chemical interactions with Zry and hence on the mechanical properties of the cladding tubes depends to a different extent on the O/U-ratio of the fuel. In addition to oxygen tellurium and iodine enter into reactions with Zry and this results in embrittlement of the cladding material. By contrast, Cs reacts preferably with the hyperstoichiometric UO_2 forming Cs-uranate. In the presence of stoichiometric UO_2 part of Cs is present in elemental form in the fuel simulator and therefore substantially contributes to pressure enhancement at elevated temperatures leading to failure of the tubes.

5. Next Steps

Systematic investigation of the oxygen and fission product influence on the mechanical properties of Zry-cladding tubes at high differential pressures. The experiments will be performed with temperature and pressure being measured simultaneously. The strain of the cladding tubes will be measured continuously by means of an optical method. Experiments are planned under oxidizing test conditions (air, water vapor).

6. Relation with other Projects

PNS 4235.1 and 4235.2

7. Reference documents

/1/ 1st PNS Semi-Annual Report 1975, KFK 2195 (German with English abstracts)
/2/ 2nd PNS Semi-Annual Report 1975, KFK 2262 (German with English abstracts)
/3/ KFK-Nachrichten 3 (1975) page 34 (German)

Classification 11.1

Title 1	Fuel Clad Distortion under Accident Conditions.	**Country** UK
Title 2		**Sponsor** UKAEA **Organisation** Reactor Fuel Laby. Springfields
Initiated 1971 **Status** Continuing	**Completed** **Last updating**	**Project leaders** Dr. K. M. Rose

1 General Aim

To secure acceptable core conditions following a loss-of-pressure accident (LWR).

2. Particular Objectives

To discover whether internal pressure across zircalloy clad pins causes deformation which obstructs further cooling.

3. Experimental Facilities and Programme

Fuel bundles up to 1 m. length are heated in a furnace, through defined temperature and pressure sequences. The bundles are dismantled and distortions measured.

4. Project Status

Many such tests have been performed. Extension blockage can be produced particularly by low temperature (850°C) sequences.

The case is being considered for a more representative form of heating.

Reference Documents

SNI Report SNI 1/14. Oct. 1973

Classification 11.1

Title 1 High Temperature Creep of Zircalloy	Country UK
Title 2	Sponsor UKAEA Organisation Reactor Fuel Laby. Springfields
Initiated 1971 Completed Status Continuing Last updating	Project leaders Dr. K. M. Rose

1 General Aim

To secure acceptable core conditions following a loss of pressure accident (LWR).

2. Particular Objectives

To define limits of temperature and stress for which the zircalloy can will not creep significantly.

3 Experimental Facilities and Programme

Short lengths of fuel can are taken through a defined sequence of temperature and internal pressure. The final deformation of the can is measured.

4 Project Status

Many tests have been performed, originally with inert gas atmosphere, currently with a steam atmosphere as this significantly affects the results. The test is specific to the reactor temperature transient, so the tests are repeated from time to time as changes are made to the reactor design.

Reference Document

SNI Report SNI/1/14 Oct. 1973

Classification 11.1

Title 1 OVERPOWER TESTS	Country Italy
Title 2	Sponsor CNEN
Initiated 1.3.1974 Completed sept. 75 Status Progressing Last updating End 74	Organisation CNEN Project Leaders C. Lepscky

1. General aim
   Fuel cladding interaction and mechanical properties of irradiated cladding after rupture determined by power excursion. -

2. Particular objectives
   Investigate modes of rupture and material properties by neutron-radiography dimensional analysis etc. of irradiated cladding subjected to increasing power ramps up to power burst.
   Investigate the influence of gap (150, 230, 310 cold gap).

3. Experimental facilities and programme
   Irradiation(Halden reactor)and post irradiation examination at CSN Casaccia Center of CNEN

4. Project status
   4.1. Progress to date
   Irradiation completed at Halden (IFA 131) up to 30.000 MWD/t.
   Non destructive Post-irradiation tests of 6 rods prior to ovepower test already completed.
   First overpower test on 1 rod already performed including non destructive post-irradiation analysis of the same rods.
   Experimental procedure and facilities for subsquent over power tests aleready set-up

5. Next steps

Overpower tests performance will begin in 1975 on 16 rods with irradiation up to 30.000 MWD/t

6. Relation with other projects

None.

Classification 11.1

Title 1 Burst tests	**Country** Italy **Sponsor** CNEN
Title 2	
Initiated 1.1.1973 **Completed** 1976 **Status** Progressing **Last updating** End 73	**Organisation** CNEN **Project Leaders** G. Cali

1. General aim
   Mechanical properties of irradiated cladding under rupture conditions.

2. Particular objectives
   Measurement of 1) Ultimate burst strength 2) Wall thinning at fracture 3) Total circumpherential elongation in irradiated cladding internally pressurized up to rupture conditions.

3. Experimental facilities and programme
   See above.

4. Project status
   4.1. Progress to date
   Burst tests have been completed (1973) within the followi.. range of conditions: temperature: 280°C and room temperature: Burnup: 1.500 to 16.000 MWD/t (at 15); Integrated fast neutron dose: $6\div30\times10^{20}$ n/cm^2)
   Cladding diameter and type: 15.06mm ext.dia.(VDM) 14.26 mm ext.dia. (Wolwerin)

5. Next steps
   Burst tests to be performed on cladding irradiated up to 30.000 MWD/t

6. Relation with other projects
   None

7. Reference documents
   None

PROJECT TITLE : Cladding in LOCA conditions	CLASSIFICATION 11.1
SPONSORING COUNTRY : Italy	ORGANISATION : CNEN
DATE INITIATED : January 1975 DATE COMPLETED : Dec. 1976/79	PROJECT LEADER : P. Cerretti

Description :

1. General aim

Investigating modes of rupture and ballooning of irradiated and non irradiated zircalloy cladding under simulated LOCA conditions.

2. Particular objectives

Measurement of diameter of ballooned cladding at rupture, temperature, internal pressure and other relevant parameters in Zr-cladding, under simulated LOCA conditions.

3. Experimental facilities and programme

Facilities:

a) Elliptical irradiation oven (Casaccia)
b) Cladding heating by Joule effect (Casaccia)
c) In-pile loop (ESSOR reactor) (Ispra)

Programme

a) Tests on unirradiated and irradiated material under inert atmosphere (1975-76)
b) Tests on unirradiated material under oxidant atmosphere (1975-76)
c) In-pile tests on single rod and small cluster (1979).

4. Project status

4.1 Progress to date : construction of experimental facilities a) and b).

Classification 11.1	
Title 1 Splijtstofverlenging tijdens bedrijf	**Country** The Netherlands **Organization** KEMA
Title 2 Fuel elongation during power rating	**Scientists** W.R. van Engen J. Hoekstra W. Slegers

1. General aim

Pellet-clad interactions.

2. Particular objectives

To detect possible pellet-clad interaction situations in the core of a power reactor by means of

a) absolute elongation of the fuel rods or whole assemblies
b) change in time constant for heat transfer of the fission heat to the coolant

3. Experimental facilities

Dodewaard nuclear power plant.

4. Project status

Still in progress.

5. Next steps

To come to an additional method for incore measurements to have better knowledge over actual core conditions.

6. Relations with other projects

Measuring technique is based on the Halden incore instrumentation experience with differential transformers.

7. Reference documents

None.

8. Degree of availability

Through the organization KEMA.

604

Classification: 11.1

Title 1 Multi - Rod Burst and Flow Blockage	Country BRD (U.S.A) Sponsor Babcock and Wilcox Proprietary
Title 2	Organisation BBR Mannheim
Initiated: June 1974 Completed: Status: Last updating: Dec.1975	Project leaders Dr. B. E. Bingham Dr. W. A. Fiveland

1. General aim

To provide data on rod swelling and burst characteristics of B & W production Zircaloy - 4 cladding under loss-of-coolant accident (LOCA) conditions and to investigate possible channel flow blockage conditions.

2. Particular Objectives

3. Experimental Facility and Program

In 1975, rod burst and flow blockage tests were conducted with single-rod and five-rod test sections. These tests were conducted over a range of pressure differentials from 400 to 1400 psi at a heating rate of 50F per second.
The rods are internally heated by a resistance element. The test section containment is evacuated to minimize natural convection and the shroud is heated at the same rate as the cladding to simulate the proper boundary conditions of adjacent rods during the postulated LOCA. Test section power, rod pressure and temperature, and containment pressure and temperature are monitored. After the rods have been burst, blockage factors are determined over a wide range of Reynolds numbers. Finally, the test sections are encased in epoxy and sectioned at 1/2" intervals to determine the diametral expansion, blocked flow area, and thinned cladding thickness as a function of axial length at both ruptured and swelled locations.

4. Project Status

4.1 Progress to Date

The experimental phase has been completed.
Evaluation of the data is in progress.

4.2 Essential results

5. Next Steps

Publication of topical report.

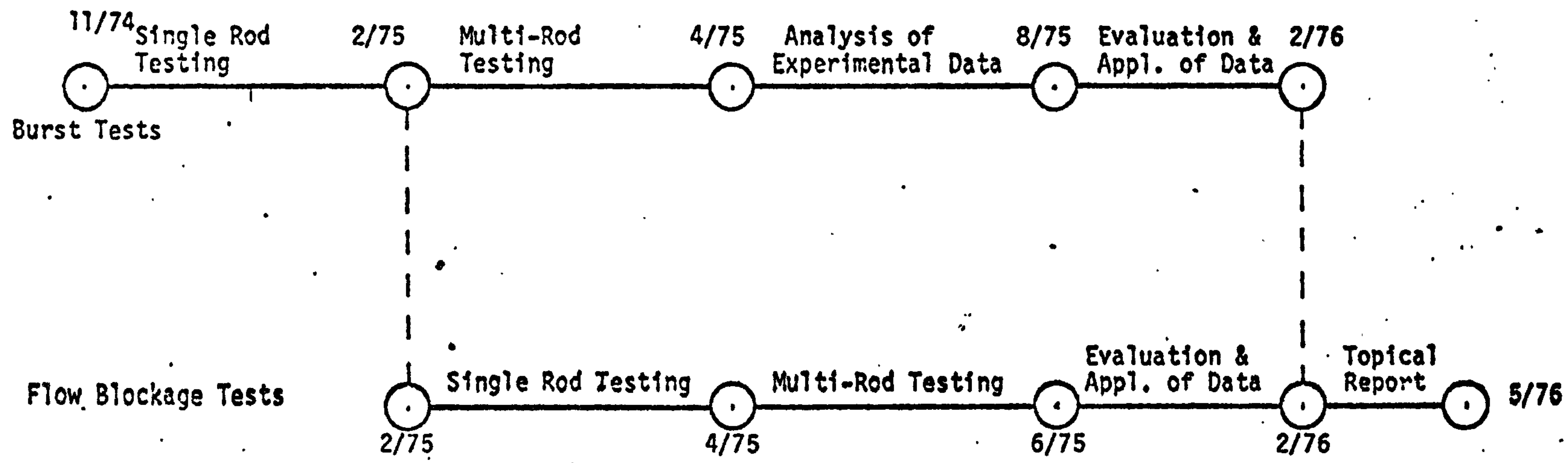
6. Relationship With Other Projects

7. Reference Documents

8. Degree of Available

Proprietary
(It is intended to publicly release the results during the fourth quarter of 1976)

MULTI-ROD BURST & FLOW BLOCKAGE



1/10/75

603

Classification:

Title 1 Fuel Post Irradiation Examination	Country BRD (U.S.A) Sponsor Babcock and Wilcox Proprietary
Title 2	Organisation BBR -Mannheim
Initiated: Mid 1974 Completed: Status: Progressing Last updating: End 1975	Project leaders M. H. Montgomery L. J. Ferrell

1. General Aim

This program consists of two parts; Part I, surveillance of the Mark B (15 x 15) assembly, and Part II, surveillance of the Mark C (17 x 17) assembly, Part I provides verification of performance for the 15 x 15 design in the present generation of B&W reactors. Part II will provide verification of performance for the Mark C (17 x 17) design of fuel.

2. Particular Objectives

To obtain performance data on irradiated fuel.

3. Experimental Facilitica and Program

The following data will be accumulated with a non-destruction examination at the reactor site of Oconee I fuel at the end of each of three cycles.

a) fuel assembly length and grid location
b) fuel assembly grid spring and holddown spring constants
c) guide tube inside diameter
d) fuel rod length, diameter and bow
e) rod to rod spacing
f) surface deposit characterization
g) gamma scans
h) visual examination

The destructive examination of a fuel assembly at the end of each of three cycles of Oconee I will be performed at the B&W Lynchburg Research Center. This will include the following work.

a) visual examination
b) gross and spectral gamma scan
c) fuel rod length
d) fuel rod diameter and ovality
e) eddy current examination of clad integrity
f) ultrasonic clad thickness
g) internal gas volume pressure and composition
h) mechanical properties of cladding
i) pellet density determination
j) crud analysis
k) chemical burnup analysis
l) graphics of tube and clad

4. Project Status

A. Progress to Date

Precharacterization was performed on selected fuel assemblies in Oconee I core 1 .

The non-destructive examination of Oconee I cycle 1 has been completed and the second cycle examination initiated.

The destructive examination of a selected fuel assembly started September 1975 and will be completed by mid 1976.

B. Essential Results

Mark B (15 x 15)

The first refueling of a B&W 177-Fuel Assembly NSS took place in later 1974. During this refueling, six (6) fuel assemblies that were reinserted in the core, as well as thirty (30) discharged fuel assemblies, were examined. All fuel assemblies were visually and/or dimensionally examined.

All four sides of each fuel assembly were viewed along the entire fuel length. The visual examination showed the fuel assemblies to be structurally sound with no evidence of wear or gross rod bowing. One hundred seven (107) fuel rods were gamma scanned for fuel stack length. Measured fuel column shortening was conservatively predicted by the B&W densification kinetics model. In excess of 7000 water channel spacings were measured and results showed that 95 % of all rod-to rod spacings changes less than 14 mils. Fuel rod and fuel assembly growth measurements were well within the design envelope. Fuel assembly holddown spring measurements show little or no spring relaxation resulting from irradiation after one cycle of operation.

Mark C (17 x 17)

The two demonstration fuel assemblies have been fabricated for insertion into Oconee II cycle 2.

-. Next Steps

Program will be continued as indicated in attached time schedule diagrams.

6. Relation With Other Projects

7. Reference Documents

8. Degree of Availability

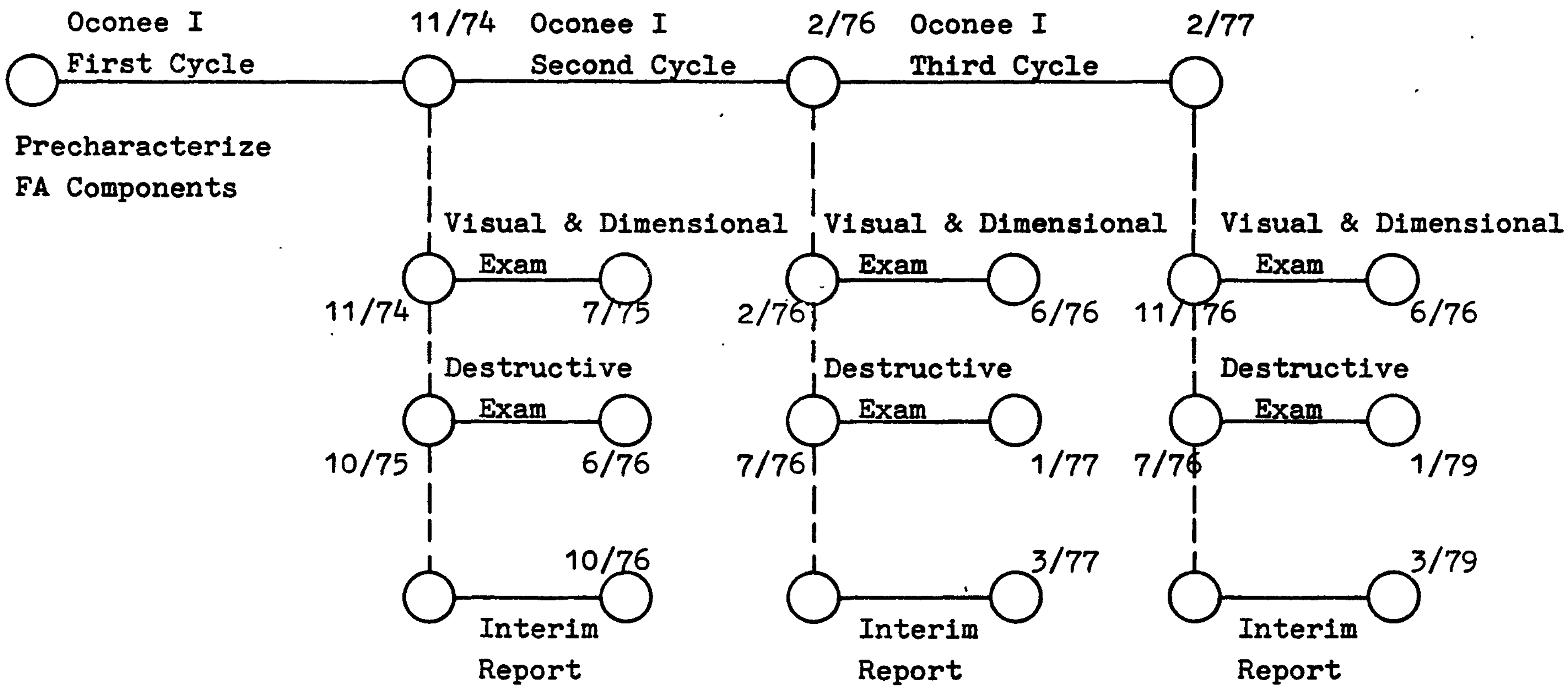
Proprietary

1. Budget

2. Personnel

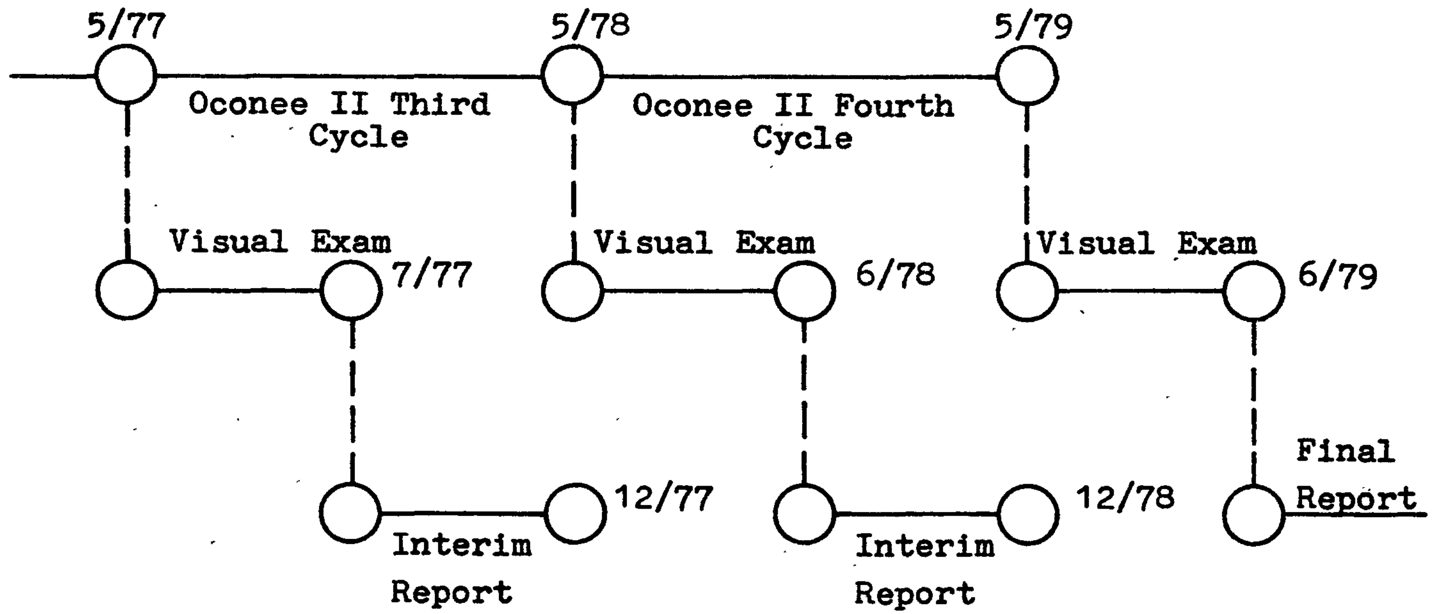
3. Additional informations
See attached time schedule diagrams.

FUEL POST IRRADIATION EXAMINATION PART I - MARK B (15 x 15)



Subject to Plant Operating Schedule

RADIATION EXAMINATION PART II - MARK C (17 x 17)



Subject to Plant Operating Schedule

Title 1 (original language)	COUNTRY France
SURETE DES CHAUDIERES - PROBLEME DES ACIERS.	SPONSOR CEA
	ORGANIZATION CEA/DSN
Title 2 (english)	Project leader C.RINGOT Scientists : B.BARRACHIN A.PROT
Initiated (date) Completed : (date) Status : progressing Last updating (date)	

1. OBJECTIF PRINCIPAL

S'assurer que la conception et le comportement des circuits primaires des réacteurs à eau ordinaire construits en France répondent aux critères de Sûreté.

2. OBJECTIFS PARTICULIERS

- Tenue des aciers de cuve
- contrôle et inspection en service.

3. SITUATION

1/ - Comportement sous irradiation de l'acier A 508 d3 utilisé pour la construction des réacteurs à eau PWR du programme français.

- conditions d'irradiation - dose 10^{14} à 6×10^{19} n/cm2 ($\geqslant$1 Mev)
  - température - 260°C à 290°C
- éprouvettes tirées des cuves en construction
- traitements thermiques représentatifs des diverses étapes de fabrication.
- propriétés mesurées : traction - résilience
- planning 72/75.

2/ - Comportement à la rupture brutale des aciers utilisés pour la construction des cuves PWR.

a/ Mesure K_IC sur grosses éprouvettes - CT et sur petites éprouvettes (mesure de J)

effet de l'irradiation examiné.

.../...

premiers résultats publiés dans le cadre du programme coordonné de l'AIEA.

b/ Restauration des propriétés des aciers pour cuves PWR dégradées par l'irradiation.

Planning 74/75

c/ Détermination des vitesses de propagation des fissures sur l'acier A508 d3 irradié.

essai sur éprouvette CT en cellule chaude

planning 74/76

d/ Préciser le comportement en fatigue plastique des matériaux utilisés dans les réacteurs à eau.

- influence de l'effet de l'environnement en fatigue olygocycle
- essai de fatigue en flexion - en température et en pression en présence d'eau.

planning 73/75

e/ Améliorer la valeur des analyses de tenue à la fatigue des enceintes sous pression de manière à mieux apprécier la sûreté sous l'action des chargements olygocycliques.

Les essais ont pour but de préciser les marges de sécurité obtenues dans le cas de l'application des recommandations du code ASME pour l'analyse à la fatigue.

efforts cycliques - de pression
- de flexion sur les piquages

en cours.

f/ Evaluer l'imprécision de la mécanique de la rupture en cas de grande déformation.

analyse de structures fissurées de géométrie simple

en cours.

3/ Contrôles non destructifs

Amélioration des méthodes de contrôle non destructif, dans le cadre des inspections périodiques des circuits primaires de réacteur, en particulier en vue d'une meilleure définition des paramètres caractéristiques des défauts décalés.

Moyens mis en oeuvre :

a / Ultrasons :

- mise au point de traducteurs à haute résolution permettant :
  - le contrôle au travers du revêtement inoxydable des cuves
  - le dimensionnement correct des défauts
  - le contrôle des soudures d'acier inoxydable austénitique

- évaluation de l'holographie acoustique : comparaison de ses possibilités avec les méthodes précédentes.

.../...

b/ Courants de Foucault :

Amélioration des procédés actuels en vue des inspections périodiques des tubes d'échangeurs (générateurs de vapeur).

c/ Traitement des données :

Mise au point de procédés permettant une évaluation rapide des résultats d'inspections périodiques.

Etat d'avancement :

- Ultrasons :
  - les traducteurs sont actuellement au point. Il reste à qualifier la méthode sur défauts réels et à comparer les dimensionnements obtenus par ultrasons et par micro ou macrographie.
  - en cours les développements pour le contrôle des soudures d'acier inoxydable austénitique, la détection des défauts mal orientés, l'holographie acoustique.
- Courants de Foucault :

  développement en cours de sondes spéciales et de l'électronique modifiée.
- Traitement des données : en cours

d/ Utilisation des techniques de contrôle par émission acoustique pour la surveillance en service des cuves.

- but : développement d'un système complet de surveillance en service avec localisation des défauts des cuves

  appareillage opérationnel

  1969/76

—

Classification: 11.2	
Title 1 (Original Language): Konstruktive und rechnerische Entwicklungsarbeiten an Druckbehältern in Stahlbauweise (RS 35 - II.1.7., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fried. Krupp GmbH
Title 2 (english): Design and Calculation of Steel Pressure Vessels	Project Leader: Jorde
Initiated (Date): May 1, 1969 — **Completed (Date):** December 31, 1975	
Status: finished — **Last Updating (Date):** December 1975	

1. General aim

Steel pressure vessels are currently finding an ever increasing field of application. Their use in modern technologies, such as reactor engineering, requires considerable design and calculation procedures as well as intensive theoretical and experimental work, because reliability of operation here is a prime consideration aside from the ever important aspects of economy.

This research programme, therefore, aims to analyze, by theoretical and experimental work and by development of appropriate designs the reliability and manufacturing economy of pressure vessels and to develop new approaches. Detail problems are to be solved and progress is to be made in the two, often opposing, aspects mentioned: reliability and economy.

2. Particular objectives

- Theoretical and experimental investigation of the strength behaviour, i.e. strain and stress characteristics, of such vessel components as closure shells with holes, flanges, seals, nozzles and nozzle arrays, supports.
- Design of a removable vessel seal
- Analysis of the plastic behaviour of vessel components for the assessment of their bursting properties.

- Experimental fabrication of nozzle bodies while joining them to the shell of the vessel
- Extensive static and cyclic loading tests on vessel models made of 22 NiMoCr37 steel with arrays of nozzle.
- Accoustik emission tests

3. Experimental facilities and programme

Extensive static and cyclic loading tests on 6 vessel models (diameter 2200 mm, wallthickness 30 mm) made of 22NiMoCr37 steel with arrays of nozzle. Stress and strain measurements in the elastic and plastic range, accoustik emission analysis, bursting tests.

4. Project status

1. Progress to date
2. Essential results

- Methode to calculate flanges weakened by bolt holes and loaded by concentrated and distributed moments directed tangentially (upturning moments)
- Study of the available literature on stress distributions in cylindrical and spherical pressure vessels with nozzle connections
- Approximate strain and stress analysis of perforated shells of revolution
- Program to calculate plastic deformations of shells
- Removable vessel seal (patent application P 2118254.7)
- Extensive results of the stress and strain measurements and of the bursting tests
- Results of the accoustik emission analysis

5. Next steps

———

6. Relation to other projects

- Development of large pressure vessels of multi-layer steel construction
- Investigation relating to the manufacture of components with large cross-sectional areas by the deposition of Ni-(Cr)-Mo steels by machine welding
- Connection of multi-layer wall areas to solid components for pressure vessels.

7. Reference documents

(in German language with 1 exception)

- Study report 50/70
  Design and calculation of steel pressure vessels,
  Report No. 1
- Study Report 79/70
  Design and calculation of steel pressure vessels,
  Report No. 2
- Study Report 1001/71
  Design and calculation of steel pressure vessels,
  Report No. 3
- Patent application P 2118254.7,
  Removable vessel seal
- The behaviour of circular rings with bores loaded by equidistant upturning moments; F. Baumgart, R. Dietze, J. Jorde. Structural mechanics in reactor technology, Berlin September 20 - 24, 1971.
- Approximate determination of the strains and stresses of perforated shells of revolution; H. Coenen, H.P. Lehrke, J. Jorde. Structural mechanics in reactor technology, Berlin September 20 - 24, 1971.
- Quarterly and annual reports in the series IRS-Forschungsberichte, report period from January 1971 to December 1975.

8. Degree of availability

Address for request:

BMFT, Bonn

9. Budget

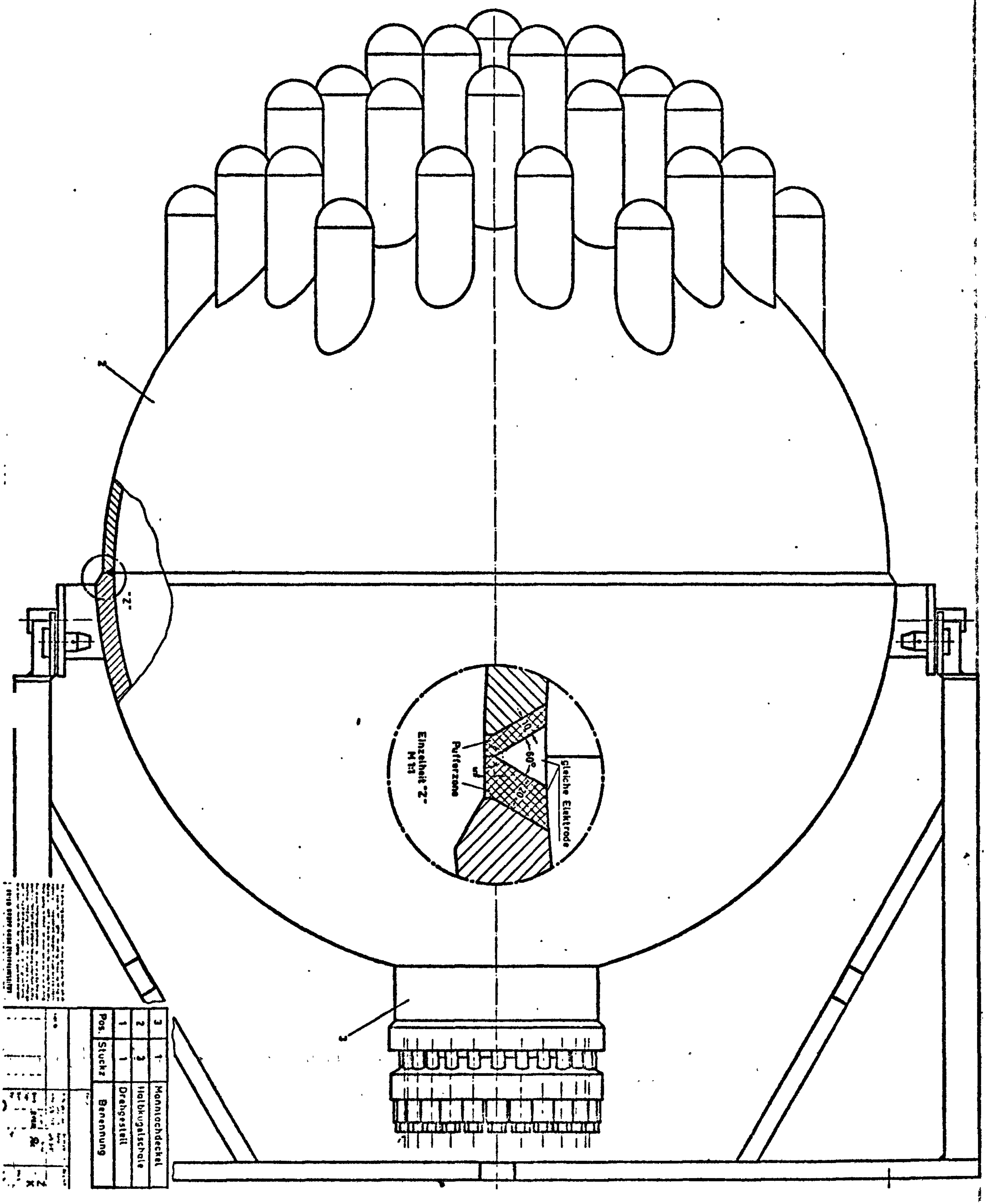
Cost of labor	DM 130.000,--
Cost of material	DM 90.000,--

10. Personnel

14 men/month

11. Additional informations

Side view of the vessel models



Classification: 11.2	
Title 1 (Original Language): Statusbericht Reaktordruckbehälter (RS 63 - II.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: IRS, Köln
Title 2 (english): Status Report Reactor Pressure Vessel	Project Leader: Röhrs Rittig
Initiated (Date): Dec. 28, 1971 Status: continuing	Completed (Date): 1976 Last Updating (Date): December 1975

1. General aim

Within the Reactor Safety Research Program the studies relating to the reactor pressure vessel technique are one of the key points. This Status Report was made to determine really substantive priorities for the use of the available funds.

2. Particular objectives

In detail the following targets are aimed:

1. To show the latest status of the technical development in the Federal Republic of Germany with respect to design, fabrication, test, and examination of reactor pressure vessels (RPV) for light water reactors fabricated by means of rolling construction or forged steel construction by pointing out the principal aspects for the safety assessment.
2. To set up a research program for the extension of the basis of the RPV safety assessment and for an improvement of the techniques of design, manufacturing, and test.
3. Concentrated representation of targets, research programs, and essential results of the 12 topics of the Heavy-Section-Steel-Technology- (HSST) Program executed in the USA.

3.1 Experimental facilities

Not relevant

3.2 Program

To provide a broad spectrum of experiences and guaranteed objectivity to a large degree it was decided to proceed as follows.
By order of the Federal Minister for Education and Science (BMBW) the Institute of Reactor Safety (IRS) took the lead. Experienced industrial firms, associations, and safety consultant institutions elaborated special contributions according to specifications and by order of the IRS. A group of independent experts analysed these reports with respect to principle essentials as to the safety assessment of RPV. An editing board consisting of leading experts with special experience in RPV-technique took the whole leading.

The Status Report will consist of three volumes.
Volume I will contain all results in concentric form. Starting from the state of RPV-technique at the end of 1973 positive as well as negative knowledge aspects will be shown. As the status report will essentially serve for setting up substantiated priorities for research activities the concise form of volume I will concentrate partly strongly on the gaps of knowledge.

Volume II will consist of text and figures of the data- and information collection. Here the status of RPV-technique will be layed down as found out by the authors of the special reports and as coordinated subsequently in detail.

Volume III will give a summary of the HSST Program presently running in the USA. For each of the 12 topics of this program information will be given as regards objective, research program, project status and essential results.

4. Project status

4.1 Progress to date

Drafts of Volume II (Collection of data and information) and Volume III (HSST-Program) were finished.

4.2 Essential results

Volume I of the Status Report "Reactor Pressure Vessel" (summary of results) is available as a print.
Volume II and III are ready to be reviewed by the editing board.

5. Next steps

The review of Volume II and III by the editing board will take place at the beginning of 1976. After passing the board Volume II and III will be printed and published.

6. Relation with other projects

RS 46 Study on Current and Planned Research Projects in the Pressure Vessel Field
STAATLICHE MATERIAL-PRÜFUNGSANSTALT, Stuttgart University, 1969 - 1971

7. Reference documents

Reports in the series IRS-FORSCHUNGSBERICHTE:

Jan.	-	March 1975	IRS - F - 25	
Apr.	-	June 1975	IRS - F - 26	
July	-	Sept. 1975	IRS - F - 27	
Oct.	-	Dec. 1975	IRS - F - 28	(in German)

Annual reports:

A 72	IRS - F - 12	
A 73	IRS - F - 18	
A 74	IRS - F - 24	(in English)

8. Degree of availability

Unrestricted distribution.

620

Classification: 11.2	
Title 1 (Original Language): Grob- und Detailspezifikation des Forschungsprogramms "Kompenentensicherheit" (RS 192 - II.2., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** MPA, Stuttgart
Title 2 (english): Rough and Detailed Specification Research Program Structural Integrity of Components	**Project Leader:** Dr. Sturm Dr. Issler
Initiated (Date): November 13, 1975 **Completed (Date):** April 30, 1976 **Status:** continuing **Last Updating (Date):** December 1975	

1. GENERAL AIM

The research program "Structural Integrity of Components" integrates all activities concerning reactor pressure vessel problems running or planned in the Federal Republic of Germany. Objectives of this program are to give basic information for the quantification of safety margin in order to further reduce the remaining risk through a specified further development of the reactor pressure vessel technique.

2. PARTICULAR OBJECTIVES

In order to be able to work on this research program "Structural Integrity of Components" with largest possible efficiency it has been planned to divide the program into subsequent phases. For the first phase, which will last approx. 4 years, a rough specification will be worked out and in addition for the first year a detailed specification.

3. EXPERIMENTAL FACILITIES AND PROGRAM

EXPERIMENTAL FACILITIES

- not necessary -

PROGRAM

The first phase of the research program "Structural Integrity of Components" covers the following single projects :

1) Possibility of Defect Detection (Defect Classification)
2) Homogeneity of the Mechanical Properties (small scale specimens - not irradiated, irradiated)
3) Integral Test on Large Scale Specimens
4) Integral Test on Intermediate Size Vessels
5) Theory

They have to be classified in regard to basic proceedings including test setup, test performance, test time, temporal operating sequence and costs.

4. PROJECT STATUS

4.1 PROGRESS TO DATE

Statement of the problem list for the five single projects.

5. NEXT STEPS

Completion of the rough specification, consisting of problem list, structural project plan, network plan with dates and costs, as well as the detailed specification, in which the tests for the first test year ought to be listed and described in detail.

6. RELATION WITH OTHER PROJECTS

Reactor Pressure Vessel / High Priority Program 22 NiMoCr 37 (RS 101)
HDR Safety Program (RS 123)

7. FORMER REPORTS

- none -

8. DEGREE OF AVAILABILITY

Upon request at BMFT

9. BUDGET

DM 329.000,-

10. PERSONNEL

51.5 MM

11. ADDITIONAL INFORMATION

- none -

	CLASSIFICATION: 11.2
TITLE: Breukanalyse Onderzoek aan Stompen (BROS)	COUNTRY: NETHERLANDS SPONSOR: Ministry of Economic Affairs ORGANIZATION: Rotterdam Dockyard and others
TITLE (ENGLISH LANGUAGE): Fracture analysis on nozzle intersections (BROS)	PROJECTLEADER: C.J. Drijver
INITIATED: March 1972 COMPLETED: end 1977 STATUS: in progress LAST UPDATING: february 1976	SCIENTISTS: M.J.G. Broekhoven H.J.M. van Rongen A. de Sterke

1. General aim

   Crack extension behaviour in heavy section nuclear steel pressure vessels in areas of complicated geometry.

2. Particular objectives

   In particular attention will be paid to cracks occuring in areas of complicated geometry and stress distribution, notably nozzle corner regions. The programme covers the following items:
   - early detections of defects
   - detailed surveillance of the growth of defects
   - establishment of prediction for further growth of defects.

3. Experimental facilities and programme

   The main activities are:
   - theoretical research directed towards the development of efficient computation methods to calculate the crack extension behaviour for complicated configurations, such as nozzle corner cracks, using advances analytical and finite element techniques,
   - experimental investigations of crack extension behaviour in uni-axial and bi-axial loaded cracked nozzle-on-flat-plate models, manufactured from nuclear grade pressure vessel steel (ASTM A508 Cl.2),
   - determination of fatique and fracture (elastic and elastic-plastic) related material parameters for A508 and model materials,
   - research on the applicability of Acoustic Emission technics to detect, locate and characterize crack extension in the materials of concern,
   - the main test facility to be used is a bi-axial loading bench, suitable for plate dimensions 700x700x15 mm, capacity 200 tonnes static force, 150 tonnes dynamic loading, frequency 2 Hz.

4. Project status

   - Test pieces have been stressed on an Amsler fatigue test bed.
   - Computer programmes have been completed.
   - The bi-axial loading bench is under construction.

5. Next steps

   The next step will be the testing of the nozzle on flat plate test pieces.

6. Relation with other projects: EPOSS

7. Reference documents

   A series of about 50 technical and progress reports have been prepared. Nearly all these reports are written in the Dutch language.

8. Degree of availability

   The reports have been submitted to the Ministry of Economic Affairs, Laan van N.O. Indië 123, The Hague. Requests for obtaining copies should be sent to this address.

TH - Delft	CLASSIFICATION: 11.2
TITLE: Elastisch-plastisch breukmechanica onderzoek aan stompen van nucleaire hoge-drukvaten (EPOSS)	COUNTRY: NETHERLANDS SPONSOR: Ministry of Social Affairs ORGANIZATION: TH - Delft
TITLE (ENGLISH LANGUAGE): Elastic-plastic fracture analysis research on cracks at nozzle intersections of nuclear high-pressure vessels (EPOSS)	PROJECTLEADER: Latzko
INITIATED: August 1974 COMPLETED: 1977 (scheduled) STATUS: in progress LAST UPDATING: january 1976	SCIENTISTS: Broekhoven V.d. Ruijtenbeek Peeters

1. General aim

   Crack extension behaviour in heavy section nuclear steel pressure vessels in areas of complicated geometry.

2. Particular objectives

   Evaluation of the applicability of the J-integral concept (an elastic-plastic fracture mechanics concept) for predicting elastic-plastic crack extension for complex crack configurations in nuclear pressure vessels, notably cracks in nozzle corner regions.

3. Experimental facilities and programme

   Main activities are:

   a. theoretical investigations
      - Computation of J-integral values by the finite element method for 2-dimensional configurations, ranging from simple test specimens to uniaxially loaded plates with cracks emanating from a central hole;
      - Computation of J-integral values by the finite element method for some 3-dimensional configurations, viz.
        - bars with a quarter-circular edge crack
        - uniaxially loaded plates with quarter-circular cracks emanating from a hole
        - flat plates with a central nozzle and a crack at the nozzle corner
      - Evaluation of the applicability of simplified approximation procedures to determine J for said configurations.

   b. experimental investigations
      - J_{Ic}-tests on standard specimens for the model material, i.e. Al 2024-T3
      - Model-tests on 2-dimensional configurations, i.e. uniaxially loaded plates with cracks emanating from a central hole
      - Model-tests on the 3-dimensional configurations mentioned under a.

4. Project status

   - Procedures for efficient computation of J-values by the finite element method have been established.
   - Computations of J-values for simple 2-dimensional configurations have been completed.
   - Computations of J-values for more complicated 2-dimensional configurations are in progress.
   - J_{Ic}-tests on standard specimens are in progress.
   - Experimental investigations on uniaxially loaded plates with cracks emanating from a central hole are in progress.

5. Next steps

   Continuation of investigations of 2-dimensional configurations; initiation of theoretical and experimental investigations on 3-dimensional configuratio

6. Relation with other projects

   The study is an extension of the BROS-project into the elastic-plastic regime.

7. Reference documents

   Report MMPP-110, Delft Un. of Techn., Lab. for Thermal Power and Nucl. Engineering.

8. Degree of availability

   Through Ministry of Social Affairs, C.R.V., Postbox 69, Voorburg, Netherlands.

9. Budget

   Approx. Dfl. 750.000,-

Classification 11.2.1.	
Title 1 Dynamisk Brudmekanik	COUNTRY Denmark SPONSOR DAEC Risø
Title 2 Dynamic-Fracture Mechanics	ORGANIZATION DAEC Risø
Initiated: January 1973 Status : progressing Completed Last updating	Project leader: A. Nielsen C. Debel C. Engel

1. General aim

Evaluation of the influence of weld defects on the integrity of steel structures.

2. Particular objectives

The model used in treating the problem is a crack propagating with a certain speed in a low toughness material surrounded by a high toughness material. The conditions for a continued crack propagation in the tough material is considered.

The set up of relations between crack length and crack propagation speed and available energy is attempted.

3. Experimental facilities and programme

Experiments are carried out in common tensile testing machines using special equipment to control fracture conditions. Instrumented impact and fast bending machines are available for testing of small test pieces. Large test peices are prepared by welding dissimilar steels together. Smaller test pieces are prepared by welding brittle weld metal in tough steel or by local embrittling tough steel by electron beam fusion of Titanium in the base metal.

4. Project status

A mathematical analysis has been carried out indicating that there is a large difference in the critical length between a resting and a running crack.

Locally embrittled small test pieces have preliminarily been tested by instrumented impact testing.

5. Next steps

Experiments are continued.

6. Relation with other projects

This project is related to work carried out by students at the Technical University of Denmark and the Danish Engineering Academy.

7. -

8. Degree of availability

No limits to the availibility of results of this investigation.

Classification: 11.2.1		
Title 1 (Original Language): Bestimmung bruchmechanischer Sicherheitskriterien für elastisch-plastisches Werkstoffverhalten (RS 90 - II.2, Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Determination of Fracture Mechanical Safety Criteria for Elastic-Plastic Behaviour of Materials		Project Leader: E. Klausnitzer, Prof. Dr. Detert
Initiated (Date): Dec. 1972	Completed (Date): Sept. 1975	
Status: Completed	Last Updating (Date): Dec. 1975	

General Aim and Particular Objectives

Three approaches have been made for treatment of fracture which considers the elastic-plastic deformation occuring during fracture process.

1. The COD-concept
2. The J-integral concept (G-curves)
3. The concept of resistance-curves (R-curves)

Experimental Facilities and Program

The accuracy of the experimental results depends on the exact determination of the crack opening displacement at the frontside of the specimen. A measuring system for COD with high accuracy was manufactured.
The next approach was to establish for each specimen-type a calibration curve, showing the relationship between the ratio displacement/load and the crack length. The calibration-curve once established would serve to control the fatique crack propagation under cyclic loading. Furthermore in evaluating R-curves such calibration-curves are indispensible.
For each specimen geometry which is provided to be investigated in this program the corresponding calibration-curves have been established. The G-curves and the correction functions have been calculated for each specimen geometry.

In order to obtain the various data necessary for establishing a critical fracture mechanics criterium a great number of specimen had to be tested. The specimen have been prepared for tensile test by initiation fatique cracks of different lengths under cyclic loading. The experiments were carried out on samples of steel 22NiMoCr37 (similar to ASTM A 508 Cl 2) at room temperature. Load deflection diagrams were taken from which the necessary data could be gained to determine the values corresponding to the three above mentioned methods.
The observation were so far concentrated to CT- and SEN-specimen. In order to study the influence of specimen thickness on the experimental results CT-specimen of 20, 30 and 40 mm thickness were used. Using specimen of a particular thickness but with different initial crack-lengths the parameter of crack lengths were investigated.

Project Status/Prggress to Date

Three samples with the shape of 6T-CTS-samples, but different thickness of the samples, were tested at room temperature. The test run was observed by ultrasonic emission to determine crack initiation.

The 5T-CTS and CCT-samples were analyzed.

Project Status/Essential Results

The evaluation of the results of the 3T-CTS-samples has shown, that the $COD_{max.}$-values are in the same order compared with the 2T-CTS-samples. Stable crack growth was not observed.

The evaluation of the R-curve-concept has shown, that the crack propagation resistivity increases linearly with the effective crack length. 2T and 3T-samples have different slopes, the larger sample results in a steeper slope.

The 6T samples showed different fracture behaviour and various critical COD-values, which agree with the plastic deformation capability. The R-curves are equal and have a constant slope. The crack length has no influence on these results.

The CCT-samples showed no instable crack propagation, because the whole residual cross-section was deformed plastically. The R-curves were independent from the crack length.

Next Steps

Work on this project has been completed

Relation to Other Projects

RS 102 Fracture Behaviour Under Complex Loading Conditions

RS 69 Evaluation of Tube and Vessel Tests by Means of Fracture-Mechanics Relations

Reference Documents/Degree of Availability

No reports available.

<table>
<tr><td colspan="3">Classification: 11.2.1</td></tr>
<tr><td colspan="2">Title 1 (Original Language):
Schweißversuche zum Plattieren von Reaktor-Druckgefäßen
(RS 91 - II.2., A 75)</td><td>COUNTRY:
BRD
SPONSOR:
BMFT
ORGANIZATION:
KWU, Erlangen</td></tr>
<tr><td colspan="2">Title 2 (english):
Welding Tests on Cladding of Reactor Pressure Vessels</td><td rowspan="2">Project Leader:
E. Klausnitzer
Prof. Dr. Detert</td></tr>
<tr><td>Initiated (Date):
Dec. 1972
Status:
Continuing</td><td>Completed (Date):
July 1976
Last Updating (Date):
December 1975</td></tr>
</table>

General Aim

It is the aim of the present work to examinate the influence exerted by different parameters on residual stresses using a suitable measurement method.

Particular Objectives

In order to protect the inner surface of reactor pressure vessels against corrosion, band or strip welding procedure is used for cladding the ferritic vessel with austenitic stainless steel. During cladding high residual stresses occur in thick-walled components. In addition, the different thermal expansion coefficients of stainless steel and ferritic steel induce further residual stresses. These stresses cause under-clad-cracks during postweld heat treatment. Therefore, the knowledge of the magnitude and of the distribution of residual stresses after welding and after postweld heat treatment are of special interest.

Experimental Facilities and Program

Cladding was made on A 508. The dimension of the plates was about 125 mm thick, 240 mm wide and 300 mm long. The thickness of the austenitic cladding was about 5 mm. The following measurement method was used: strain gauges were placed on the surface of the ferritic side of the plate and the cladded, opposite side was machined in steps. The strain response to the removal of the material was measured using strain gauges. These strains are used for compensating the stress distribution in cladding direction of the plate in dependance of the plate thickness.

Project Status/Progress to Date

Residual stress investigations on plates with different welded cladded materials were terminated. Samples which were welded under low thermal stresses were glowed to stress-relieve.

Creep rupture tests and relaxation tests were carried out on specimen with simulated superheated material in the temperature range of 550 ° - 620 °C. During the creep rupture tests the load was chosen to cause rupture within 100 hours. The elongation of the samples during the relaxation tests was 0,5 %. The results of the acoustic emission analysis on stress-relieved cladded specimen were compared with metallografic investigations.

The residual stresses were measured

a) on a helically welded ring before stress-relieve-glowing

b) on a close welded bore hole of a 205 mm thick specimen of 22NiMoCr37

With the finite elements method the residual stresses of a circular welding seam on the RPV were calculated.

Regarded were the thermal history (melting, solidification, heating velocity and cooling velocity) and the mechanical properties as a function of time and temperature.

Project Status /Essential Results

Austenitic cladding materials with different heat extension coefficients showed no rather different residual stresses on cold basic material. Under-clad-cracks arise however from such single welded claddings during crack-relieve-glowing, which are welded under rather low heat transaction.

The error of experimentally determined stresses increases with lower detract rate. When layers of 0,5 mm are detracted the error will not exceed 12 N/mm^2 (usually lower $\pm$ 2 N/mm^2). The seathering data are greater on the begin and decrease linear with the thickness of the sample. An extension error of $\pm 5 \cdot 10^{-6}$ m/m results in a stress error of $\pm$ 12 N/mm^2. On the helically welded ring with the bore-kernel-method residual stresses up to 450 N/mm^2 were found on the inner surface. On the exterior a value of 274 N/mm^2 was measured.

The finite elements calculations showed that stresses under a first band welding zone are not increased by next overwelded bands. The highest stresses were found in the heat affected zone. The deviatoric residual stresses are reduced about 14 % of the value after welding, while the hydrostatic tensile stress reduce about 35 %. This results in a very high three axial tensile stress condition.

Next Steps

a) Experimental investigation on material characteristics at higher temperatures (for 22NiMoCr37 and 20MnMoNi55)

$$\dot{\varepsilon} = f(\varepsilon, \sigma, T)$$
$$E = f(T)$$
$$\sigma_B = f(T)$$

Melting temperature (Liquidus and solidus)
$\Delta l/l$-curves with T, t-dependancy

b) Introduction of better material characteristics in the computer program

c) Measuring of the residual stresses with the bore-kernel-method and x-ray methods.

Relation with Other Projects

RS 101 - Reactor Pressure Vessel Urgent Programm 22NiMoCr37

Reference Documents/Degree of Availability

No reports available.

Classification: 11.2.1	
Title 1 (Original Language): Forschungsprogramm Reaktordruckbehälter Dringlichkeitsprogramm 22 NoMoCr 3 7 (RS 101 - II.2., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** -
Title 2 (english): Reactor Pressure Vessel High Priority Program 22 NiMoCr 3 7	**Project Leader:** Prof. Dr. Krägeloh Prof. Dr. Uetz
Initiated (Date): December 1, 1972 **Completed (Date):** March 31, 1976 **Status:** continuing **Last Updating (Date):** December 1975	

1. GENERAL AIM

Determination of possible critical conditions in welding seams (strength seams) and their effect on strength or safety against fracture with the objective of optimizing the material choice and the parameters during welding and heat treatment (see also annual reports IRS-F-18).

2. PARTICULAR OBJECTIVES

Thorough metallographic as well as electron microscopic investigations have confirmed that intercrystalline micro cracks can develop in the coarse grain zone of weldings (strength seams) in the material 22 NiMoCr 37 (A 508 Cl 2) during heat treatment after welding - especially during stress-relief heat treatment. While the reason for crack formation can be explained by the relaxation embrittlement in the range of the stress-relief heat treatment temperature, the question is, how much such coarse grain zones can influence the strength and toughness of the structural component. The urgent Program 22 NiMoCr 37 is supposed to compile the necessary documents for a safety analysis for reactor pressure vessels in accordance with the specific material crack conditions.

As a way towards solutions parameter studies have been considered, which can be divided in tests with small and large specimens as well as "component tests". In

addition to the conventional simulated welding treatments, which can only be carried out on small specimens, simulated tests on large specimens are scheduled. For the creation of geometrically defined HAZs electron beam welding is intended.
Parallel to the experimental tests it is tried to determine temperature fields and heat stresses theoretically.

3.1 EXPERIMENTAL FACILITIES

See annual report IRS-F-24.

3.2 RESEARCH PROGRAM

The research program has been classified in the following nine partial projects :

a) intermediate vessels (ZB 1)

b) welding of discs into reactor pressure vessel ring (S-UP-Ro)

c) simulated specimens - tests on the welding simulator (Si-SSi)

d) simulated specimens - annealing heat treatment and relaxation tests (Si-G/R)

e) simulation with electron beam (Si-E-Strahl)

f) submerged-arc-welding, circumferential seam (S-UP-RN)

g) submerged-arc-welding, longitudinal seam (S-UP-LN)

h) deformation and fracture mechanism (VBM)

i) device for the testing of large specimens (10 000 Mp Project)

4. PROJECT STATUS

a) ZB 1

In 1974 a bottom of a vessel tore off during pulsating internal pressure test. After the carrying vessel (material : 13 MnNiMo 5 4, o.d. x s = 1713 mm x 120 mm) was repaired with two new bottoms (material : 13 MnNiMo 5 4 and 22 NiMoCr 3 7) and two new welded-in discs (material : 22 NiMoCr 3 7, MPA-melts 14 and 15), the pulsating test was continued under the former conditions (p_{max} = 320 bar, p_{min} = 3 bar).

When after 500 cycles still no sign of crack development could be found, the longitudinal seam of the disc number 3 (MPA-melt 15) was weakened by a notch (320 mm long and 40 mm deep) cut from inside into the HAZ of the radial welding flank. After 6290 cycles, as referred to the original condition of the disc, the fatigue crack, originating in the notch, had cut through the wall; the disc is taken out for investigation purposes.

The welding seam areas of the taken-out disc number 1 were investigated mechanically and metallographically.

b) S-UP-Ro

The mechanical investigations of the disc materials, the metallographic investigations of the longitudinal disc seams, the evaluation of the measurements with the stress-probing extensometer as well as the evaluation of the temperature measurements on the forged discs 1 and 2 were finalized.

c) + d) Si-SSi and Si-G/R

The simulation tests were continued with variation of the overheating temperature, of the heating rate during stress-relief heat treatment and of the chemical composition of the material (basis 22 NiMoCr 37), as well as with the simulation of the multi-layer effect.

e) Si-E-Strahl

Possibilities and limitations of the electron beam welding were investigated with parameter studies. The coarse grain zone of the HAZ was simulated by the application of simple and special dummy seams and its mechanical characteristics were determined. During the welding the temperature field was measured in regard to location and time. The possibilities for the manufacturing of compound specimens were clarified under consideration of the test results.

f) S-UP-RN

The submerged-arc-welded circular seam was investigated after the post weld heat treatment non-destructively by ultrasonic and magnetic particle inspection. For the mechanical testing of the parent material, HAZ and weld metal, notch-impact-bending specimens, DWTT specimens, Pellini- and large scale bending specimens as well as fracture mechanics specimens were taken from the test body and tested at different temperatures. Transversal and tangential cuts were manufactured for metallographic investigations.

g) S-UP-LN

On the 143 mm thick seams I and II Pellini-, creep- and DWT-tests as well as one notch-bending test (notch within the HAZ) were carried out. Further two 150 mm repeated seams IB and IIA were welded according to the parameters of seam I and II. On the seams III (150 mm thick) and IV (250 mm thick) the following tests were made : tangential cuts, tensile and notch-impact-bending tests (weldment and HAZ), fracture mechanics tests (HAZ, parent material), Pellini- and DWT-tests, creep- and simulation tests, weld metal analyses as well as notch-bending test.

Besides that two faulty boring samples (322 mm Ø) from a forged half-ring (770 mm Ø) were tested in regard to ultrasonic findings, strength and toughness characteristics and chemical analysis.

h) VBM

The evaluation of the measurement with the stress-probing extensometer was further followed up within the scope of the project S-UP-Ro and finalized. The calculation of temperature fields was discontinued.

i) 10 000 Mp PROJECT

The single parts of the plant including hydraulics with complete electric and electronic installation as well as hydraulic damping equipment are completed. The setup can be started.

4.2 ESSENTIAL RESULTS

a) ZB 1

The mechanical and metallographic tests of the seams in disc number 1 showed no indications of local embrittlement or stress-relief cracking. In disc number 3 (longitudinal seam with notch) leakage before fracture occured at an average tangential stress of 210 N/mm^2.

b) S-UP-Ro

The metallographic investigations of the discs and longitudinal seams showed that after the stress-relief heat treatment relaxation cracks are existing on all seams in the coarse grain zone of the HAZ. In some cases also hot cracks were found.

When evaluating the temperature fields the course of the peak temperatures within the HAZ generated by the welding layers could be determined according to size; connections between the temperature cycles, associated with the single points of the HAZ, and the created grain sizes could be proved.

c) + d) Si-SSi and Si-G/R

The most important results are shown in the following table (next page).

e) Si-E-Strahl

By synchroneously parallel applied seams coarse grain zones with a useful cross section of 4 to 5 x 50 mm^2 may be generated, however, the resulting hardness is lower than on the submerged-arc-welded seams on account of the lower cooling rates. Abrupt drop or transition temperature of notch-impact strength are displaced by 30 to 40 ^{o}C in the direction of higher temperatures as compared to the parent material.

The coarse grain zones generated on dummy or connecting seams show characteristics, which practically are in agreement with those of the submerged-arc-welded seams, since comparable cooling rates are on hand, however, they are only 0.3 to 0.6 mm wide.

The developed welding technology allows the production of seams with 150 mm thickness. The compound specimens, of which a greater number (approx. 1000 pieces) was produced through connection seams, proved useful as the tests have shown. On the CT compound specimens the resulting residual stresses are influencing the result.

f) S-UP-RN

The non-destructive testing of the circular seam showed no indications of larger defects. Neither did the metallographic investigation with the aid of transversal and tangential cuts show any faults; especially no cracks could be found. The mechanic test made significant that the ductility is better within the HAZ than within the weld metal.

Results Si - SSi / Si - G/R

22 NiMoCr 37 influencing parameter	comparative treatment, condition	influencing of crack sensitivity (or crack danger) negative	positive	toughness [1] (service suitability) negative	positive
overheating simulation	1300 °C/5s $\Delta t_{800/500}$ = 10s				
variation of the peak temperature		>1100 °C	<1100 °C	>1100 °C	<1100 °C
multi layer effect double overheating		1300+1200°C 1300+ 700°C	1300 + 900÷1000°C	1300+1200°C 1300+ 700°C	1300 + 900÷1100°C
very slow cooling		x		x	
simulation of the stress-relief heat treatment	610°C/6h				
larger relaxation- (creep-) strain during stress-relief heat treatment		x		x	
extended annealing time without mechanically evident damage	creep strain $\varepsilon_{bl} \leq 0{,}2$ %	0	0		x
with mechanically evident damage	$\varepsilon_{bl} > 0{,}2$ %	0	0	0	0
slower heating rate	600 °C/h	<approx. 100°C/h		< approx. 100°C/h	
temperature of stress-relief heat treatment	610 °C	550	650	550	650
analysis variation basis 22 NiMoCr 37	1300°C/5s; 600°C/6h				
segregations		x		x	
increasing contents of Mo		>0,5		>0,5	
increasing contents of Cu, P, Sn		x		x	
vanadium		>0,01 %		>0,01 %	
chromium 0,3 - 0,7 (with Mo ≈ 0,8)		0	0	0	0
nickel			≥ 1 %		≥ 1 %
large number of MnS-slags		x		x	

[1] from subsequent tensile, notch-impact-bending and fracture mechanics test

g) S-UP-LN

Seam I and II : creep tests at 610 ^{o}C with transversal specimens from the weld metal (S) and the parent material (GW) showed that S of seam II is showing higher tendency for creeping than of seam I. S of seam II is creeping faster than the GW next to it.

Seam IB and IIA : a macro crack with relaxation cracks like on seam I were not found on seam IB after the heat treatment. The transversal and tangential cuts show hot cracks. Notch-impact values of specimens from the HAZ are unchanged on seam IIA as compared to seam II, on seam IB lower than on seam I.

Seam III and IV : hot cracks were found in the tangential cuts along the HAZ. The results of the notch-impact-bending and tensile tests with specimens from the HAZ and the weld metal are meeting the requirements of the parent material. All K_{Ic}-values in the fracture-toughness-temperature diagram are above the reference curve (NDTT = -15 ^{o}C).

Boring Samples : the found ultrasonic defect indications concern separations within the segregation zones. The strength and toughness characteristics do not allow a conclusion of embrittlement.

h) VBM

The evaluation of the measurements with the stress-probing extensometer shows that the method is not sufficiently exact for a reliable determination of strain variations, which occur during the stress-relief heat treatment.

5. NEXT STEPS

a) - h)

Compilation of the 3rd technical report as Final Report.

i) 10 000 Mp PROJECT

Presumably fromon March 1976 erection of the foundations. Transport of all large pieces from the manufacturers (Osnabrück) to Stuttgart and starting of the machine assembly fromon September/October 1976. Beginning of operation of the plant may be expected in spring 1977.

6. RELATION WITH OTHER PROJECTS

See annual reports IRS-F-18.

7. REFERENCE DOCUMENTS

Quarterly Reports in IRS Research Reports, period reported on :

January 74 - December 74 (German)

Annual Report IRS-F-24, May 75 (English)

8. DEGREE OF AVAILABILITY

Upon request at BMFT.

Classification: 11.2.1	
Title 1 (Original Language): Bestrahlung von Druckgefäßstählen im Rahmen des "Koordinierten Programmes" der IAEA (RS 85 - II.2., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GKSS, Geesthacht
Title 2 (english): Coordinated Programme on Irradiation Embrittlement of Pressure Vessel Steel	**Project Leader:** Dr. Schmitt
Initiated (Date): April 1973 — **Completed (Date):** 1976	
Status: continuing — **Last Updating (Date):** December 1975	

1. General aim

The purpose of the programme is to promote coordinated research in the field of irradiation embrittlement of reactor pressure vessel steels, at an international level. The results obtained will give a possibility for a more precise interpretation of individual surveillance results.

2. Particular objectives

Eight institutes in seven nations are participating in this programme whose basic goal is to establish that the bases for describing embrittlement, measurement of the neutron spectrum and fluence and mechanical properties (on a typical - standard reference - heat of steel) were sufficiently standardized to permit direct intercomparison among international programmes and to compare embrittlement sensitivity of national steels with the standard steel.

3. Experimental facilities and programme

Samples of the pressure vessel steel ASTM A533 Grade B Class 1 will be irradiated and different materials tests are to be carried out. 192 specimens Charpy V-notch, 48 tensile and 10 WOL-1X-specimens will be neutron irradiated in several capsules in the Material Test Reactor FRG-2 in Geesthacht.
The irradiation temperature is 100 °C and 290 °C respectively. The neutron doses are $5 \cdot 10^{18}$, $2 \cdot 10^{19}$ and $1 \cdot 10^{20}$ n/cm^2. The fluence is measured with several neutron monitors (Co, Ag, In, Ni etc).

After the irradiation, the following tests will be done in the hot cell facilities:

- Impact test at the whole transition region and determination of the transition temperature
- Tensile test (tensile strength, yield strength, elongation, reduction of area)
- fracture toughness test.

4. Project status

1. Progress to date
Two irradiation series with Charpy-V-notch- and tensile test specimens, which were taken from the standard steel blocks in the parallel and transverse rolling direction have been completed. The irradiation temperature was 290 °C and the doses were $2 \cdot 10^{19}$ n/cm^2 and $1 \cdot 10^{20}$ n/cm^2. The impact and tensile tests were carried out in the hot cell facilities. Several Charpy-V-notch specimens, which were irradiated with a neutron dose of $2 \cdot 10^{19}$ n/cm^2 were postirradiation annealed at different temperatures to determine the degree of recovery.

2. Essential results
The results of the first irradiation test with a dose of $2 \cdot 10^{19}$ n/cm^2 are:

Impact tests

The shift of the NDT-temperature for transverse specimens is

NDTT = 45 °C (5,2 mkp/cm^2-criterium)
" = 40 °C (50 % fibrous fracture)

The lateral expansion of 0,9 mm was reached by a temperature of 56 °C.

The shift of the NDTT for longitudinal specimens is

NDTT = 52 °C (at 5,2 mkp/cm^2)
" = 30 °C (at 50 % fibrous fracture)

The lateral expansive of 0,9 mm was reached at 54 °C.

Tensile tests

Increase of the tensile strength at room temperature from 63 kp/mm^2 to 69 kp/mm^2 = 9,5 %.
Increase of the yield strength from 47 kp/mm^2 to 55 kp/mm^2 = 18 %.
The decrease of the total elongation and the reduction of area are 6 % resp. 4,5 %.

The test with an irradiation dose of $1 \cdot 10^{20}$ n/cm^2 gave the following results:

Impact tests

Transverse specimens:

NDTT = 120 °C (at 5,2 mkp/cm^2)
" = 98 °C (at 50 % fibrous fracture)

The lateral expansion of 0,9 mm was reached at 135 °C.

Longitudinal specimens:

NDTT = 113 °C (at 5,2 mkp/cm^2)
" = 86 °C (at 50 % fibrous fracture)

The lateral expansion of 0,9 mm was reached at 120 °C.

Tensile tests

Increase of the tensile strength at room temperature from 63 kp/mm^2 to 78 kp/mm^2 = 23 %.

Increase of the yield strength from 47 kp/mm^2 to 67 kp/mm^2 = 43 %.

The decrease of the total elongation and the reduction of area are 22 % resp. 13 %.

Recovery annealing

Charpy-V-notch specimens, which were irradiated at 290 °C to a dose of 2 . 10^{19} n/cm^2 were annealed at two temperatures of 700 °F and 800 °F for 168 hours.

Using the three criteria (5,2 mkp/cm^2, 50 % fibrous fracture and 0,9 mm lateral expansion) for the comparison of the only-irradiated and postirradiation annealed conditions, the recovery is 50 % for the lower and 70 % for the higher temperature.

5. Next steps

Further irradiations with Charpy-V-notch and tensile specimens at an irradiation temperature of 100 °C and three doses are planned.

Other specific goals include the advancement of fracture toughness measurement procedures and post-irradiation annealing for restoration of initial properties.

6. Relation with other Projects

RS 101 "Reactor pressure vessel urgent programme 22NiMoCr37".

RS 61 "Experimental investigations of brittle fracture behaviour of thickwalled cylindrical components".

7. Reference documents

Quarterly reports in the series IRS-Forschungsberichte RS 85 - (V 73/2 - V 75/4) (in German).

8. Degree of availability

Available from the IRS, Cologne, Germany

Classification: 11.2.1		
Title 1 (Original Language): Tiefspaltschweißen (RS 102-01 - II.2., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Ges. IfaM
Title 2 (english): Narrow-Gap Welding		**Project Leader:** Dr. K. Seifert I.D. Henderson
Initiated (Date): 1.5.1973	**Completed (Date):** 31.12.1976	
Status: continuing	**Last Updating (Date):** December 1975	

1. General aim

The research project involves the joining of heavy sections of the fine-grained low-alloyed pressure vessel steel 22 NiMoCr 3 7 using the gas shielded arc welding process of Narrow-Gap welding. During the report period, welds were produced in 100 mm rolled sections and thereafter in 200 mm forgings, whereby the mechanical properties of the welded joints were found to be equivalent to or better than those exhibited by the base material.

2. Particular objectives

a. Following preliminary investigations on a range of welding consumables 3 solid welding wires were subjected to a program of Narrow-Gap welding tests on 100 mm rolled steel with a series of pre- and post-welding heat treatment conditions. The results of metallurgical and mechanical testing of the resulting joints permitted the selection of a single welding wire and the optimal heat treatment conditions for the subsequent welding tests on 200 mm forged sections. A series of 8 welds were then produced and initially subjected to non-destructive testing. For the X-ray investigations, full thickness specimens were tested with a 17 MeV-Betatron and revealed a low and acceptable level of porosity. Ultrasonic testing supported this result. Mechanical testing involved tensile, impact bend and fracture mechanics investigations and the results were compared with those from base material specimens. Metallographic studies were primarily aimed at ensuring that no micro-cracking occurred either in the weld seam or in the HAZ. Electron microscopy has been used to study fracture characteristics on impact and fracture toughness specimens.

b. Since Narrow-Gap welding employs a one-sided welding technique, additional tests on 500 mm thick steel were conducted to investigate the use of ceramic backing strips. In laboratory tests the welds are usually made with steel backing strips, which must be removed prior to cladding operations.

c. The welding of heavy section steels using a gas shielded welding process with heat inputs in the order of 6 KJ/cm is not common. Therefore primary interest rests on the soundness of the bond at the fusion boundary and the size of the HAZ. In order to specify the HAZ during Narrow-Gap welding, tests on 100 mm sections were done to establish the temperature distribution in the base material region adjacent to the fusion boundary.

3. Experimental facilities

The Narrow-Gap welding machine has proven to be reliable equipment although specifically the life of the welding torches remains a problem to be solved. Fig. 1 shows the welding head on the track and carriage system and the dual torch assemblies consisting of welding torch, gas feed tubes and torch position control probes. Test sections from 200 mm forgings are shown prepared with end-pieces and held in a specially developed fixture which acts to maintain a parallel gap of 8 mm width during welding. The weldments are pre-heated to 200 °C prior to starting the weld and thereafter an inter-pass temperature of 200 to 250 °C is maintained.

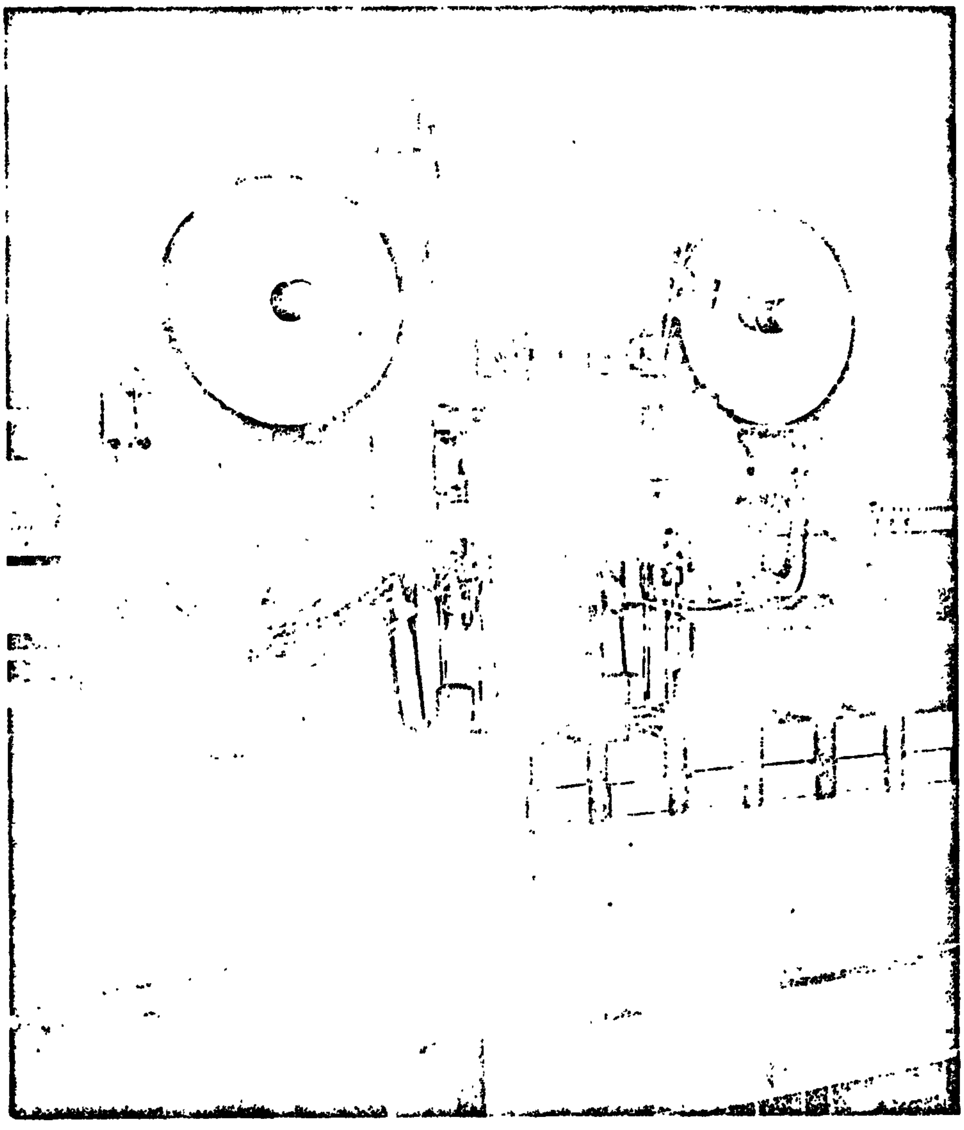


Fig. 1: Narrow-Gap welding head and 200 mm thick weldments prepared for a welding test.

4. Project status

4.1 Progress to date

Satisfactory welded joints have been produced in 200 mm forged sections of the steel 22 NiMoCr 3 7 using the MIG-wire Griduct SF-K4 and a welding heat input of 6 KJ/cm. Preheating is essential to obtain optimum mechanical properties and when fracture toughness is an important criterion the welded joints should be stress-relief heat treated at 600 °C. The problem of reheat cracking in the HAZ during or following this thermal treatment has not been detected in metallographic investigations. This result can be directly attributed to the low heat input during Narrow-Gap welding.

4.2 Essential results

4.2.1 Base material: Surprisingly large variations in the notch impact toughness and fracture toughness of rolled and forged sections of the steel 22 NiMoCr 3 7 have been measured. Fig. 2 summarizes the results of tests on base material and welded specimens in the as-welded condition (3D) and following stress-relief heat treatment at 600 °C (3F). In all tests Schnadt-sharp notched specimens were used.

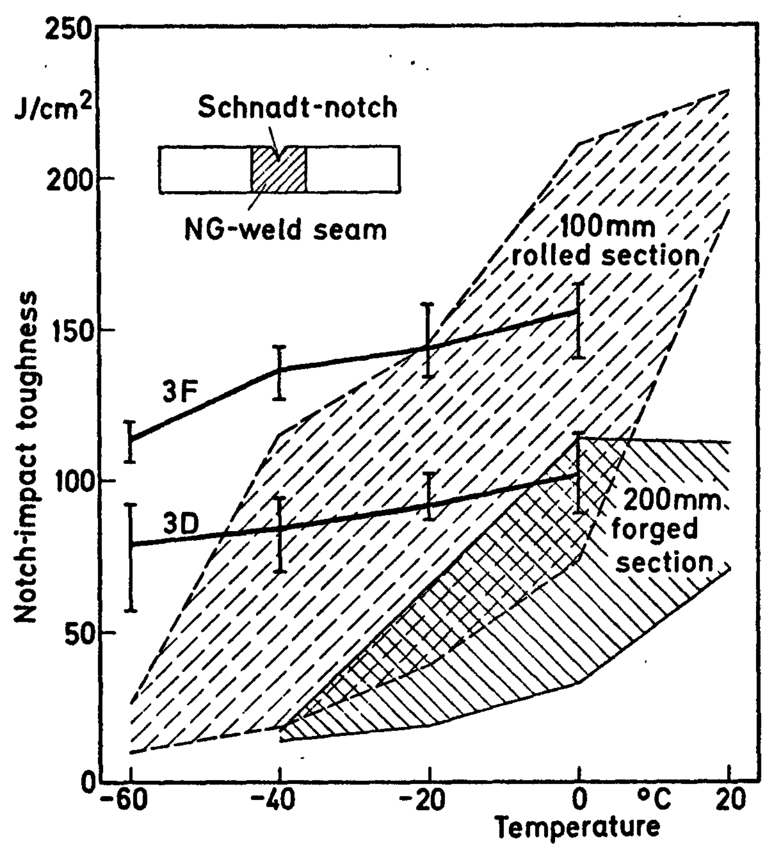


Fig. 2: Notch impact toughness of base material 22 NiMoCr 3 7 and Narrow-Gap welded specimens as a function of test temperature

The large spread of results in the transition region is due to the large number of specimens taken from various positions in the material section. The results of COD-tests also reveal the forged material to have a transition temperature which lies approximately + 40 °C above that for the rolled material. With regard to tensile properties, similar strength values resulted for each charge of the base material.

4.2.2 A macrograph of a Narrow-Gap weld in 200 mm forged reactor pressure vessel steel is shown in Fig. 3. The weld seam is parallel and only 13 mm wide and approximately 55 passes of the welding head are required to complete such a weld. The HAZ is only 1,2 mm wide and microscopic studies have revealed the zone of coarse grain growth to be only 250 μm wide. During these tests a steel backing strip was used for starting the weld.
Fig. 4 contains a macrograph of a Narrow-Gap weld produced with a ceramic backing strip. An acceptable root geometry is obtained which permits the cladding of the undersurface without need for additional preparation.

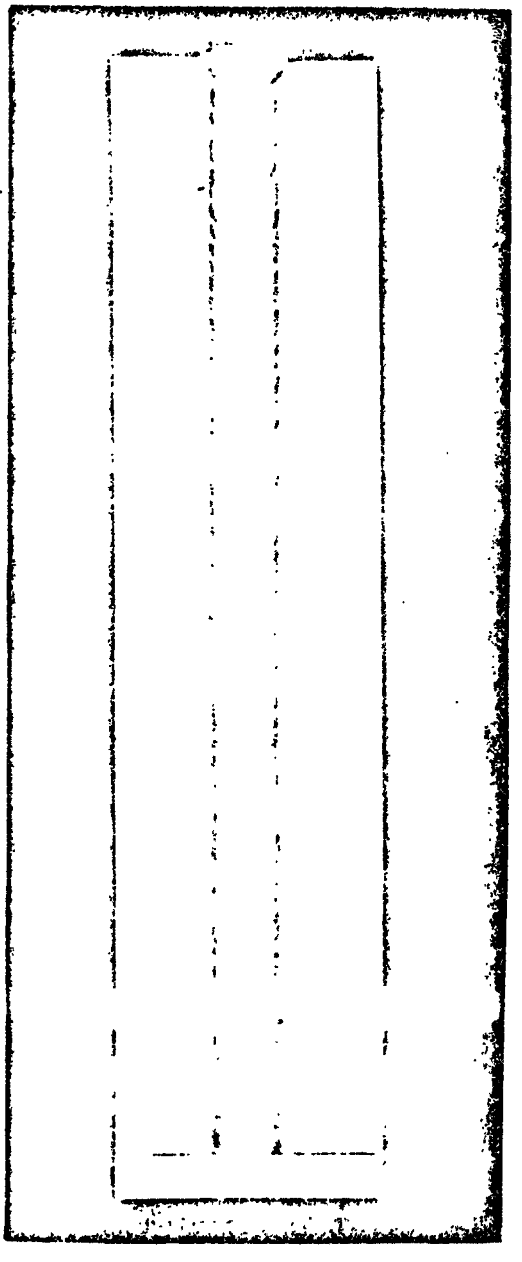


Fig. 3: Section of a Narrow-Gap weld in 200 mm 22 NiMoCr 3 7

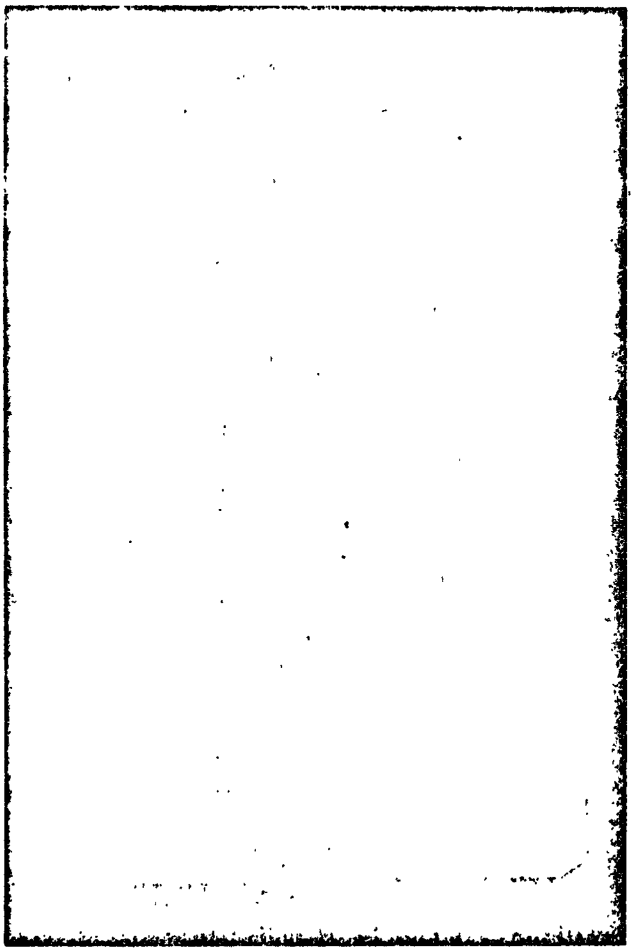


Fig. 4: Section of a Narrow-Gap weld produced with a ceramic backing strip
Some porosity is evident in the section which is due to unavoidable rust spots on the welding wire.

4.2.3 Using 18 thermocouples during two welding tests to determine the temperature distribution in the HAZ, peak temperatures of 1400 °C were measured at the fusion boundary and at a distance of 1,1 mm from the fusion line the peak temperature had fallen to 700 °C. Temperature cycling due to the dual torch welding system and the multi-pass welding technique was observed at each measuring point in order to correlate the metallographic structures in different parts of the HAZ.

5. Next steps

The preparatory work for the Narrow-Gap welding investigations in the horizontal-vertical position are completed and following minor alterations to the welding head, these tests can proceed. Initially these tests will be conducted on 50 mm thick plates and later on 100 mm thick sections.
The Narrow-Gap welding of 300 mm forgings of the steel 22 NiMoCro3 7 will complete the planned research program. Since the welding equipment is designed only to accomodate 230 mm using a one-sided welding technique, an attempt will first be made to modify the equipment to permit 300 mm to be welded. Should this attempt fail then the welding of these heavy sections from both sides will be investigated.

6. Relation with other projects

With respect ot the fracture toughness testing and non-destructive testing of base materials and welded joints, the related projects are:

RS 102-11 Quantitative investigation of the fracture behaviour of part through cracks in structural components.
IFKM, Freiburg; May 1973 - April 1976

RS 102-12 Investigation of unstable crack initiation and crack arrest by fracture mechanics analysis.
IFKM, Freiburg; May 1973 - April 1976

RS 102-13 Crack-opening-displacement for the assessment of structural components.
IFKM, Freiburg; May 1973 - April 1976

RS 102-16 Non-destructive testing of metallic structures by ultrasonic methods.
IzfP, Saarbrücken; May 1973 - April 1976

7. Reference documents

I.D. Henderson H.-D. Steffens "Narrow-Gap welding of alloyed steel". Welding Congress '75 in Brno Czechoslovakia, June 10-15, 75

I.D.Henderson H.-D.Steffens:	"Fracture toughness of Narrow-Gap welded joints in the nuclear pressure vessel steel 22 NiMoCr 3 7 " Paper G 5/3 SMIRT, September 1 - 5, 1975 London.
I.D.Henderson	"Tiefspaltschweißen - ein neues Verfahren für den Behälterbau ", DVS-Bericht 38, Entwicklungsstand und Anwendungsmöglichkeiten neuzeitlicher Schweiß-, Löt-und Prüftechnologien, Seite 152 - 157.
I.D.Henderson H.-A.Crostack H.-D.Steffens	"Untersuchungen zur Bruchzähigkeit mit Schallemissionsanalyse", DVS-Berichte 38, Entwicklungsstand und Anwendungsmöglichkeiten neuzeitlicher Schweiß-, Löt- und Prüftechnologien, Seite 174 - 180.

Classification: 11.2.1	
Title 1 (Original Language): Bruchverhalten von teilweise durchgehenden Rissen (RS 102-11 - II.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Gesell-schaft, IFKM
Title 2 (english): Fracture Properties of Part-Through Cracks	Project Leader: Dr. E. Sommer H. Kordisch
Initiated (Date): 1.5.1973 **Completed (Date):** 30.4.1976 **Status:** continuing **Last Updating (Date):** December 1975	

1. General Aim

The general aim is to study the fracture properties of part-through cracks in nuclear reactor components with experimental methods, based on the concepts of fracture mechanics.

2. Particular Objective

To show the propagation characteristics of part-through cracks, artificial markings on the fracture surfaces are induced during a fatigue test and are used to analyse the process of crack growth in relation to such parameters as initial crack geometry, boundary conditions and type of loading. In order to establish the test conditions, a calibration curve of da/dN versus ΔK measured on compact-specimens is used.

3. Experimental Facilities and Programme

The fatigue tests were performed on a 1.6 MN servohydraulic machine with an internal pressure device giving up to 150 N/mm^2 and the calibration tests on a 0.25 MN servohydraulic machine.

4. Project Status

4.1 Progress todate

The models of reactor pressure vessel components which have been examined, are tubes of the reactor steel 22 Ni Mo Cr 3 7 with part-through cracks aligned either axially or circumferentially and loaded by a pulsating internal pressure, with an axial tension superposed. Crack growth characteristics and their dependence on the initial notch geometry have been obtained from the final fracture surfaces through an analysis of the crack front extension using the artificially induced fracture surface markings. During the test the crack front positions which were to be marked were estimated with the aid of an ultrasonic device.

4.2 Essential Results

The safety analysis of partially cracked nuclear reactor components requires a prediction of the crack behaviour. Such predictions can be obtained from this work and, for example, in fig. 3 are shown the results from the fracture surface analysis for an axial (fig.1) and a circumferential crack (fig.2). The normalized crack extension through the thickness, a_i/d, is plotted versus b_i/d, the normalized extensions in the axial and the circumferential directions respectively.

5. Next Steps

Further investigations of the effects of the same parameters on part-through cracks in tubes (22 Ni Mo Cr 3 7) and on surface cracks in plates of the same material are under way.

6. Relations with other Projects

This investigation is closely related to the other reactor safety projects in the IFKM.

7. Reference Documents

Annual Reports in the series IRS Forschungsberichte

A 73: IRS - F-18

A 74: IRS - F-24

Report at the 7. Meeting AK "Bruchvorgänge" of DVM, Aachen, W.-Germany, Oct. 1975

8. Degree of Availability

The IRS-Reports can be ordered through Institut für Reaktorsicherheit der TÜV e.V., 5 Köln 1, Glockengasse 2, the Report of the 7. Meeting of the AK Bruchvorgänge through Institut für Festkörpermechanik, Rosastraße 9, 78 Freiburg, Germany.

Fig.1: Fracture surface of an axially aligned crack in a tube

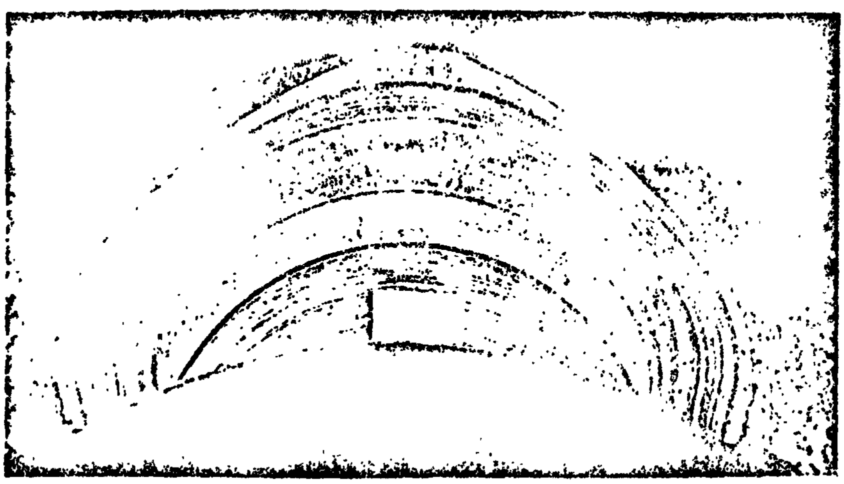


Fig.2: Fracture surface of a circumferentially aligned crack in a tube

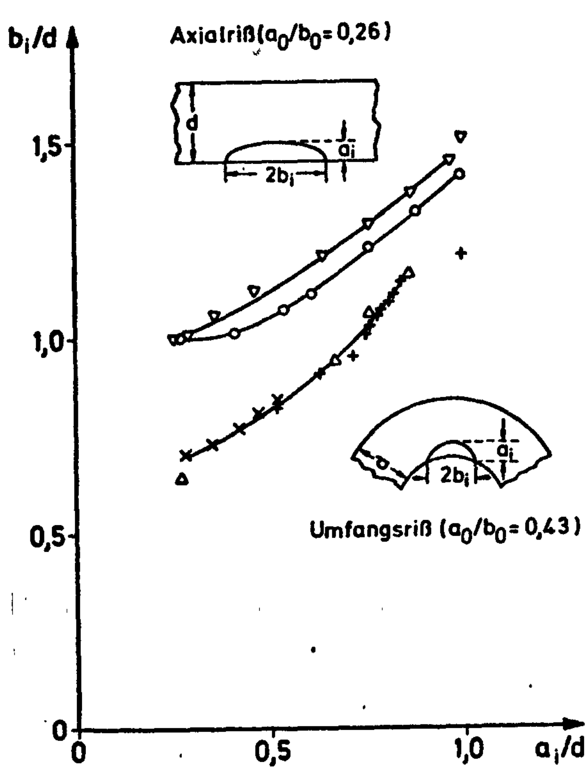


Fig.3: Crack extension behaviour of axial and circumferential cracks in tubes

Classification: 11.2.1	
Title 1 (Original Language): Rißeinsatz und Rißarrest (RS 102-12 - II.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Gesell-schaft, IFKM
Title 2 (english): Investigation of Crack Initiation and Arrest	Project Leader: Dr. J.F. Kalthoff
Initiated (Date): 1.5.1973 / Completed (Date): 30.9.1976 Status: continuing / Last Updating (Date): December 1975	

1. General Aim

The general aim in connection with the reactor safety program is to improve the knowledge of the fracture behavior of reactor components in which cracks or cracklike defects have to be assumed.

2. Particular Objectives

The particular objective is the investigation of crack initiation and crack arrest phenomena. Stress analytical and materialspecific aspects are considered for components of the reactor pressure vessel steel 22 Ni Mo Cr 3 7. The tools of analysis are from linear elastic fracture mechanics.

3. Experimental Facilities and Programme

A wedge loading system (ideally stiff) to initiate cracks in double-cantilever-beam (DCB) specimens has been developed and is used together with a Cranz - Schardin high speed camera to measure velocities and stress intensity factors for propagating cracks.

The research program is carried out in 2 stages: a) In model experiments the principal influence of dynamic effects on the crack arrest process is investigated. b) Experiments with steel specimens are performed to measure crack initiation and especially crack arrest toughness values. In both cases the crack velocities prior to arrest - varying as a consequence of length and sharpness of the starter notches - are measured.

4. Project Status

4.1 Progress todate

a) Applying the shadow spot method dynamic stress intensity factors for propagating and subsequently arresting cracks in DCB-specimens (made from an epoxy resin - Araldit B) have been measured and compared to corresponding statically determined values. Several test series in which the crack velocity prior to arrest was varied from 15 up to 300 m/s were carried out.

b) Following the guide line of the model experiments a series of 18 DCB-specimens (25 mm thick) with different heat treatment of the material and different test temperatures has been tested for a number of crack initiation conditions. The experimental devices for measuring crack opening and crack velocity have been tested.

4.2 Essential Results

a) The model experiments showed that dynamic effects influence the crack arrest process and have to be taken into account in determining crack arrest toughness values and in applying a crack arrest safety analysis. In particular it was found: At the beginning of crack propagation the dynamic stress intensity factor K^{dyn} is smaller than the corresponding static value K^{stat}, at the end of crack propagation $K^{dyn} > K^{stat}$ and after arrest K^{dyn} oscillates around K^{stat} at arrest (see fig.1). Crack arrest toughness values K_{Ia}^{stat} determined conventionally under use of a static analysis depend on the crack velocity; only values determined by a dynamic analysis (K_{Ia}^{dyn}) seem to characterize the arrest capability of a material correctly.

b) The optimized wedge loading system with an anti friction insert allows for an undisturbed symmetric (controlled by means of stress coating techniques) loading of the steel specimens and a faultless measurement of the crack opening to calculate stress intensity factors. Substantial variations of the condisions of crack initiation and of the crack velocity prior to arrest could only be produced for the prevailing material and room temperature by hardening of the whole specimen or by using so called Duplex specimens which have an electron beam welded starter region made from an ultra high strength steel. In these cases the crack velocity strongly could be influenced by a variation of length and sharpness (still of poor reproducebility) of the starter notch. Then crack arrest has been observed after a few mm or some cm. The crack velocity could not be measured in a reliable manner by contact strip methods because of the effect of tunneling. But the momentary crack tip could be optically localized and registered by high speed photography. First evaluations gave values of maximum crack velocity between 30 and 150 m/s.

5. Next Steps

Further crack arrest toughness values for 22 Ni Mo Cr 3 7 in particular for different conditions of crack initiation and velocity prior to arrest will be determined and analysed in comparison to the results found by the model investigations.

6. Relations with other Projects

This project is closely related to RS 102-11, -13 and -14.

7. Reference Documents

Annual Reports in the series IRS - Forschungsberichte

A 73 IRS - F - 18

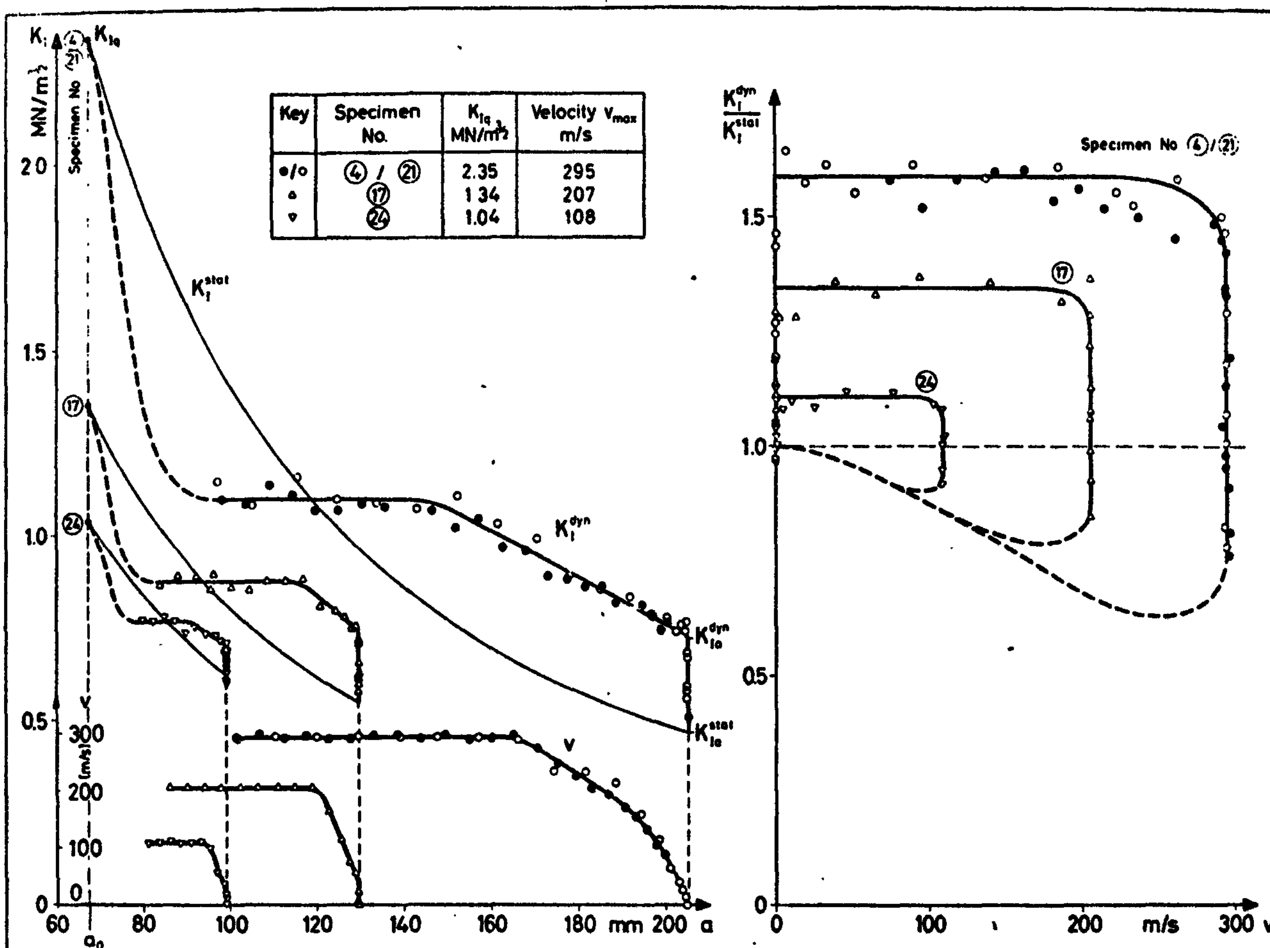
A 74 IRS - F - 24

J.F. Kalthoff; S. Winkler; J. Beinert "Dynamic Stress Intensity Factors for Arresting Cracks in DCB Specimens" will be published in the Inter. Journ. Fracture, April 1976.

8. Degree of Availability

The reports can be ordered from

Institut für Reaktorsicherheit der TÜV
Glockengasse 2, 5 Köln -1



Key	Specimen No.	K_{Ia} $MN/m^{3/2}$	Velocity v_{max} m/s
•/○	4 / 21	2.35	295
▵	17	1.34	207
▿	24	1.04	108

Fig. 1 Stress intensity factors for arresting cracks

Classification: 11.2.1	
Title 1 (Original Language): Rißaufweitungsmessungen COD/COS (RS 102-13 - II.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Gesell-schaft, IFKM, Freiburg
Title 2 (english): The Use of COD/COS Fracture Criteria in the Assessment of Component-Strength	Project Leader: Dr. J.G. Blauel T. Hollstein
Initiated (Date): ..y 1973 — Completed (Date): Dec. 31, 1975 Status: continuing — Last Updating (Date): December 1975	

1. General aim

The general aim is to improve the knowledge on the assessment of safety for components of a reactor plant, in which cracks or cracklike defects have to be assumed.

2. Particular objectives

The particular objective is the assessment of the fracture behavior of components by measuring the crack opening displacement as a function of geometry and load parameters and comparing it to the material specific values COD_c/ COS_c.

3. Experimental facilities and programme

The crack opening displacement is measured by mechano-electrical, photo-electrical and photographical means for Compact (C) and Single-Edge-Notched (SEN) specimens with different thicknesses and crack lengths up to fracture. The load-displacement curves are evaluated in terms of the crack resistance R and the energy integral J. The material is 22 Ni Mo Cr 3 7 corresponding to the ASTM 508 Cl 2.

4. Project status

4.1 Progress to date

The crack opening displacement has been determined experimentally for increasing loads along the full crack length for a series of C-specimens and SEN-specimens with different crack lengths and thicknesses at room temperature. Critical values of COD, G, J of the material at fracture have been deduced and their correlations have been examined.

4.2 Essential results

4.2.1 For the C-specimens the COD-values allow a reasonable definition of the crack opening stretch (= crack tip opening displacement) by linear extrapolation of the straight parts of the crack contour to the visible crack tip. This value COS^{lin} can be deduced from the clip gage reading V_g^{LA} at the points of load application if a simple rotation mechanism (plastic hinge model of Elliott and May,1968) is assumed:

(1) $$COS^{lin} = \frac{r(W-a)}{r(W-a)+a} \; V_g^{LA}$$

with W being the specimen width and a the crack length. The rotational factor r has been found to be independent of thickness and crack length and to be best described by

(2) $$r = 0.48 \sqrt{COS^{lin}} .$$

Fig. 1 compares this with results from other authors.

4.2.2 As a critical value the crack opening stretch COS_c^{lin} at the onset of the stable crack growth has been determined by the method of interrupted loading. We have found for C-specimens (thickness B=30 mm) from two different plates

$$COS_c^{lin} = 0.28 \pm 0.04 \text{ mm and } 0.42 \pm 0.04 \text{ mm} .$$

Up to now it has not been possible to derive the same detailed information from either magnetic flux or acoustic emission techniques.

4.2.3 For one series of C-specimens the J- Integral has been calculated according to Sumpter (1974)

(3) $$J = \frac{2.2\,U}{B\,(W-a)} .$$

U is the energy-represented by the area under the load-displacement record - to open the crack to a certain value. In accordance with existing theories a linear relationship between J and COS^{lin} is found :

(4) $J = \beta \sigma_{0.2} \; COS^{lin}$

where $\sigma_{0.2}$ is the yield stress. For the plastic constraint factor β the experiments give 1.51 ± 0.15. Using the critical value COS_c^{lin} = 0.28 mm one gets a critical value of the J-Integral J_c = 0.2 MN/m. From this value one can derive critical loads and flaw sizes, respectively.

5. Next steps

The results of the SEN-specimens will be evaluated and the programme will be supplemented by more experiments with specimens of the afore mentioned geometry. Critical COD/COS-values will be determined and the correlation to other fracture mechanics parameters will be investigated.

6. Relation with other projects

This project is closely related to RS 102 -11, RS 102-12 and RS 90.

7. Reference documents

Annual reports in the series IRS- FORSCHUNGSBERICHTE

A 73: IRS-F-18

A 74: IRS-F-24

8. Degree of availabitily

The reports can be ordered from

Institut für Reaktorsicherheit des TÜV
5000 Köln -1, Glockengasse 2

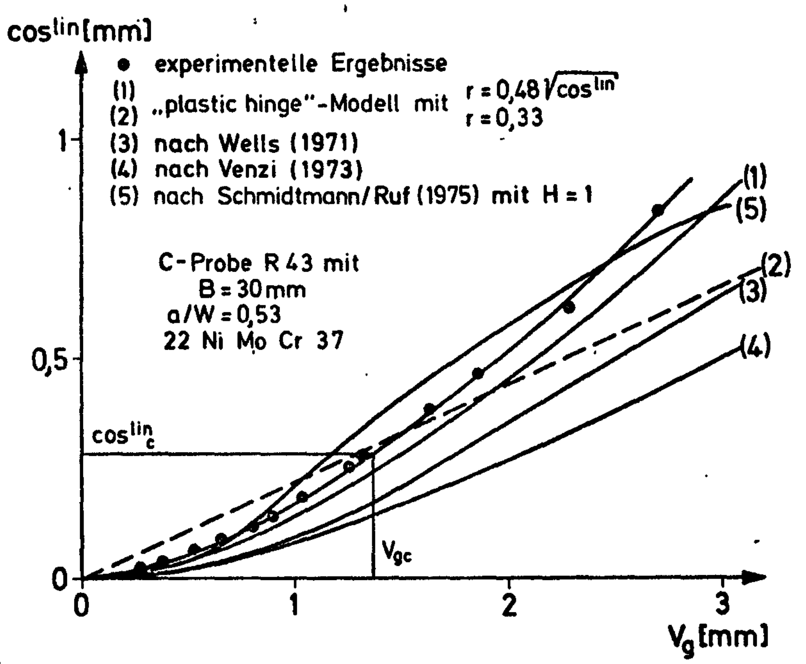


Fig. 1: Crack tip opening displacements according to different models

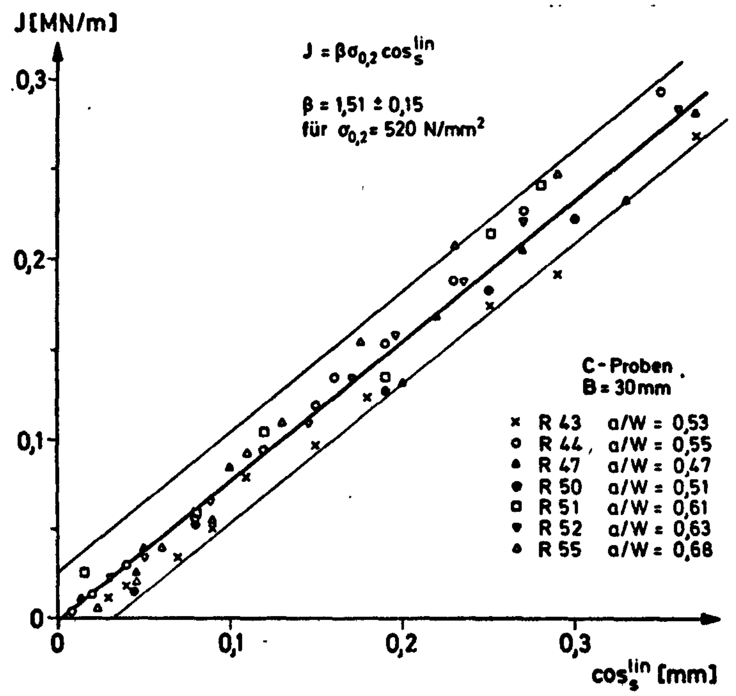


Fig. 2: Correlation between the J-Integral and the crack tip opening displacement

Classification: 11.2.1		
Title 1 (Original Language): Experimentelle Spannungsanalyse (RS 102-14 - II.2., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Gesellschaft, IFKM
Title 2 (english): Damage of Nuclear Reactor Components by Shock Wave Loading		Project Leader: S. Winkler
Initiated (Date): 1.5.1973	Completed (Date): 30.4.1976	
Status: continuing	Last Updating (Date): December 1975	

1. General Aim

In addition to static or quasistatic and thermal loading of reactor pressure vessels and components - which are very well investigated - extremely dynamic loading conditions (shock waves) may occur, for example by impinging fragments or projectiles, by internal explosions or - after certain cases of partial failure - by impact of the accelerated reactor pressure vessel against the containment. The investigation of damage in these dynamic cases is general aim of this work.

. Particular Objectives

The particular objective is the investigation of (1) the distribution of the produced dynamic stress fields (shock waves) (2) the interaction of these shock waves with the material and (3) the initiation of damage in the material.

3. Experimental Facilities and Programme

In this programme plane shock waves are produced by impinging a target with a flat impactor in a gas gun. Reflected shock waves cause tension fields in the target by which microcracks are nucleated and subsequently activated to grow further, thus leading to internal damage in the material. Metallographic methods are used to study the crack initiation process. The influence of material structure, temperature, and material history, e.g. heat treatment on the crack initiation process are investigated. The decay of the dynamic stress fields with time is of special importance, therefore stress pulse profiles are measured.

4. Project Status

4.1 Progress todate

Damage thresholds - especially for the commonly used steel 22 NiMoCr 3 7 - have been measured in relation to material structure, annealing and specimen temperature during impact. The nature of crack nuclei has been investigated.
The strength of the shock loaded (i.e. damaged) material has been investigated in various directions in tensile experiments.

Effort has been made in applying a new method for measuring the shock profile at the rear surface in order to get a better knowledge of the stress distribution at the instant of damage.

4.2 Essential Results

a) For steel, in particular the reactor pressure vessel steel 22 NiMoCr 3 7 the following results on the nucleation and growth of microcracks were found: Damage occured by exceeding a shock wave pressure theshold of approximately 25 - 26 kbar ($\approx$ 120 m/s impact velocity at 25^{o}C and about 20 - 22 kbar at 300^{o}). The exact value depends slightly on texture and annealing.
Two main types of crack nuclei were found: impurities (sulphides and oxides) and ferrite precipitatations, latter are less important for the reactor pressure vessel steel. Microcracks were nucleated either by brittle cleavage of grains or brittle cracking of grain boundaries. Ductile extension and coalescence of microcracks lead to macroscopic fracture, by which the integrity of the vessel or component may be endangered severely.

b) Tensile specimens were machined from the damaged zone of the targets impacted with different velocities. Especially because of cracks nucleated at flat inclusions parallel to the rolling plane the tensile strength perpendicular to the rolling plane decreases linearly by about 50 % by increasing impact load from 30 to 60 kbar. Controversely the tensile strength parallel to the rolling or the transverse direction decreases only by less than 10 % of the strength of the unloaded material.

c) Shock profiles are measured successfully by using the method of voltage generation at a bimetallic junction during pressure loading. From these measurements the stress distribution at the damaged plane inside the specimen can be extrapolated with a higher accuracy than before.

The conclusion which can be made at this status of the investigations is that the effective tensile stress at the damage plane in our experiments does not reach the value of the impact pressure. The reason is the finite risetime of the shockwave (e.g. 0.5 microseconds for an impact pressure of 45 kbar in a 22 Ni Mo Cr 3 7 sample of 6 mm thickness. That means that the material especially at lower impact velocities, where the risetime increases, will be damaged at even lower stresses than reported above.

5. Next Steps

The work in progress will be continued. This is in particular: Damage threshold values for the austenitic steel X 6 Cr Ni Mo 18.11 will be measured; shock wave behaviour will be investigated in more detail, to correlate the damage within the target with the actual load history in the respective plane.

6. Relation with other Projects

This work is related to the investigations 11, 12 and 13 within the project RS 102.

7. Reference Documents

Results are published in the IFKM-Report 8/74 "Shock Waves as Possible Causes of Damage to Nuclear Reactor Components", Dec. 1974

Annual Reports A 73: IRS-F-18
A/74: IRS-F-24 (Englisch)

8. Degree of Availability

The Reports can be ordered from

Institut für Reaktorsicherheit der TÜV
Glockengasse 2, 5 Köln - 1.

The IFKM Report can be ordered from Institut für Festkörpermechanik Rosastr. 9, 78 Freiburg

Classification: 11.2.1	
Title 1 (Original Language): Bestrahlungseinfluß auf Festigkeit und Relaxation von hochfesten austenitischen Stählen und Nickellegierungen für Verbindungselemente der Kernstruktur (RS 133 - II.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Irradiation Influence on Solidity and Relaxation of Austenitic Steels and Ni-Alloys for Core Structure Connections	Project Leader: H. Debray
Initiated (Date): 1. 2. 75 — Completed (Date): 31. 8. 76	
Status: Qontinuing — Last Updating (Date): 31. 12. 75	

General Aim and Particular Objectives

The influence of neutron irradiation on the solidity and relaxation of materials for core structure connections of light water reactors has to be investigated.

Experimental Facilities

Irradiation in special irradiation channels in the Obrigheim Power Reactor (KWO). Post irradiation testing in the hot cell facility of KWU, Erlangen. Testing of unirradiated samples in the materials testing laboratory of KWU, Erlangen. Machines used for this program: Tensile testing machine and high-frequency pulsator (Fatique tester).

Research Program

For three characteristic materials and two heat-treatment conditions of each quantitative experimental results are expected:

1. Steel Nr. 1.4550, representative for austenitic steel as used in LWR's.
2. Inconel-X-750, representative for connection material (frequently used in USA).
3. Steel Nr. 1.4980 (USA: A 286) with lower tendency towards selective corrosion and lower thermal expansion coefficient compared with Inconel-X-750.

Project Status/Progress to Date

The irradiated samples were dismantled in the hot cell, assorted and identified. The monitor capsules were opened and inspected. As a first step the temperature monitors and neutron flux monitors (Nb and Fe) were investigated and evaluated. The residual spring power of the relaxation samples was measured. Tensile stress was determined on some samples. An oscillation fatique test was conducted on a nonirradiated auxiliary sample.

Project Status/Essential Results

The auxiliary sample test showed, that the samples will preferably break on the thread of the screw and not on the shaft.
The inspection of the temperature monitors showed, that the irradiation temperature was lower than 304 °C; because no monitor was molten.
Preliminary results from the irradiated samples point out, that the relaxation increases under irradiation more than under temperature (at 300 °C).

Next Step

Evaluation of the results from the irradiated samples. Determination of the fatique behaviour from tensile stress data.

Relation with Other Projects

No relations with other projects.

Reference Documents/Degree of Availability

No reports available.

Classification: 11.2.1	
Title 1 (Original Language): Literaturstudie: Selektives Korrosionsverhalten von in Leichtwasser-Reaktoren eingesetzten Werkstoffen (RS 159 - II.2, A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Literature Study on Selective Corrosion of LWR-Materials	Project Leader: Stieding
Initiated (Date): April 1975 Status: Continuing	Completed (Date): June 1976 Last Updating (Date): December 1975

General Aim

Some materials (austenitic, ferritic and martensitic steels and Ni-alloys) which are used in the nuclear steam generating system are to be investigated, whether they can be attacked by selective corrosion under certain conditions. As a first step a literature study will be conducted on the behaviour of some selected materials.

Particular Objectives

In order to raise the safety of operating reactors and to get more information about the risks of longtime effects the knowledge of the behaviour of primary materials has to be extended over the limits of the usual operation lifetime data.

Experimental Facilities

No experimental facilities necessary.

Research Program

In the first step the following details will be studied from the literature:

- corrosion under RPV-cladding, caused by material failures

- influence of the corrosion on the fatique behaviour and the crack growth

- hydrogen induced corrosion

- determination of the corrosion limits of materials containing Ni

- influence of gaps and different material combinations

- electrochemical investigations on the corrosion mechanism.

Project Status / Progress to Date

A literature study of the hydrogen induced corrosion and vibration induced corrosion was performed. The literature was evaluated with respect to the influence of gaps on the corrosion resistance of reactor materials.

Project Status / Essential Results

A first result was that the corrosion process in gaps can be increased by diffusion of ions into the gaps, leading to concentrated pollution of the water in the gaps. This procedure depends on the geometry of the gap and the existing electrochemical and chemical relations.

Next Steps

The literature study will be concentrated on more quantitative values of the electrochemical and chemical relations in a gap.

The program will be continued corresponding to the research program.

Relation with Other Projects

No relation with other projects.

Reference Documents / Degree of Availability

No reports available.

Classification: 11.2.1		
Title 1 (Original Language): Fremdstoffe in Leichtwasserreaktorkühlmitteln (PNS 4123 - II.2., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Corrosion Substances in Light Water Reactor Coolants		Project Leader: J. Michael
Initiated (Date): 1974	Completed (Date): 1977	
Status: continuing	Last Updating (Date): December 1975	

1. General Aim

The metal loss of primary loop materials to pressurized water is examined with a view to reducing the primary loop contamination of pressurized water reactors. This examination mainly relates to the INCOLOY 800 steam generator material.

2. Particular Objectives

A lower primary loop contamination is expected from the reduction of metal loss mainly by means of water chemistry measures.

3.1 Experimental Facilities

A high pressure autoclave system with a volume of 4 liters is available. The inner surface of that autoclave is coated with a thin layer of gold.

Atomic absorption spectrometry is used for the water chemical analysis. The surfaces of test samples are examined according to the solid-state analytic methods of the secondary ion and Auger electron spectroscopy.

3.2 Research Program

The main components of the alloy - iron, chromium and nickel - released from INCOLOY 800 to pressurized water - are determined by 500 h tests at 342°C and 150 bar. The water chemistry conditions are varied during this process.

4. Project Status

In several test runs the oxygen content of the pressurized water was lowered by means of thermal degasing, flushing with noble gas, and addition of hydrazine.

4.1 Progress to Date

The measures described above brought about a reduction by one power of ten of the deionate oxygen content by one power of ten.

4.2 Essential Results

In the process of lowering the oxygen content the metal content in the pressurized water was reduced by 33% of iron after 500 hours at 342°C and 150 bar. The metal content of nickel decreased by 40% while the chromium content could not be well defined due to the very low intitial values which were within the range of the detection limit of the atomic absorption spectroscopy.

The surface layers formed on the INCOLOY samples after 500 h under operating conditions exhibited oxide, hydroxide and hydride fractions of the main components of the alloy. As compared to the fractions of the other main alloy components, there is a noticeable enrichment in the chromium content in the surface layer so that such surface can be assumed to behave chemically as a protective layer.

5. Next Steps

The investigations of the metal loss to pressurized water are supplemented by variation of the pH-value of the pressurized water. Also, the solid content in the pressurized water will be especially analyzed.

To speed up the program, a second autoclave system is presently integrated.

6. Relations with Other Projects

Work is performed in contact with reactor operators and reactor manufacturers.

7. Reference Documents

G. Bechtold, I. Michael, R. Prümmer
Zur Gold-Innenbeschichtung von Autoklaven durch Explosivplattieren (in German)
Metall 29 (1975) H. 7, pp. 685 - 687

8. Degree of Availability

Preprint available at GfK, Karlsruhe

Classification 11.2.1

Title 1	Metal Fracture (Crack Growth)	**Country** UK **Sponsor** UKAEA **Organisation** Reactor Fuel Laboratory Springfields
Title 2		
Initiated 1971 **Status** Continuing	**Completed** **Last updating**	**Project leaders** Dr. B. Tomkins

1 General Aim

To recognise and assess modes of failure of nuclear reactor pressure vessels.

2 Particular Objectives

To examine the possibility of component defects growing to a critical size as a result of reactor transients.

3 Experimental Facilities and Programme

A number of special purpose machines have been assembled, which permit specimens of about 1" section to be stress cycled and thermal cycled under well-controlled conditions.

4. Project Status

(a) Cladding integrity. Crack propagation tests have been performed on 304 stainless steel at room temperature under push-pull loading conditions to derive basic crack growth data under elastic and plastic straining conditions. The information from these tests is being used to assess whether under-clad cracks can be expected to propagate through the cladding as a result of service cycles. Further tests on weld metal cladding/base metal compound specimens are being considered.
(b) Fatigue properties of base metal - A programme of work is continuing to assess the fatigue crack growth properties of pressure vessel steel. A series of tests have been done on Ducol steel (owing to the scarcity of A533-B) and crack growth has been studied in both elastic and plastic regions under both push-pull and zero-tension loading. The study of the growth of small cracks has enabled correlation to be made between crack growth data and the more conventional S-N data used in nuclear vessel codes such as ASME III. Further work will consider the effect of stress gradient and environment on crack propagation at relevant transient frequencies. Recent US work has shown marked environmental effects on crack growth rate.
(c) Thermal-biaxial fatigue - A rig is being developed to study crack propagation under thermal shock loading with varying constraint. As reactor transients are mainly of a thermal nature, most cycling strains which cladding experiences are from this source and are biaxial in nature. These laboratory tests on thermal shock loading should give results for use in the fracture models.

Reference Documents

Internal documents.

Classification 11.2.1

Title 1 Metal Fracture (Stability)	**Country** UK
Title 2	**Sponsor** UKAEA **Organisation** Reactor Fuel Laboratory Springfields
Initiated 1971 **Completed** **Status** Continuing **Last updating**	**Project leaders** Dr. B. Tomkins

1 **General Aim**

To recognise and assess modes of failure of nuclear reactor pressure vessels.

2 **Particular Objectives**

To develop a way of producing a sharp crack in a component, to enable the critical crack length to be measured.

3 **Experimental Facilities and Programme**

A high pressure technology exists at Springfields; steady pressures of 100,000 psi (6000 bar) may be employed and cycles of about half that magnitude. This has been applied to locally fatigue crack the base of a notch and thus produce well-defined crack starter conditions.

Reference Documents

Internal documents.

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Clansifioution 11.2.1

Title 1 Vessel Failure Tests	**Country** UK
Title 2	**Sponsor** UKAEA **Organisation** REML Risley
Initiated 1969 **Completed** **Status** Continuing **Last updating**	**Project leaders** Dr. A. Cowan

1 General Aim

To quantify critical crack length for use in decisions arising in inspection.

2 Particular Objectives

To compare K_{1C} and COD correlations for failure of intermediate-size pressure vessels above valid K_{1C} critical temperatures.

3 Experimental Facilities and Programme

The test vessels are 5' Diam. and 12' long. Thicknesses to date, 1" and 3". Currently defects (partial and full thickness) are placed in the membrane region.

4 Project Status

Investigation of the factors governing cyclic crack growth and fracture in pressure vessel steels is continuing. Two tests have been completed in a series on a 3" thick x 5' diam. vessel to investigate the effect of plate thickness. The tests were made at temperatures of 84 and 40°C corresponding to the upper end and middle of the ductile/brittle transition range. A test at normal ambient temperature was then performed. After curring an axial through wall thickness slit, the vessel was pressure cycled to establish crack growth rates and then pressurised to failure. The test will also be used to assess the usefulness of acoustic emission monitoring for detecting crack growth and proximity to failure.

Reference Documents

Internal documents.

Classification 11.2.1

Title 1 Mechanics of Plane Strain Fracture	**Country** UK
Title 2	**Sponsor** UKAEA **Organisation** Fuel Element Labs. Springfields
Initiated 1972 **Completed** **Status** Continuing **Last updating**	**Project leaders** Dr. B. Tomkins

1 General Aim

To recognise and assess modes of failure of nuclear reactor pressure vessels.

2. Particular Objectives

To measure the deformation in a thick cracked plate up to the point of fracture.

3. Experimental Facilities and Programme

A rig has been constructed in which a high pressure (circa 3000 bar) may be applied within a crack in the plate.

The use of internal pressurisation means that fracture criteria in thick section material can be studied under well defined boundary conditions. Direct surface strain measurement around the crack tip will be done and fractography employed to assess fracture development prior to fast fracture propagation.

Reference Documents

Internal documents.

PROJECT TITLE : Fracture Mechanics for Ductile Materials	CLASSIFICATION 11.2:1
SPONSORING COUNTRY : ITALY	ORGANISATION : PISA - UNIVERSITY
DATE INITIATED : 1968 DATE COMPLETED : 1977	PROJECT LEADER : C. CARMIGNANI

Description :

The aim of this research program is to evaluate the possibility of using linear elastic fracture mechanics concepts to forecast the starting of crack propagation in a ductile material.

In the first phase it was developed a theoretical work to evaluate the elastic stress field around the tip of the crack (Ref. 1.2). After, several cylindrical pressure vessels of diameter 100 ÷ 200 mm, thickness 5 ÷ 6 mm, with on axial through the thickness cracks of length 60 ÷ 100 mm, were bursted by internal pressure (Ref. 3.4.5). The results of this experimental work demonstrated the possibility of using cylindrical pressure vessels, like those above mentioned, to determine the critical parameters (K_c or COD) of the material.

Now is under execution an experimental work devoted to evaluate size effect on the determination of the critical parameter with the method above indicated.

Facilities used are a pressurization system with a manual pump for bursting test at room temperature, using oil, and an electric pump for bursting test at low temperature (-170 °C), using liquid nitrogen.

The research program was initiated with a research contract with the CNEN and now it is presented with an analogy contract with the CNR.

REFERENCES

1. C. CARMIGNANI, A. DEL PUGLIA
Applicazioni di metodi di calcolo elettronico all'analisi del comportamento elastico di piastre rettangolari contenenti una fessura passante centrale, caricate normalmente al loro piano medio.
Atti Ist. Mecc. Appl. - Univ. PISA, n° 11 - Anno Acc. 1968-'69.

2. C. CARMIGNANI, A. CELLA
Elastic analysis of cracked thin shells by the finite element method.
Proc. 1st SMIRT, Berlin, 1971, vol. 5 pp. 57÷76.

3. C. CARMIGNANI
Ricerca teorica e sperimentale sulla propagazione delle rotture nei recipienti in pressione.
Atti del Convegno Nucleare di PISA, vol. 1, CNEN, 1971, pp. 279÷319.

4. C. CARMIGNANI, S. REALE
Elaborazione dei risultati della ricerca sperimentale sulla propagazione delle rotture nei recipienti in pressione.
Atti Istituto Impianti Nucleari - Univ. PISA. RP 153(73), 1973.

5. C. CARMIGNANI, P. CIBECCHINI, S. REALE
Caratterizzazione di un acciaio al Nichel per impieghi a bassa temperatura con i criteri della meccanica della frattura.
Proc. 5th Int. Conf. on Exp. Stress Analysis, Udine, 1974, paper 18 pp. 2.5 ÷ 2.12.

TNO - metaalinstituut	CLASSIFICATION: 11.2.1
TITLE: Literatuurstudie - (1) Invloed van omgevingscondities op vermoeiïngsscheurgroei; (2) Invloed van veroudering op de mechanische materiaal-eigenschappen.	COUNTRY: NETHERLANDS SPONSOR: Ministry of Social Affairs ORGANIZATION: TNO-metaalinstituut
TITLE (ENGLISH LANGUAGE): Literature study - (1) Influence of Environment on Fatigue crackgrowth; (2) Influence of aging on fracture related material properties.	PROJECTLEADER: Van Rongen
INITIATED: June 1974 COMPLETED: mid 1976 STATUS: near completion LAST UPDATING: February 1976	SCIENTISTS:

1. General aim

   To improve the understanding of material behaviour during lifetime of nuclear vessels and other pressurized components as influenced by aquous environment and aging effects. A "state of the art" report will be submitted.

2. Particular objectives

   To review the knowledge in the areas
   - influence of reactor coolant on fatigue crack-growth in LWR pressure vessels, with particular attention to chemical composition of the coolant, type of pressure vessel steel, frequency, amplitude and threshold phenomena. Interpretation will be in fracture mechanics terminology.
   - phenomena which deteriorate the fatigue and fracture related material parameters of nuclear grade pressure vessel steels, such as thermal aging and high strain aging.

3. Experimental facilities and programme

   Not applicable.

4. Project status

   1. Progress to date : substantial part of literature survey completed
   2. Essential results: to be reported mid 1976.

5. Next steps

   Systematic analysis of data from literature to provide final conclusions.

6. Relation with other projects: -

7. Reference documents: -

8. Degree of availability

   Through Ministry of Social Affairs, C.R.V., Postbus 69, Voorburg, The Netherlands.

9. Budget Hfl. 50.000,-

10. Personnel 2 scientists

NIL & MI - TNO	CLASSIFICATION: 11.2.1 11.2.3
TITLE: Literatuurstudie van materiaalparameters die nodig zijn bij de interpretatie van defect-localisatie met AE-apparatuur.	COUNTRY: NETHERLANDS SPONSOR: Ministry of Social Affairs and others ORGANIZATION: NIL, MI - TNO
TITLE (ENGLISH LANGUAGE): Literature survey of material parameters which are necessary for the diagnosis of defects localized with AE techniques.	PROJECTLEADER: Steenhuizen
INITIATED: September 1974 — COMPLETED: September 1976 STATUS: in progress — LAST UPDATING: february 1976	SCIENTISTS:

1. General aim

This particular survey is a preliminary study of effects to be included in an extensive study which will be initiated in the future and which can be characterized as follows:
"Evaluation of the practical application of Acoustical Emission (AE) techniques during construction, testing and operation of welded constructions, in particular pressurized components to be used in the energy and process industry in order to improve safety, reliability and economic construction of components".
Optimum utilization of AE-apparatus for defect localization and diagnosis is only possible f sufficient data on the AE behaviour of constructural material are available (AE material parameters). This study comprises the use of AE material parameters for nuclear vessels and their application to the diagnosis of defects localized in experiments.

2. Particular objectives

Study and inventarization of AE research in literature.
Visits to industry and institutes in the Netherlands.

3. Experimental facilities and programme: -

4. Project status: just started.

5. Next steps: -

6. Relation with other projects: see under 1.

7. Reference documents: NIL-lastechniek, 40^e jaargang, no. 5, mei 1974.

8. Degree of availability

Through Nederlands Instituut voor Lastechniek.

9. Budget: Approx. Hfl. 30.000,-

10. Additional information

Survey will be executed by Metaal Instituut (TNO) and will take approx. 4 months.

	Classification 11.2.2/11.2.3.
Title 1 Spændingsanalyse af primære trykbærende stålkomponenter: Sammenligning mellem beregnede og målte tøjninger og spændinger i en BWR-pumpestuts.	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2. Stress analysis of primary steelcomponents: A comparison between calculated and measured stress and strain in a BWR main circulation pump nozzle.	Project leader S.I. Andersen Scientists:
Initiated (date) 1974.04.01 — Completed: (date) 1976 Status: progressing — Last updating (date) February 1976	S.I. Andersen P. Engbæk S. Krenk

1. General aim.

The purpose of the project is to evaluate the accuracy, which can be obtained by stress analysis of a complicated pressure vessel component, and to determine the degree of sophistication, required in such calculations.

2. Particular objectives

The pump nozzle in a BWR-steel pressure vessel has been selected as object for this investigation. The nozzle is located in the transition zone between the spherical bottom head and the cylindrical vessel part. The nozzle axis is parallel to the centerline of the vessel.

3. Experimental facilities and programme

Comprehensive strain measurements were performed on the pump nozzle during the manufacturer's hydrotest of the vessel in september 1975.

4. Project status

4.1. Progress to date

A 3-dimensional finite element model of the nozzle has been generated, and 3 load cases run: hydrotest, stresses due to stationary temperatures, and stresses during normal operation conditions.

Strain measurements have been performed on a pump nozzle as mentioned above.

4.2. Essential results

The stresses and strains due to the above mentioned loads have been established and compared to hydrotest strain measurements.

5. Next steps

Evaluation of the comparison between the calculated stresses and the measured strains.
Strain measurements on a second BWR vessel during hydrotest.
Documentation of the investigations.

6. Relations with other projects

The work is of importance to the Risø project, dealing with probabilistic fracture mechanics, where information about the degree of confidence of calculated stresses are needed (classification 14).

7. Reference documents

None released up to now.

8. Degree of availability

Summary report will be published. Detailed project information is restricted.

PROJECT TITLE : Structural and acoustic vibrations of piping systems	CLASSIFICATION 11.2.2
SPONSORING COUNTRY : ITALY	ORGANISATION : UNIVERSITY OF PISA
DATE INITIATED : 1973 DATE COMPLETED : 1976	PROJECT LEADER : A. DE PAULIS

Description :

Research program:

A multi-purpose computer program has to be developed with the aim to study structural and acoustic vibrations of piping systems.

This program will be useful to minimize objectionable piping vibrations at resonance and to control and/or eliminate excessive vibrations developed in service. An experimental research in the same field has been planned to check program evaluations.

Facilities:

IBM 370/168 Computer belonging to CNUCE of Pisa.

Reference documents:

1. C. CARMIGNANI, A. CELLA, A. DE PAULIS
Structural Dynamics by Finite Elements: Modal and Fourier Analysis
Proc. 2nd Int. Conf. S.M.i.R.T. - Berlin 1973.

Classification 11.2.3.	
Title 1 Anvendelse af Akustisk Emission	COUNTRY Denmark SPONSOR DAEC et. al. ORGANIZATION DAEC Risø Danish Wel. Inst.
Title 2 Industrial application of Acoustic Emission	Project leader Arved Nielsen
Initiated 1970 Completed Status : progressing Last updating	Scientists: W. Swindlehurst S.E. Iversen N. Thorp

1. General aim

Increase of the effeciency of non-destructive control and surveillance by application of AE methods.

2. Particular objectives

Development of a system to store on magnetic tape information from a large number of AE transducers in order to do a careful location analysis after testing or surveillance.

Development or testing of commercially available discriminators to operate with the above mentioned storing system or to operate in connection with other applications of AE than location.

3. Experimental facilities and programme

Suitable tape recorders and suitable electronic equipment is available. Large steel structures in which the need for advanced surveillance is severe are present particularly within the conventional power production.

Laboratory experiments are planned in order to provide information on materials properties with respect to acoustic activety. This is to form basis for further development of discriminators.

4. Project status

The storing system has been finished and is ready for field use when a set of discriminators have been build.

A more simple system for surveillance of smaller pressure vessels during periodic inspection is operating under field conditions.

5. Next steps

Development of more advanced discriminators and development of software to handle the information stored on tape.

6. -

7. Reference documents

A. Nielsen: Acoustic Emission Surveillance Methods. Risø Report No. 277.

A. Nielsen: European Progress Report on Acoustic Emission. July 6, 1974. IIW X-752-74.

8. Degree of availability

For commercial reasons the availability of information of the storing system might be restricted.

Classification: 11.2.3

Title 1 (Original Language): Zerstörungsfreie Wiederholungsprüfung an Reaktordruckbehältern mittels Wirbelstromverfahren (RS 89 - II.3.2; Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Non-Destructive In-Service Inspection for Reactor Pressure Vessels with Eddy Current Methods	Project Leader: Gräbener
Initiated (Date): December 1972 Status: Continuing Completed (Date): March 1976 Last Updating (Date): December 1975	

General Aim and Particular Objectives

The eddy current method was to be qualified for the examination of cracks on the surface of cladded reactor pressure vessels. Appropriate conditions, detectors and procedures of registration for this method were developed. The limits for detecting flaws on sections under water have been investigated.

Experimental Facilities and Program

The program was devided into two main parts:
In a pre-program first tests were carried out with existing detectors and evaluation equipment concerning the detection of flaws depth, considering thereby possible interference effects (detector-surface distance, material inhomogenities). The pre-program showed which detectors and equipment were the most appropriate.
In the main program apropriate detectors and evaluation equipment were developed according to the know-how obtained in the pre-program. With this prototype experiments were carried out under defined conditions. Data registration and processing were equally developed, as well as a system for carrying the detectors.

<u>Project Status and Progress to Date</u>

The main tests are finished. Various laboratories (KWU, IzfP and Inst. Dr. Förster) have examined vibration induced flaws and cuts in the weld deposit cladding of the test probes.

Parameters with respect to flaw geometry were:
- length, depth and breadth of flaws and cuts
- expansion along or transverse to the welded zone of the deposit cladding
- partial expansion in austenitic and ferritic material

Parameters with respect to disturbing influences were:
- plate materials (δ-ferrite content)
- welding parameters (cracks under plate)
- wavy plates (wobble effect)

All the investigations were carried out under laboratory conditions. The transfer of the results on power station conditions was seen to be possible. Of some special interest was the influence of water during underwater inspections, the pressure stable detector device and the influence of contaminated materials. It was calculated that the conductivity will not be influenced markedly after neutron irradiation and the permeability will not be changed.

<u>Essential Results</u>

The evaluation of the results is still under work. The first impressions are:
- the eddy current method can be used for the inspection of austenite cladded ferrite steel; 15 - 20 mm deep material cracks which start from the surface can be detected. Under certain conditions even covered cracks can be found
- disturbing influences will be eliminated by suitable frequencies and detector arrangements
- the eddy current method gives rather detailed information of the expansion and type of the material separation when the disturbing influences (structure, surface) are well known.

Next Steps

Documentation of the results. The measuring technique has to be improved, visual inspection device and simple recording are not sufficient for rational inspections. Automation will be necessary.

Relation with Other Projects

RS 27 "Development of Non-Destructive Inspection Techniques for In-Service Inspection Tests of Reactor Pressure Vessels"

Reference Documents and Degree of Availability

No reports available.

Classification: 11.2.3	
Title 1 (Original Language): Studie zur Anwendung der akustischen Holografie bei der zerstörungsfreien Prüfung von Reaktorkomponenten und Reaktoranlagen (RS 102-20/2 - II.3.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Gesellschaft, IzfP
Title 2 (english): Studies Concerning the Application of the Acoustical Holography in Nondestructive Testing of Reactor Pressure Vessels	Project Leader: Dr. V. Schmitz
Initiated (Date): 1.8.1974 Completed (Date): 30.11.1976 Status: continuing Last Updating (Date): December 1975	

1. General aim

In non-destructive testing they have not yet succeeded in reaching an identical three-dimensional picture of flaws, respectively of greater areas with defects, by conventional ultrasonic methods. An exact knowledge of these flaws is necessary to determine the influence of such areas on the stability or safety of reactor pressure vessels.
The acoustical holography links the conventional method of pulse-echo with the attitude of optical holography and allows non-destructive testing and a geometrically identical projection of flaws concerning either the basic testing or the repetitional testing.

2. Particular objective

We are working at two kinds of application:

- testing of the basic material, the cladding and the interface between austenitic and ferritic junctions of the reactor pressure vessels,
- testing of thick-walled components.

For determination of exact size and locus of international flaws it is very important to have an appropriate choice of parameters like frequency, long- of shear-wave, inclination angle, number of wavelengths of the emitting pulse, repetition rate, diminuishing of the hologram, inclination of the reference source etc.
For further developing of practical applications we are going into laboratory tests working with models.

3. Experimental facilities and programme

This project is divided into the following two points:

1) study of literature

2) development of a test system.

4. Project status

4.1 Progress todate

The results of the first part have been published in a report (see reference documents).
In the meantime first experiments have been done (see reference documents).
An acoustical holographic equipment has been installed.

4.2 Essential results

Beside tests by holographic methods, experiments with focusing probes have been carried out which allow to show an isometric projection of a three-dimensional object from different views. This principle is called D-scan.
Concerning ultrasonic holography we have done systematic laboratory experiments by changing frequency, inclination angle, size, orientation and depths of artificial flaws. The frequency was changed between 10 mm and 180 mm, the orientation from 90° to 45°. In all cases the resolution capability of holography has been demonstrated.

5. Next steps

We shall continue with systematic experiments at test-specimens with artificial and natural defects, experiments with focussing probes, examination of the mathematical and physical possibilities for numerical reconstruction.

6. Relations with other projects

none

7. Reference documents

/1/ Acoustical holography, Vol. 6. Ed. by Newell Booth. (New York: Plenum Press 1975)

/2/ Schmitz, V.:
Studie zur Anwendung der akustischen Holographie bei der zerstörungsfreien Prüfung von Reaktorkomponenten und Reaktoranlagen. Teil 1: Literaturrecherche.
IzfP-Bericht Nr. 740106-TW zu RS 102 - 20

/3/ Schmitz, V.:
Eignung des Holscan 200 der Firma Holosonics zur Fertigungs- und Wiederholungsprüfung an Reaktorkomponenten und -anlagen. Teil 2: Prüfsystemanalyse.
IzfP-Bericht Nr. 750207-TW zu RS 102 - 20

/4/ Quarterly report in the series IRS-Forschungsberichte, Report period Jan-Dec 1975 (IRS-F-28)

Classification: 11.2.3	
Title 1 (Original Language): Durchführung von Untersuchungen zur Rißerkennung an druckführenden Reaktorbauteilen mit Hilfe der optischen Holografie (RS 132 - II.3.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: TU-Hannover
Title 2 (english): Crack Detection in Pressurized Vessels and Reactor Components by Optical Holography	Project Leader: Prof. Dr. Mayinger
Initiated (Date): ıly 1974 — Completed (Date): June 1978	
Status: continuing — Last Updating (Date): December 1975	

1. General aim

The aim of the activities is to develop a quick and reliable optical method in order to detect cracks in reactor vessels during fabrication and later in repeating tests.

2. Particular objectives

In the last few years a new nondestructive technique - holographic interferometry - was developed by which one can detect material defects. Up to now it was, however, ›nly used for nonmetallic components.

Our aim is to apply this technique also for the testing of steel. The special objective of this investigation is to detect cracks in or near the surface of reactor vessel steels using surface waves which are recorded by double-pulse holography.

3.1 Experimental facilities

The principle of this holographic technique is to make visible the deformation of surface waves due to small irregularities in the material. This waves cause surface deformations which can be measured with holographic interferometry.

To produce the waves it is usually necessary to strike the testpiece at a small point. This impact excitation may be generated e.g. by a free falling steel ball, by the bullet of an air-gun, or an electromagnetic stress wave generator. The local impact is only a short-time excitation after which the waves will travel in all directions. In order to record this very fast event holographically a Q-switched ruby laser must be used which produces two short light pulses. The first giant pulse illuminates the testpiece shortly before the impact and thus the testpiece in its unstrained condition is recorded. The second laser pulse appears at a certain time after the impact to make visible the propagation of the wave. The whole experimental set-up was already discussed in the annual report A 74.

3.2 Research program

To detect cracks with the described method it is necessary to strike the testpiece in such a manner, that the amplitude of the wave has a minimum value of 1/2 λ (λ = 694 nm, wavelength of the ruby laser light). In order to estimate the influence of the various parameters on this amplitude the surface deformation after a single point excitation was at first studied theoretically.

Experimental investigations were made subsequently to verify the theoretical results. After these experiments several types of defined cracks in testpieces of simple geometry were examined in order to find the smallest crack that can be detected with this holographic method. Further experiments shall be carried out with testpieces of any geometric shape. After these preliminary investigations experiments will be made with real reactor components like pipelines, elbows, T-pieces, etc. to proof the reliability of this new nondestructive test-method. For these measurements in a nuclear power plant during a repeating test the holographic set-up must be reconstructed to a compact and versatile device.

4. Project status

4.1 Progress to date

The theoretical and experimental investigations have been made in order to describe the surface deformation which depends on the impact energy, the thickness of the plate and some impact parameters like E-modulus and the mass ratio of the striking ball and the plate. The experiments with the testpieces of simple geometric shape (steelplates of different thickness) and cracks of various length have been partially completed. In addition, basic investigations were made for the construction of a stress generator which allows the easy variation of the impact parameters in a wide range. A model of such an electromagnetic stress generator was built and is now under test. This device operates by discharge of a high-energy, high voltage capacitor bank. In its present configuration the impact force rises from zero to maximum in 5 µs and decays to approximately zero in another 5 µs.

Simultaneously the whole holographic set-up was rebuilt from the first optical bench construction into a device that is compact enough to be readily moved. The illuminating wave can be turned around the axis of the laser beam and thus allows the recording of heavy objects in any given position.

4.2 Essential results

After an impact the maximum of the surface deformation of a plate concerning the first mode vibration can be described with the following equation

$$U_{o_{max}} = 2\sqrt{\frac{2\,E_s}{D(1+\frac{\gamma^2}{\chi}\frac{M}{m})(1+\frac{\chi}{\gamma^2}\frac{m}{M})}} \qquad (1)$$

$U_{o_{max}}$ = Deformation at the point of impact
E_s = Energy of the impact
D = Function of static deformation
γ, χ = Deformation parameters
m = Mass of the striking steelball
M = Mass of the plate

The function of $U_{o_{max}}$ versus the impact-energy is shown in fig. 2 in comparison to the experimental results measured with holographic interferometry. In this technique the surface deformation becomes visible in form of an interference pattern. From this the amplitude at any given point of the surface may be computed using the following equation (fig. 1).

$$U = \frac{n\,\lambda}{2(\cos\theta_1 + \cos\theta_2)} \qquad (2)$$

n = Number of fringes
λ = Wavelength of the laser-light
θ_1, θ_2 = Angles of the optical illuminating set-up

In fig. 2 it can be seen that the correspondence between the theoretical and experimental results is very good. However, the deformations, measured in the experiments are always a little smaller than the theoretical values. This is due to the fact that an ideal elastic stroke cannot be realized in the experiments.

After finishing the investigations of flawless components cracks were cutted into the steelplates to measure the surface deformation near that material defects after a wave-exciting impact. The arrangement of the cracks is shown in fig. 3. With this crack-configuration it was possible to determine the approximate length of the detectable crack by means of this new holographic test-method. As can be seen in figs. 4,5,6,7 in all the different plates the 10 mm-crack can be realized clearly because it creates a good visible irregularity in the interference pattern. After this positive results, cracks, even smaller than 10 mm were examined (fig. 8). The interference pattern obtained from this cracks are shown in fig. 9. The irregularity of the fringes is of course smaller than in the case of the 10 mm-crack, but it is yet possible to detect cracks down to a length of 2 mm in up to 40 mm thick plates.

5. Next steps

Future investigations will be made with cracks of different shape and depth and here after with natural cracks. As can be seen in fig. 2 the surface deformation of a 40 mm thick steelplate is very small. For testing the even much thicker walls of reactor pressure vessels it is necessary to increase the sensitivity of the test method. This might be possible with an electronic interpolation proceedure which allows the measurement of surface deformations in the range of about 1/100 λ (~ 7 nm).

6. Relation with other projects

There are some more nondestructive test methods like ultrasonic inspection, radiography, acoustic emission etc. which will be applied to the testing of reactor vessel steels. The new holographic technique of detecting the motion and disturbance of surface waves near cracks is an additional test method of high sensitivity and accuracy.

7. Reference documents

Annual report A 74

Quarterly reports in the series IRS-Forschungsberichte

Report-period	Jan. 1975 - March 1975	IRS-F-
Report-period	April 1975 - June 1975	IRS-F-
Report-period	July 1975 - Sept. 1975	IRS-F-
Report-period	Oct. 1975 - Dec. 1975	IRS-F-

8. Degree of availability

The quarterly reports are available by Institut für Reaktorsicherheit (IRS).

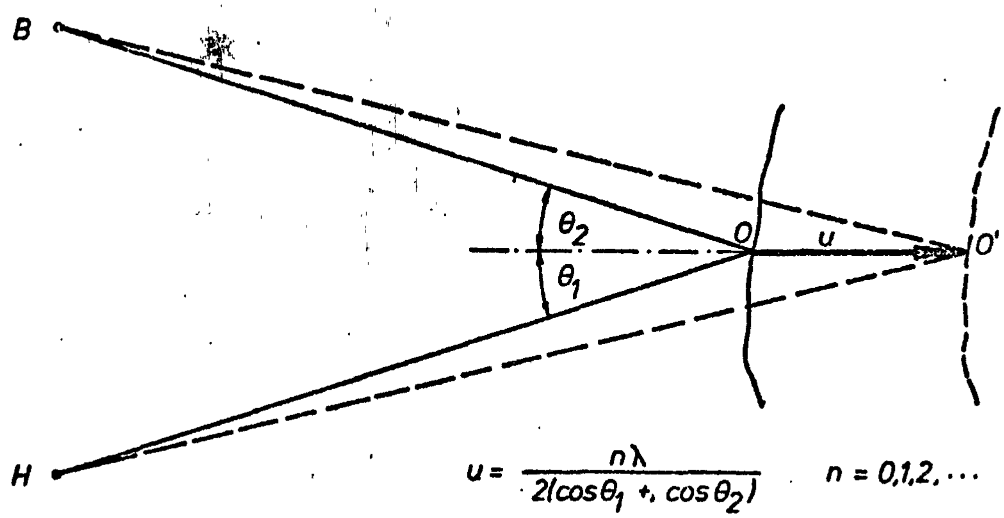

Fig. 1 Path length difference between the lens B and a point H of the Hologram-plate after moving the objectpoint O to O'

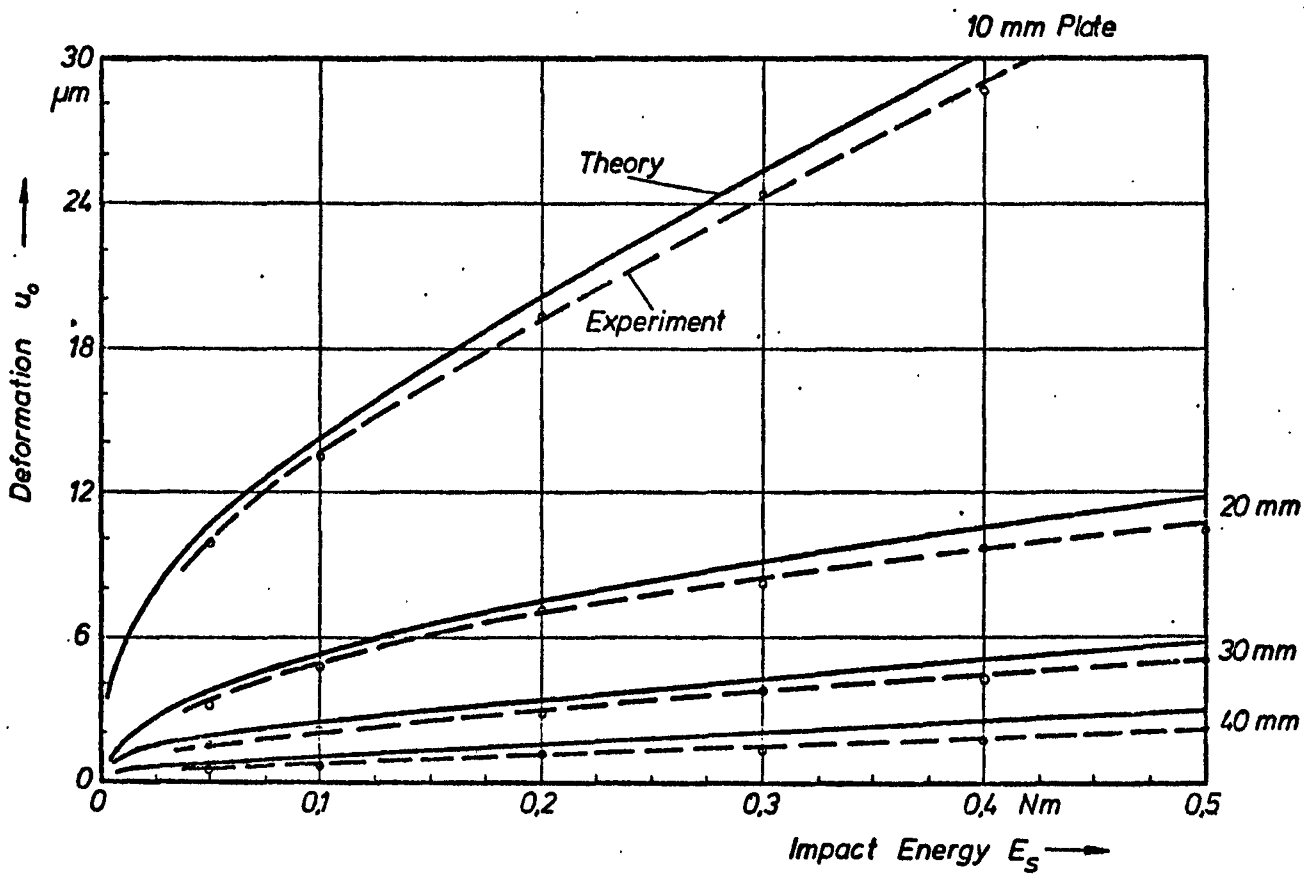

Fig. 2 Surface deformation versus Impact Energy for plates between 10 and 40 mm (ball: 20 mm ⌀)

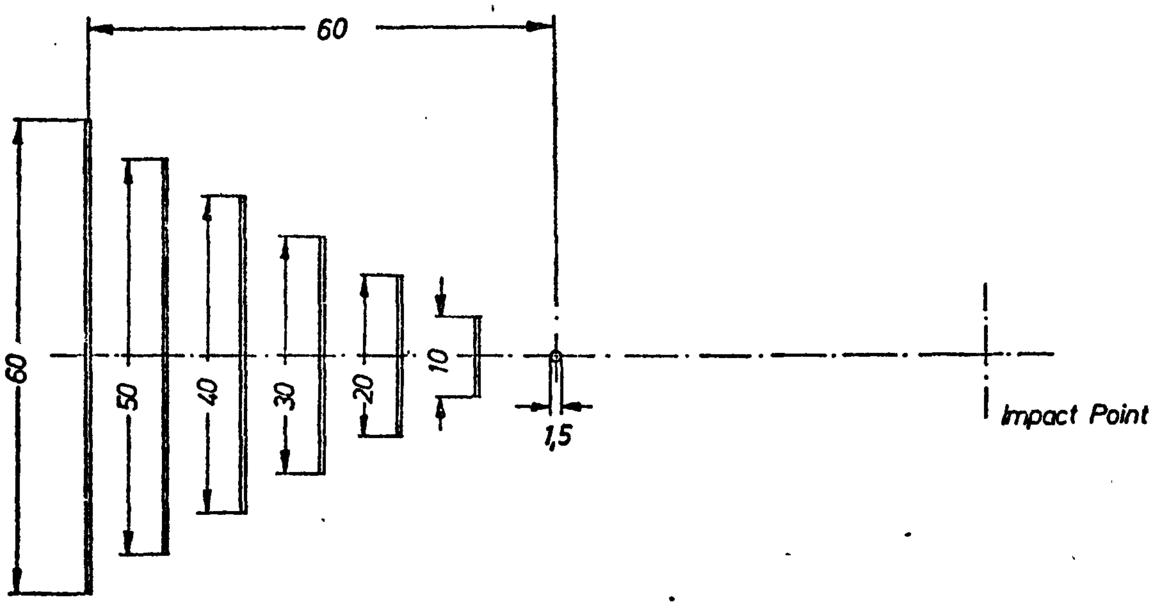

Fig. 3 Crack arrangement with cracks between 10 and 60 mm

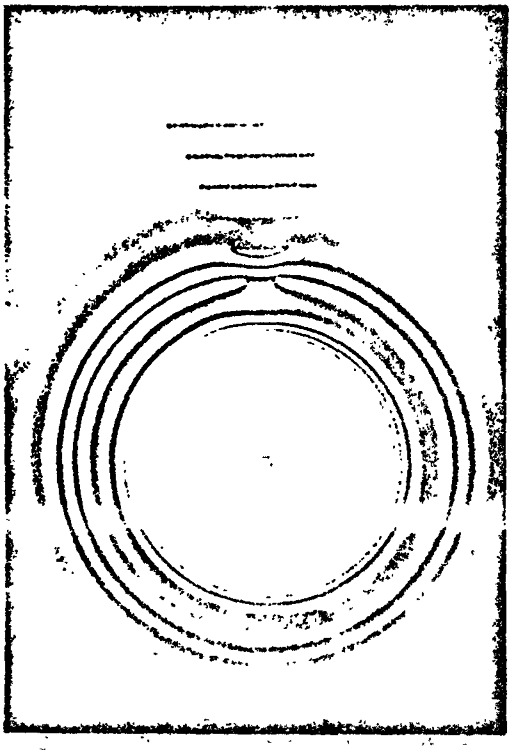

Fig. 4 Interference-Pattern of a 10 mm Steelplate

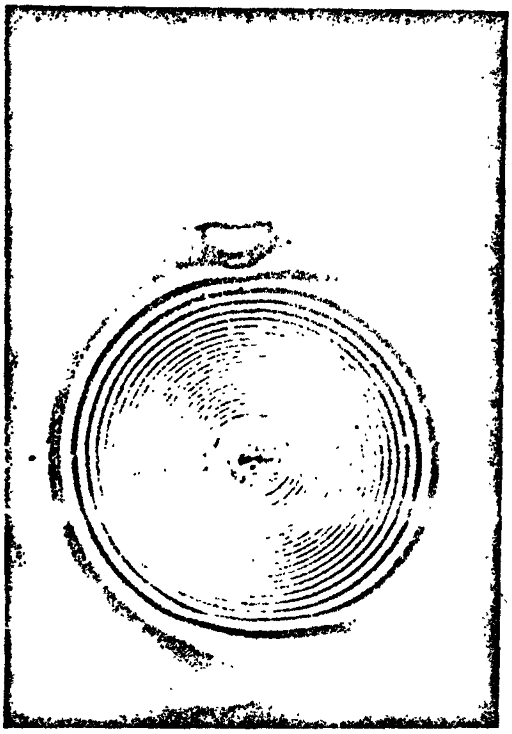

Fig. 5 20 mm Steelplate

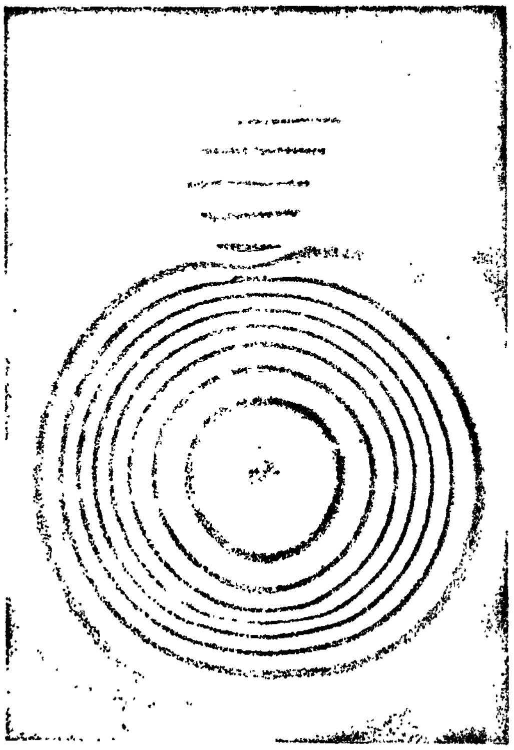

Fig. 6 30 mm Steelplate

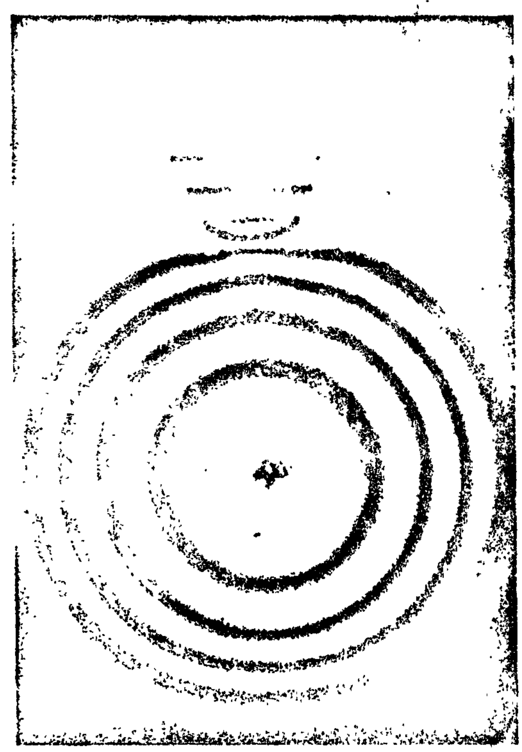

Fig. 7 40 mm Steelplate

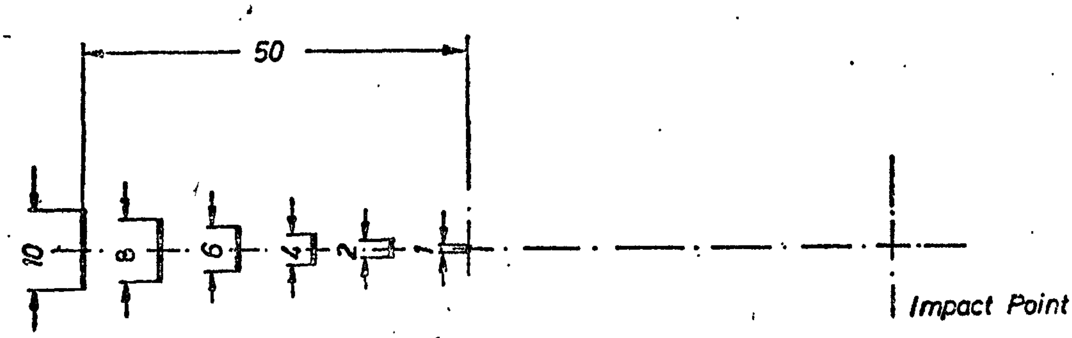

Fig. 8 Crack arrangement with cracks smaller than 10 mm

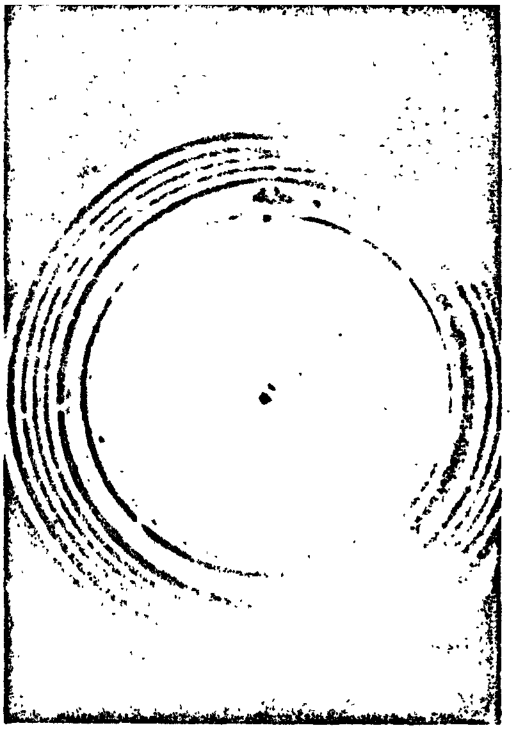

Fig. 9 Interference-Pattern of a 20 mm Steelplate with crack arrangement of Fig. 8

Classification 11.2.3

Title 1 Development of Acoustic Emission Measurement

Title 2

Country UK

Sponsor UKAEA

Organisation REML, Risley

Initiated 1969

Status Continuing

Completed

Last updating

Project leaders P. Bentley

1 General Aim

To determine from measurements made during a non-destructive pressure test, whether a vessel has a significant crack.

2. Particular Objective

To characterise emissions from cracks in many practical test conditions.

3. Experimental Facilities

The mobile laboratory is being used to locate signal sources during tests on vessels and small test pieces. Further development is necessary to permit identification of the types of acoustic signals due to the differing types of defect e.g., cracks, inclusions - slag or porosity, crack growth, localised yielding, brittle cracking and ductile tearing.

To permit the signals produced in different tests to be compared in a more quantitative manner, calibration devices are under development.

Reference Documents

Internal documents.

703

Classification 11.2.3

Title 1 Development of Holography	**Country** UK
Title 2	**Sponsor** UKAEA
	Organisation SRD, Culcheth
Initiated 1973 **Completed**	**Project leaders**
Status Continuing **Last updating**	D. Martin

1 General Aim

To determine from measurements made during a non-destructive pressure test whether a vessel has a significant crack.

2. Particular Objectives

To characterise holograms of typical vessel regions - membrane, junction, nozzle etc. - with and without cracks.

3. Experimental Facilities

The photographic equipment - including a pulsed laser - is being assembled. Tests are in hand on: a 5' vessel at Culchet; a 3" test plate in a 4000 T machine at Risley; a 2" test plate in a machine at Windscale; any thickness of test pla (currently 3") in the hydraulic pressuriser at Springfields. Part-through defects in a series of small cylinders are being hologrammed.

Reference Reports

Internal documents.

PROJECT TITLE : Assessment of nuclear pressure vessel integrity through acoustic emission monitoring.	CLASSIFICATION 11.2.3.
SPONSORING COUNTRY : Italy	ORGANISATION : CISE sponsored by ENEL
DATE INITIATED : October 1971 DATE COMPLETED : 1978	PROJECT LEADER : G. Possa and F. Tonolini

Status (latest updating): June 1975

Description :

1. General aim: to develop non-destructive techniques based on acoustic emission monitoring for the assessment of nuclear pressure vessel integrity.

2. Particular objectives: to develop acoustic emission monitoring systems and methods to be utilized 1) during pressure vessel hydrotests in cold unirradiated conditions, 2) during pressure vessel hydrotests in hot and possibly irradiated conditions, 3) during power operations as a continuous automatic pressure vessel surveillance.

3. Experimental facility and programme

3.1. Experimental facility
No applicable

3.2. Programme

3.2.1. Development of an acoustic emission instrumentation system capable of detection and precise location of emitting defects during pressure vessel cold hydrotests in unirradiated conditions.

3.2.2. Development of acustic emission sensors capable of withstanding the environment of a nuclear pressure vessel in operation (temperature; γ and neutron flux; remote handling).
Development of devices for long distance ($\sim$100 m) signal transmission from sensors to preamplifiers.
Development of instrumentation for automatic continuous auscultation of nulear pressure vessel acoustic emission during power operation.

4. Project status

4.1. Progress to date

- (3.2.1.): design and construction phases almost completed; the acoustic emission instrumentation system is now being tested in various pressure vessel hydrotests.
- (3.2.2.): activity not yet initiated.

4.2. Essential results

- Development of an entirely new non-destructive technique, which allows detection and location of all relevant pressure vessel propagation defects.
- Significant improvement of nuclear pressure vessel integrity assessment during power operation.

5. Next steps

- Completion of program item (3.2.1.), in particular with extensive tests of acoustic emission instrumentation in various steel pressure vessel hydrotests.
- Activity on program item (3.2.2.) will begin in 1976.

6. Reference documents (Main titles)

1) E. Fontana, G. Grugni, B. Pirovano, G. Possa, F. Tonolini: "Controllo non distruttivo di recipienti a pressione mediante analisi dell'emissione acustica nel corso della prova a pressione idrostatica", Energia Nucleare, Vol. 21 n° 10, ottobre 1974.

2) E. Fontana, G. Grugni, C. Panzani, B. Pirovano, G. Possa, F. Tonolini: "Acoustic Emission Monitoring During Hydrotests of a Thin Wall Pressure Vessel", Energia Nucleare, Vol. 22 n° 5, maggio 1975.

7. Cost estimate

The estimate cost of this R & D program is about 200 millions lire per year.

8 Means of cooperation

Various means of cooperation may be envisaged: for example, Italian partecipation in research programs carried out by partner countries, cost sharing, etc.

Classification 11.2.3.	
Title 1 Automatisch ultrasoon onderzoek van PWR-componenten	**Country** The Netherlands **Sponsor** PZEM **Organization** KEMA, MatEVal NDT Co
Title 2 Automatic ultrasonic examination of PWR-systems	**Projectleader** R. Huizing
Initiated 1975 **Status** Tests are being performed during shut-down February 1976	**Scientists** H. Jackson (MatEval) V.d. Berg) De Jong) KEMA Boer) Vroman)

1. General aim

Performing inservice ultrasonic inspection on PWR-systems in accordance with the APME-Code and Dutch authorities.

2. Particular objectives

Development of automatic positioning and scanning devices used to carry inspection equipment, such as ultrasonic wheels, to remote areas where manual acces is restricted for reliable inspection operations.
The data acquisition equipment is a special purpose mini-computer to provide optimum data; easy command of the data and mechanical systems; and convenient, easily interpreted, real-time display of all essential information.

3. Experimental facilities and programme

Pre-tests have been performed in KEMA- and MatEval (Warrington) laboratories.

The programme consists of:

- ultrasonic inspection of circumferential welds and nozzle welds in a pressurizer;
- ultrasonic inspection of circumferential welds and nozzle welds of a steamgenerator;
- ultrasonic inspection of circumferential and longitudinal welds in mainloops.

4. Project status

Tests are being performed during shut-down February 1976.

5. Next steps

The main devices to carry the inspection equipment will not be removed and will be used for the next inspection period.

6. Relations with other projects

Not applicable.

7. Reference documents

Not applicable.

8. Degree of availability

Through the organization KEMA.

Classification 11.2.4	
Title 1 Onderzoek naar het scheuruitbreidingsgedrag van onderplateringsscheurtjes in PWR's	**Country** The Netherlands **Sponsors** PZEM, GKN, TNO, KEMA **Organizations** KEMA, TNO
Title 2 Evaluation of the behaviour of undercladding cracks in a PWR	**Projectleader** R.M. van Kuijk
Initiated 1971 **Status** Completed 1975	**Scientists** A.P.A.M. Beyers L.B. Dufour H.C. van Elst

1. General aim

Crack extension behaviour of undercladding cracks in heavy section nuclear steel pressure vessel during service.

2. Particular objectives

Investigation of the crack extension in A508-02 under cyclic load conditions, load and temperature simultaneously. Reasurements of crack depths were performed by sampling and statistical approaches.

3. Experimental facilities and programme

The tests have been performed in TNO and KEMA laboratories. The programme consisted of:

- making a testpiece of A508-2 containing undercladding cracks;
- development of apparentus for realizing simultaneously cycling load and temperature;

- testing two testpieces with undercladding cracks under cyclic loading (1000 times) and constant temperature;
- testing two testpieces with undercladding cracks under cyclic loading (1000 times) and cyclic temperature from 70° - 300°C simultaneously;
- sampling the testpieces after the tests, measurements of cracky depths, statistical approaches.

4. Project status

Completed end 1975. It was shown that undercladding cracks grow very little in the base material and no indications were found that the cracks will grow through the cladding.

5. Next steps

None.

6. Relations with other projects

Through TNO with similar projects (TNO, Breda

7. Reference documents

A serie of about 30 technical and progress reports have been prepared. All these reports are written in Dutch.

8. Degree of availability

Through the organization KEMA.

PROJECT TITLE : Research Program on Concrete for Nuclear Reactor Vessels	CLASSIFICATION 11.3.1
SPONSORING COUNTRY : ITALY	ORGANISATION : ENEL
DATE INITIATED : 1970 DATE COMPLETED : -	PROJECT LEADER : P. Bertacchi

Description :

1. General Aim

Determination of concrete properties to provide data for prediction of vessel behaviour.

2. Particular Objectives

- Systematic investigation on the behaviour of concrete subjected to multiaxial stresses.
- Determination of concrete strength after wet and dry thermal treatments
- Study of concrete creep.

3. Experimental Facilities and Programme

All the tests are carried out both at ENEL - Niguarda Laboratory (Milan) and at ISMES Laboratory (Bergamo).
These investigations are carried out in the frame work of a joint research program with CEGB-CERL.

4. Project Status

- Systematic tests have been carried out on concrete specimens (cubical and cylindrical) subjected to bi- and triaxial stresses with an aim at determining the "rupture surface."
- Investigations have been carried out on the mechanics of onset and propagation of micro-cracks to gather information on the possibility of defining the limit of concrete elastic behaviour under multiaxial stress conditions.

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Sonic methods have mainly been used with equipment for measuring propagation velocity of costant-frequency signals.

- A large number of specimens have been subjected to thermal treatment at temperatures of 90° and 120° C, both in dry and in moist ambients; moreover, they have been subjected to thermal cycling and to continuous treatment.
  The effects of thermal treatment on the bending and compression strength and on Young's modulus (static and dynamic) have been determined

5. Next steps

a. Study of concrete creep.
   The installation of automatic load control equipment at the Niguarda Laboratory, with the possibility of subjecting concrete prismatic and cylindrical specimens up to a max load of 100 tons, makes it possible to start systematic research in order to determine the specific surfaces of creep and to study the influence exerted on concrete creep by various factors, such as the shape and size of the specimens subjected to loads which are kept constant with time.

b. Development of novel techniques (optical holography) for determining the state of strain and early stage of fissuring in concrete.

6. Relation to other Projects

Development of advanced solutions for pre-stressed concrete pressure vessels (thin-wall solutions) (11.3.2 - 11.3.4)

7. Reference Documents

- R. Bellotti ,"Joint Research Programme of ENEL-DSR/CEGB-CERL on concrete for Nuclear Reactor Vessels - Enel Contribution. State of the research as at December 1973." CRIS n. 2425.
- R. Bellotti a P. Rossi ," New Prospects for Evaluating the Degree of Safety in Concrete Structures Subjected to Multiaxial Stresses."
  III - 5; Seminar on: Concrete Structures Subjected to Triaxial Stresses - 17th - 19 th May, 1974 - ISMES - Bergamo.

	Classification 11.3.2/11.3.4.
Title 1 Brudundersøgelse af Betontank-lågmodeller.	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2 Overload behaviour and ultimate load capacity of PCRV-closures for BWR-plants.	Project leader S.I. Andersen Scientists:
Initiated (date): July 1971 Completed: (date) 1976 Status: progressing Last updating (date): February 1976	N.S. Ottosen S.I. Andersen

1. General aim

To study the overload behaviour, failure mode and ultimate load capacity of PCRV-closures for a Nordic BWR/PCRV reference design.

2. Particular objectives

To investigate the influence from different design parameters, such as depth-to-span ratio, reinforcement and supporting flange geometry, etc.

3. Experimental facilities and programme

The test facility, situated at Risø, includes a steel pressure vessel, in which model specimens in scale 1:11 of the reference vessel closure can be pressurized to a max. hydraulic pressure of 450 bars.

The programme includes testing of 9 different closure models and a series of comparative calculations by means of a finite element computer programme P-479, which has been developed by Risø for the analysis of axisymmetric PCRV-structures, taking into account the effects of concrete creep, plasticity and cracking together with the plastic behaviour of the steel parts.

4. Project status

9 closure models have been tested during the period between February 1972 and January 1976. Extensive calculations on the unreinforced specimens show good agreement with the experimental observations.

In general the computer programme has been verified thoroughly by the comprehensive test results.

5. Next steps

Final reports are under preparation.

6. Relations with other projects

The programme is part of a joint Nordic development work on a PCRV for BWR application.

7. Reference documents

[1] Ultimate load behaviour of PCRV top closures.
S.I. Andersen, N.S. Ottosen
Paper H 4/3,2. Int. Conf. on Struct. Mech. in Reactor Technology
Berlin (1973)

[2] Theoretical and experimental studies for optimization of PCRV top closures.
N.S. Ottosen, S.I. Andersen.
Paper H 3/6. Trans. of the 3rd Int. Conf. on Struct. Mech. in Reactor Technology, London Sept. 1975.

8. Degree of availability

Project information restricted.

PROJECT TITLE : Development of Advanced Solutions for Pre-Stressed Concrete Pressure Vessels (Thin-Wall Solutions)	CLASSIFICATION 11.3.2 11.3.4
SPONSORING COUNTRY : ITALY	ORGANISATION : ENEL, ISMES
DATE INITIATED : 1970 DATE COMPLETED : -	PROJECT LEADER : F. Scotto

Description :

1. **General Aim**
To develop advanced economical and safe solutions for PCPVs by better exploiting the resistance of triaxially stressed concrete.

2. **Particular Objectives**
Reduction of the wall thickness of current PCPVs by applying novel design philosophies and tendon layouts.

3. **Experimental Facilities and Programme**
All the tests are carried out at the ISMES Laboratory at Bergamo.
The proposed solutions are investigated by means of suitable design tools, such as physical and mathematical models of the structure and rheological models of the material (concrete).
Sophisticated small-scale physical models (scale 1:10, 1:20) have been developed, built and tested to cover the entire range of structural behaviour up to collapse.

4. **Project Status**
   a. Progress to date
   The following models have been built and tested up to collapse:
   - One thick- and three thin-wall models (scale 1:20) for a 300-MWe HTR
   - One thin-wall "continuous" model (scale 1:10) for a 1000-MWe BWR.

   Use has been made of mathematical models for the selection of the optimal geometrical solutions by means of an interactive type of man-machine analysis.
   A computer program was developed to take into account also the frictional interaction between the lid and the main cylindrical structure.

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b. Essential Results

The proposed thin-wall solution can lead to a reduction in the PCPV weight from about 37,000 tons to 14,000 tons for a 300-MWe HTR, and from 8000 to 4000 tons for a 1000-MWe BWR

5. Next Steps

a. Development of a new concept of pre-stressed concrete lid for thin-walled PCPVs, and experimental and theoretical verification.

b. Recycling of the results of the physical model tests to the mathematical model.

c. Testing of the final solution of the thin-walled PCPV by means of one or more physical models (scale 1:10).

Relation to Other Projects

a. "Research Program on concrete for nuclear reactor vessels" (11.3.1)

b. ENEL-A.B. Atomenergi Agreement concerning "Collaboration for the Development of PCPVs for LWRs".

7. Reference Documents

- F. Scotto, "Triaxial State of Stress of 'Thin-Walled' PCPVs for HTGRs; Comparison with a Conventional 'Thick' Solution ". IABSE Seminar on Concrete Structures Subjected to Triaxial Stresses, Bergamo, 17-19 May, 1974.
- M. Fanelli et al., "Finite-Element Analysis of Prestressed - Concrete Pressure Vessels ". IABSE Seminar, 17-19 May 1974.
- E. Fumagalli and G. Verdelli, "Small-Scale Models of PCPVs for High-Temperature Gas Reactors. Modelling Criteria and Typical Results". IABSE Seminar, 17-19 May 1974.

CLASSIFICATION
11.4.1

TITLE 1 ENCEINTE DE CONFINEMENT POUR PWR. DOUBLE ENCEINTE SANS PEAU D'ETANCHEITE.	COUNTRY FRANCE SPONSOR E.D.F. ORGANIZATION E.D.F.
TITLE 2 PRIMARY CONTAINMENT FOR PWR WITHOUT METALLIC SKIN FOR TIGHENESS.	Project Leader E.D.F./SEPTEN/M Scientists
Initiated 1970 Completed 1975	M. COSTAZ
Status Last updating : 20.01.75	

I - GENERAL AIM . Définition d'un nouveau type d'enceinte de confinement plus économique, sans peau d'étanchéité. Enceinte double en béton.

II - PARTICULAR OBJECTIVES

La conception et la réalisation des peaux d'étanchéité métalliques d'enceintes en béton armé ou précontraint présentent de nombreuses difficultés provenant de la nécessité de limiter l'épaisseur des tôles à des valeurs faibles (6 ou 9 mm) pour des raisons économiques.

Un nouveau système de confinement a été mis à l'étude, il est composé de deux parties :

- une enceinte interne en béton précontraint, sans peau d'étanchéité métallique
- une enceinte externe en béton armé.

En cas d'accident, les fuites traversant l'enceinte interne, sont récupérées dans l'espace annulaire et filtrées avant rejet.

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III - EXPERIMENTAL FACILITIES AND PROGRAMME

Les études et moyens d'essais associés sont :

- étude de la structure sous l'angle du Génie Civil,
- essais d'étanchéité du béton,
- essais d'étanchéité des peintures,
- essais de contrôle d'étanchéité du béton en cours de construction.

IV - PROJECT STATUS

4.1 - Progress to date

- étude d'un avant projet détaillé d'enceinte (1973)
- essais d'étanchéité du béton armé ou précontraint (1973).

4.2 - Essential Results

Faisabilité démontrée.

Choix par E.D.F., en accord avec les Autorités de Sûreté Française, de ce type d'enceinte pour équiper les tranches nucléaires à venir W 1350.

V - NEXT STEPS

- études d'avant projet détaillées de l'enceinte W 1350,
- essais d'étanchéité de revêtements élastomères sur béton,
- essais concernant le contrôle de l'étanchéité du béton en cours de construction.

VI - RELATION WITH OTHER PROJECTS

Néant.

...

VII - REFERENCE DOCUMENTS

Note E.D.F. SEPTEN GC 72-30. Double enceinte sans peau. Spécification technique.

- note E.D.F. SEPTEN GC 73-01. Double enceinte sans peau. Synthèse des études et essais réalisés en 1972,

- note E.D.F. SEPTEN GC 73-09. Présentation et analyse des essais réalisés.

VIII - DEGREE OF AVAILASILITY

Internes E.D.F.

Classification: 11.5	
Title 1 (Original Language): Untersuchungen zum Einfluß der Größe und Form von Kühlkanalblockagen auf die Kernnotkühlung in der Flutphase eines Kühlmittelverlustunfalles (PNS 4239 - I.1.3., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Influence of the Size and Shape of Cooling Channel Blockages upon Emergency Core Cooling in the Reflood Phase of a LOCA	Project Leader: S. Malang
Initiated (Date): 1.1.1973 Status: continuing	Completed (Date): 1979 Last Updating (Date): December 1975

1. General Aim

The aim of this project is to investigate the influence of cooling channel blockages on the cooling conditions during the reflood phase of a LOCA. Such blockages can be caused by ballooning of the fuel rod claddings due to the internal pressure. Prefabricated "balloons" are attached to the fuel rod simulators prior to the tests.

2. Particular Objectives

The main objective is to investigate the influence of size and shape of the flow blockages on the cooling conditions. Velocity and spectrum of the water droplets in the mist flow have to be determined with optical methods.

3.1 Experimental Facilities

The main facility is a test loop suitable for up to 25 rods with a heated length of 3.90 m. This loop is designed for a constant flooding rate and a constant back pressure. A SCR controlled power supply is used. The digital data system with 128 channels is controlled by a central computer. Power supply and data system are used also for PNS 4238. A separate test rig is used for fuel rod simulator test and some scoping tests.

3.2 Research Program

The following configurations are to be investigated:

a) row of 5 heaters without blockages
b) row of 5 heaters with uniform blockages
c) 5x5 bundle with uniform blockages
d) 5x5 bundle with blockages only in a part of the coolant channels

The main parameters are:

a) flooding rate
b) initial clad temperature
c) back pressure
d) size of the balloons
e) shape of the balloons

The largest number of test are planned for PWR's. Some experiments will be performed later for BWR's.

4. Project Status

The construction of the test loop, the data system and the power supply have been completed. Also completed are the test section and 5 instrumented fuel rod simulators. The shake down of the entire system has been started. The heater rods had been successfully tested under operating conditions in a separate rig. Less expensive and more precise methods for the instrumentation of the heater rods and the attachment of the"balloons" are under development. Some methods to avoid a too early wetting of the observation windows at the test section have been tested and showed some improvements.
Films have been taken through these windows with a frequency of 4000 picture/sec.
The exposure time was still to long to obtain sharp pictures of the water droplets.

5. Next Steps

After an extended shake down period the experiments will start with a row of 5 fuel rod simulators. Tests with a 5x5 bundle will follow at the end of 1976.
A computer code for the calculation of heat transfer coefficients will be adjusted to the test conditions and will be used for the data reduction.

6. Relation With Other Projects

There is a close connection to a simular project using a larger bundle which is planned by KWU (RS 36). One of the goals is to define size shape and mechanical design of the flow blockages for that project. The results will also be used for the verification of the code system which is under development in the project PNS 4231.
Relations exist also to the project PNS 4238.
In that project however the main emphasis is placed on the interaction between deformation and cooling.

7. Reference Documents

/1/ 1st PNS-Semi-Annual Report 1975, KFK 2195 (German, Engl.abstr.)
/2/ 2nd PNS-Semi-Annual Report 1975, KFK 2262 (German, Engl.abstr.)
/3/ Heat Flux Density on the Surface of a Heater as a Consequence of an Artifical Blockage
G. Hofmann, H.M. Politzky, K. Rust
PNS - Arbeitsbericht Nr. 54/75, Sept. 75

8. Degree of availability

- KFK-reports : unrestricted distribution
  PNS-Arbeitsberichte: restricted destribution

PROJECT TITLE : CIRENE power channel pressure tube burst tests. Effetti dello scoppio di tubo a pressione CIRENE.	CLASSIFICATION 11.5
SPONSORING COUNTRY : Italy	ORGANISATION : CISE sponsored by CNEN
DATE INITIATED : October 1970 DATE COMPLETED : 1976	PROJECT LEADER : G. Possa

Status (latest updating): May 1975

Description :

1. General aim: to investigate the consequences of the hypothetical explosion of a power channel pressure tube for the CIRENE reactor.

2. Particular objective: to measure the pressure waves produced by explosion in the D_2O tank and to determine the associated stresses on major mechanical structural components.

3. Experimental facility and programme

3.1. Experimental facility

- BETULLA: facility located in CCR Euratom at Ispra including a pressure vessel (with nearly CIRENE dimensions) for explosion containment.

3.2. Programme

3.2.1. Preliminary burst tests to individuate most relevant plant parameters.

3.2.2. Burst tests in full height scale with a dummy explosion tube (full length rupture).

3.2.3. Burst tests as in 3.2.2. with explosion tube and adjacent target channel simulating in full scale (dimensions and structural materials) the CIRENE power channel.

3.2.4. Burst tests for determination of consequences of a hypothetical pressure tube rupture due to hot spot.

4. Project status

4.1. Progress to date

- (3.2.1.): completed
- (3.2.2.): nearly completed
- (3.2.3.): design and component manufacture completed
- (3.2.4.): in the design stage

4.2. Essential results

- Basic understanding of the dynamics of underwater explosions in a confined volume.
- Understanding of the rupture process of a pressurized Zircaloy tube artificially defected, under typical conditions of a pressure tube reactor.
- Gathering information on explosion stresses in reactor structural components.

5. Next steps

Completion of program item 3.2.2.; then program item 3.2.3.; then program item 3.2.4.

6. Reference documents

1) M. Famiglietti, A. Parmeggiani, G. Possa, L. Galbiati: "Pressure Burst Due to Power Channel Explosion in a Pressure Tube Reactor", to be presented at the 3rd SMIRT Conference, London 1-5 Sept. 1975.

7. Degree of availability: to a limited extent.

12. QUALITY ASSURANCE

Classification: 12.1	
Title 1 (Original Language): Statusbericht Qualitätssicherungssystem - Darstellung des Istzustandes - (RS 124 - II.3.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Status Report of the Quality Assurance System	Project Leader: Dr. Kaden
Initiated (Date): 8. 75 Status: Continuing Completed (Date): 31. 6. 76 Last Updating (Date): 31. 12. 75	

General Aim

In a study the present state of the quality assurance system for design, manufacturing and examination, procurement, shiping erection and commissioning of the components (NDES) including nuclear auxiliary and ancillary components and the containment will be described.

Particular Objectives

components are considered, which are used in the nuclear steam generating system.

Experimental Facilities

No experimental facilities necessary.

Research Program

a) Listing of various steps of the work for the representation of the present state.

b) Detailed description of the activities, the boundary conditions and the quality assurance securing.

Project Status/Progress to Date/Essential Results

A first draft of a status-report was finished and discussed with members of the IRS. The result was, that some new items have to be considered and some details have to be explained.

Next Step

The status-report will be revised.

Relation with Other Projects

No relation with other Projects

Reference Documents/Degree of Availability

No reports available.

PROJECT TITLE : RELIABILITY STUDIES	CLASSIFICATION 12.1
SPONSORING COUNTRY : ITALY	ORGANISATION : CNEN
DATE INITIATED : October 1974 DATE COMPLETED : In progress	PROJECT LEADER : G. TOMASSETTI

Description : A research has been initiated in the field of reliability studies focused on components (electronic and mechanical), of water reactors.

Particular attention will be given to failure mode analysis of mechanical components of experimental loops and to the methods for collecting failure data with the aim to evaluate the possibility of using, in the most suitable way, data banks.

Classification: 12.3	
Title 1 (Original Language): Untersuchungen über die Anwendbarkeit der Ultraschall-Impulsspektrometrie zur Verbesserung der Aussagesicherheit bei der Materialprüfung mit Ultraschall (RS 54 - II.3.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Bundesanstalt f. Materialprüfung
Title 2 (english): Ultrasonic Pulse-Echo Spectroscopy in Ultrasonic NDT	Project Leader: Prof. Dr. Mundry
Initiated (Date): February 10, 1972 **Status:** finished	**Completed (Date):** February 10, 1975 **Last Updating (Date):** December 1975

1. General aim

The project aims at the complete evaluation of the information furnished by ultrasonic echoes.

2. Particular objectives

By means of frequency analysis type, size and orientation of a defect in materials of larger thicknesses (reactor pressure vessels) shall be determined.

3. Experimental facilities and program

Spectrum analysis can be performed with two different devices. The first device is a swept frequency RF-receiver, which is plugged into an oscilloscope mainframe. The other device is a digital Fourier Analyser, which calculates real and imaginary part (or amplitude and phase) of a spectrum of a given pulse. Additionally a set of broad band (shock wave) probes, ultrasonic transmitter/receivers, a set of steel specimen, an immersion tank and an electronic switch for time windowing are available.

During the experiments the echoes of artificial defects of different type, size and orientation are to be analysed, to study the influence of these parameters on the frequency spectra.

4. Project status

4.1. Progress to date

The project has been finished. During the last two months of its duration investigations on the influence of the angle of incidence on amplitude and phase spectra took place. The experiments were carried out in immersion technique so, that the test reflectors were turned to vary the angle of incidence from normal incidence in steps of 0.2°. The distance between transducer and reflector was held constant.

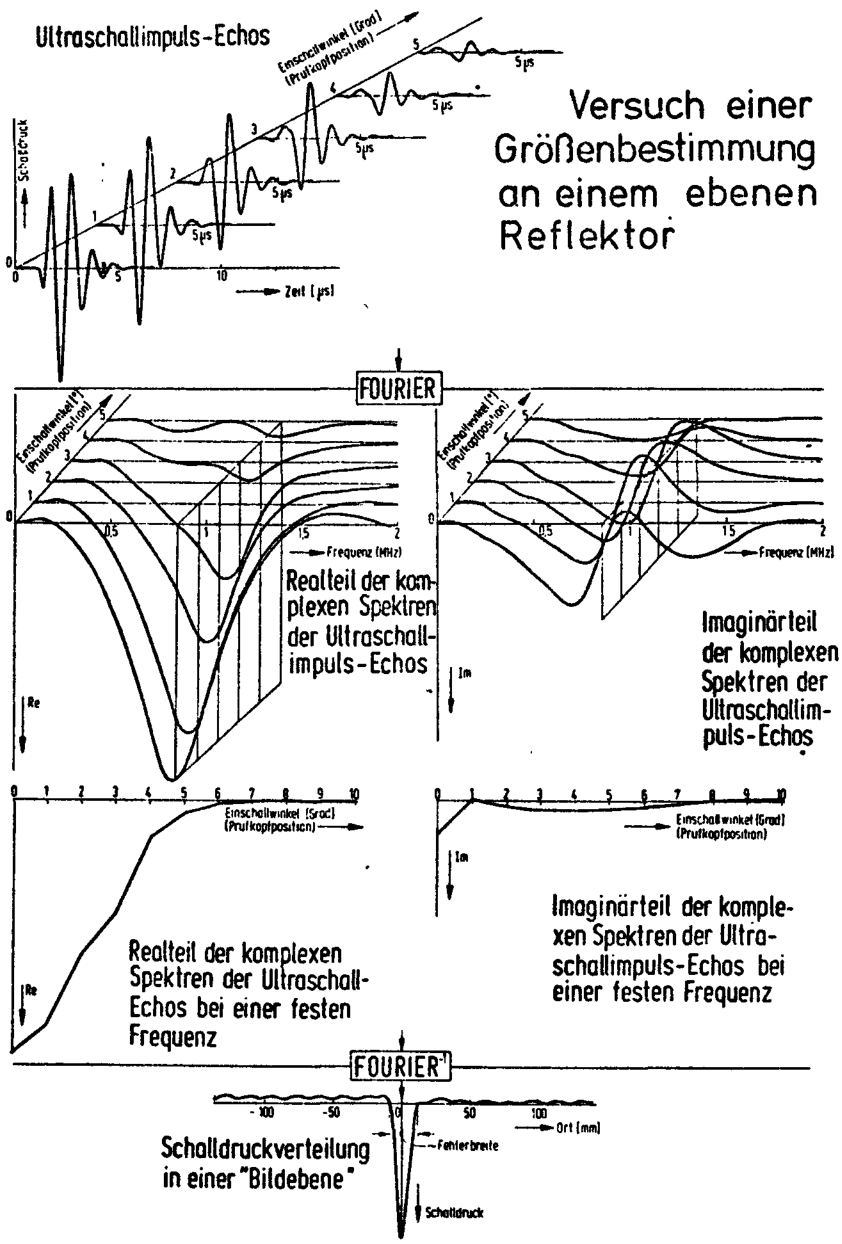
4.2. Essential results

A result of the measurements is shown in the picture. The angle of incidence for a 20 mm circular flat reflector was varied in the range of 0° to 10° in steps of 0.2 degrees. The upper part of the picture shows the time function of six of the recorded echoes in the range of 0° to 5° in steps of 1°.

The second line of drawings shows real and imaginary part of the spectra of the above pulses after Fourier transform.

Next the mountains of spectra are cut through along the drawn plane, which is normal to the frequency axis. In these two planes one can find the curves of the third line of drawings. The left curve gives the real part and the right curve gives the imaginary part of an amplitude distribution versus the angle of incidence. In a rough analogy to holography a reconstruction of the reflector may be obtained from that distribution by inverse Fourier transform. The width of the peak of the last curve is strictly proportional to the width of the reflector.

Versuch einer Größenbestimmung an einem ebenen Reflektor



Classification: 12.3	
Title 1 (Original Language): Studie über den derzeitgen Stand der Schallemissionsanalyse (SEA), ihre Grenzen und Möglichkeiten auf dem Gebiet der Reaktorsicherheit (RS 0031 D - II.3.2., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** Battelle-Institut Frankfurt/Main
Title 2 (english): Literature Survey of the State of the Art of Acoustic Emission, its Potentiality and its Limits in the Field of Reactor Safety	**Project Leader:** Dr. Eisenblätter Dr. Jax
Initiated (Date): July 1, 1975 **Status:** finished	**Completed (Date):** November 30, 1975 **Last Updating (Date):** December 1975

1. General Aim

The aim of these evaluations is to present critically the status of knowledge and technology of acoustic emission for testing of thick-walled components, especially those of nuclear reactor vessels.

2. Particular Objectives

This literature survey shall be of use

- to present completely the results of the work done so far at Battelle Frankfurt during the past six years
- to compare this knowledge with the status of research in the Federal Republic of Germany and other countries
- to mark out the problems still existing.

3. Research Program

The results of our own experimental work which are represented in more than 20 technical reports (reference RS 0031 to RS 0031 D) have been taken into consideration as well as the relevant literature published lately.

4. Project Status

The investigations are terminated. The corresponding report will be completed in February 1976.

4.2. Essential Results

First in literature the knowledge in seismology concerning the source mechanisms of earth-quakes and the propagation of earth-quake waves have been applicated on acoustic emission. Thus it was possible to understand the most important fundamentals of acoustic emission originated by different processes (crack formation, crack propagation, plastic deformation, friction) and to present them in a consistent form. Besides presenting these fundamentals, the scope of this survey was to show the still existing problems, especially in inspection of thick-walled vessels. Nowadays these are mainly concentrated in the classification of located defects. Related investigations conducted until today mostly do not furnish valuable information about this problem because of the following reasons:

- structures with unknown defects did not contain critical defects; those defects which have been found are not large enough to be identified unambigiously by other NDT methods; destructing methods are not possible,
- in the case of specimens with intentionally built-in defects mostly the natural defects occuring in structures have not been simulated correctly so far.

Therefore for the future special attention should be drawn on fundamental evaluations of thick-walled vessels with built-in natural defects.

5. Next Steps

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6. Relation with Other Projects

This literature survey represents the completion of the following projects: RS 0031, RS 0031 A, RS 0031 B, RS 0031 C.

7. Reference Documents

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Classification: 12.3	
Title 1 (Original Language): Zerstörungsfreie Prüfung des Gefügezustandes mittels Ultraschallrückstreuung (RS 102-16/1 - II.3.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Gesellschaft, IzfP
Title 2 (english): Non-destructive Structure Evaluation by Means of Scattered Ultrasound	Project Leader: Dr. G. Deuster Dr. K. Goebbels
Initiated (Date): .1973 — Completed (Date): 30.9.1976 Status: continuing — Last Updating (Date): December 1975	

1. General aim

Among other things the material constants of steels are structure-dependent. Therefore the determination of the structure of steels is an essential task and has to cover the whole volume, especially for big components like reactor pressure vessels. Under this category the structure of welds is to regard too and the evaluation has to be done in a ndt way. NDt-methods for these requirements are not yet technical standard, but in an advanced stage of development.

2. Particular objective

Measurements of scattered ultrasound can be used for qualitative and quantitative determination of metal structures. Compared to attenuation measurements, they present e.g. the following advantages:

- no need of plane-parallel, polished test specimens,
- evaluation of the structure inside the material,
- quantitative separation of absorption and scattering coefficient from attenuation coefficient,
- quantitative grain-size determination, independent of other methods.

The aim of this project therefore ist the quantitative evaluation of homogeneous and inhomogeneous (e.g. welds) steel structures.

3. Experimental facilities and program

3.1 Experimental facilities

- For very accurate measurements of ultrasound velocities and attenuation an experimental device (MATEC) is used.
- Scattering measurements are made with two apparatus:
  - At first a device built by Koppelmann[1], recording the analog scattering signal on a XY-recorder.
  - A second apparatus was built in the institute, marked especially by an analog-digital-converter (100 MHz sample rate) and the on-line-evaluation of the scattering measurement in a computer.

3.2 Program

After the first steps, done in 1974 the following program was pursued:

a) Scattering measurements on different steels with homogeneous structure, with different frequencies.

b) Measurements on welds, expecially on materials corresponding to the reactor technology.

c) Tests of the new device for scattering measurements.

d) Optimizing the judgement of weld-quality with ultrasonic scattering measurements.

e) Measurements on austenitic materials and welds for better understanding of the steel structures and for obtaining a better signal-to-noise level in ultrasound examination.

4. Project status

4.1 Progress to date

Parts a) - c) of the program mentioned under 3.2 are mainly completed.
Parts d) and e) are of special interest in 1976.

4.2 Essential Results

- Three methods are developed to determine quantitatively the scattering coefficient and thereby the mean grain size: One sample and two frequencies. If the frequency-dependence of the ultrasonic absorption is known, the separation of absorption and scattering is possible after two scattering measurements.

  One frequency and two samples. If the absorption-coefficient is equal in both samples the separation necessary for grain-size determination is possible after the scattering measurement at each test piece.
  These two methods are described further in a recent publication[2].

Multiple scattering. If multiple scattering is present, already one measurement at one sample and without suppositions on absorption delivers the quantitative value for the scattering coefficent[3].

These methods are positively tested at half a hundred samples of steels with different structures and will be tested further for optimizing the scattering measurements at about two hundred samples.

- The judgement of the structure of welds with ultrasonic scattering is difficult because of the inhomogeneous character of the composition unaffected material - heat affected zone - weld.

Today the scattering measurements are made with sound paths parallel to the weld direction either in the unaffected steel or in the heat affected zone or in the weld itself. The attenuation coefficient in each region is determined from the scattering measurement and related to its position relatively to the weld. Therefore one measures the rise of attenuation from unaffected steel to weld by a factor of two to four e.g. and this rise is characteristic for the weld. Some experiences with 22 NiMoCr 3 7 - welds and the simulation of these welds are made and described in the quarterly reports.

5. Next steps

The next steps are the parts d) and c) of the program listed under 3.2. The application test of the grain-size detemination, mentioned under 4.2 will be executed too.

Relations with other projects

No essential relations with other projects are present. The work conceming austenitic steels in RS 102 - 16/1 and RS 143 are tuned together.

7. Reference documents

1) J. Koppelmann: Materialprüfung 14 (72) 156 - 159

2) K. Goebbels: Materialprüfung 17 (75) 231 - 233

3) K. Goebbels: Ultraschall-Streuung. Möglichkeiten der zerstörungsfreien Gefügebewertung. Grundlagenbericht. IzfP-Bericht Nr. 740311-TW, 1974

Classification: 12.3	
Title 1 (Original Language): Anpassung und Weiterentwicklung elektrischer, elektromagnetischer und magnetischer Prüfverfahren für den Einsatz an Reaktoren (RS 102-18 - II.3.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Gesellschaft, IzfP
Title 2 (english): Adaption and Continuation of the Development of the Electrical, Electromagnetic and Magnetic Test Methods for the Use at Reactors	Project Leader: R. Becker, Dr. W. Mohr
Initiated (Date): 1.8.1973 — Completed (Date): 30.9.1976	
Status: continuing — Last Updating (Date): December 1975	

General aim

In the field of the electrical, magnetic and electromagnetic test methods the project has the general aim.

- to transfer the status of knowledge into the concepts of techniques and devices,
- to expand the foundations in mathematics and physics as much as necessary,
- establish rules and models for interpretation of measurement results,
- to adjust the techniques to the specific problems in reactor testing.

Based on the research results, special developments orientated towards application are to follow as individual projects.

1. Eddy Current Test Method

1.1 Particular objective

We are working at two kinds of application:

a) testing of installed heat exchanger tubes,

b) testing of the cladding, the basic material, the weld and austenitic/ferritic junctions of the reactor pressure vessel for location of cracks near the surface.

In both cases a great number of parameters give measuring effects. The signals of defects which generally are the relevant parameters can be superimposed by signals of other parameters, so that the presence of the defects cannot be recognized or the indication of the defect depth can be falsified. Therefore the signals from the disturbing parameters must be suppressed and only the signals from the defects must be given out with the indication of the defect depth.

To solve this task the multifrequency method is to apply where is worked with several frequencies simultaneously. This method is to develop further with regard to the practical application and to test in several test situations which are simulated by models.

1.2 Experimental facilities and program

a) application of computer programs to determine the optimal test frequencies considering the separation of the relevant parameters and to provide ideal coil design;

b) construction of a laboratory test device which works simultaneously with up to 4 frequencies in the range of 10 - 500 kHz;

c) application of this device concerning researchs with multifrequency methods; synthesis of processes, otpmation of numerical evaluation methods using a mini-computer;

d) trial of the test system.

1.3 Project status

1.3.1 Progress todate

The laboratory test device as mentioned in 1.2.b has been completed. As model for an austenitic heat exchanger tube we used a tube with 3 saw cuts of different depths, variation of inner and outer diameter, change of electrical conductivity, an austenitic spacer and a ferritic tube sheet.

The multifrequency device has also been applied for testing various weld-claddings of the reactor pressure vessel. We have examined a multitude of specimen with saw cuts and pulsed cracks.

mathematical and physical problems of the encircling coil around a tube and of the pick-up coil above a two-layer specimen have been treated, and we have written computer programs for numerical evaluation as mentioned in 1.2.a.

1.3.2 Essential results

The testing of installed heat exchanger tubes is carried out from the inside of the tubes with an absolute running through coil. It is possible to indicate defects at the outside of the tubes greater than 35 per cent of the tube wall thickness and to suppress simultaneously the signals from variations of the geometrical and electrical properties of the tubes, the signals from the tube sheet and the spacers and the signals from variations of the position of the coil in the tube /1/.
The testing of the austenitic plated surface in a reactor pressure vessel is made with an absolute pick-up coil which is moved over the surface. In this case undesired signals which can be suppressed are caused by local variations of the δ -ferrit content and of the electrical conductivity of the two layers, and above all by the lift-off effect as a consequence of the not plain surface /2/.

To provide the multifrequency system with appropriate frequencies we have calculated the effects of variation in tube material for any frequency /3/.

The same work has been done for pick-up coils above a two-layer specimen. The influence of the coil dimensions on the direction of the effects in the impedance plane has been examined. Furthermore the relation between choice of frequency and eddy current distribution in the two layers has been treated for different coil dimensions. Plotted results are summarized in the intermediate report /4/.

1.4 Next steps

The multifrequency device will be prepared for the 'in-situ' employment to test a heat exchanger and the pressure vessel of a reactor.

Numerical results will be applied for the examination of plated materials with subcladding cracks.

1.5 Relations with other projects

Concerning the testing of the cladding specimen there exists a relation to project RS 89.

1.6 Reference Documents

/1/ Becker, R.:
Parametertrennung bei der Wirbelstromprüfung von Rohren nach einem Mehrfrequenzverfahren. IzfP-Bericht 740205-U zu RS 102 - 18, Teil I

/2/ Becker, R.:
Prüfung von plattierten Oberflächen mit Abtastspulen nach einem Mehrfrequenz-Wirbelstromverfahren. IzfP-Bericht 760602-TW

/3/ Becker, R.; Betzold K.:
Theoretische und numerische Untersuchungen über Impedanzortskurven bei der Wirbelstromprüfung von Rohren mit Außenspulen. IzfP-Bericht 750413-TW

/4/ Becker, R.; Betzold, K.:
Theoretische und numerische Untersuchungen zur Wirbelstromprüfung geschichteter Materialien mit Abtastspulen gegebener Abmessungen. IzfP-Bericht 760501-TW

2. Electrical Resistance Probe Method

2.1 Particular objective

The method is used for crack-depth measurements of surface cracks in metallic materials. By contacting the body with potential probes an electrical flow field is impressed. The associated potential field, measured on the surface is a function

- of the electrical conductivity
- of the distance of the potential probes and
- of the surface geometry.

Every surface crack disturbs this geometry and therefore the potential distribution. The normalized voltage U/U_0 measured above the crack is only a function of the depth of the crack while U_0 is the voltage in the undisturbed case. The theoretical formulation of the problem leads to a Neumann boundary value problem.

2.2 Experimental facilities and program

a) analytical solution of the Neumann boundary value problem in the case of homogenious excitation by a direct current.

b) application and test of numerical methods as the finite differences and finite elements methods for solving the boundary value problem in the case of partial flow excited by the point sources of the potential probes.

c) Treatment of the alternating current resistance method and the phenomenon of the skin effect as a solution of an Helmholtz equation in the quasi-stationary case.

2.3 Project status

2.3.1 Progress todate

A report /1/ is made on the analytical solutions 2.2.a. The problem of partial flow is solved by finite element method for two-dimensional geometries.

2.3.2 Essential results

The test of the numerical methods by the problem of homogenious excitation has shown that the finite element method is the better method to solve our partial flow problems.

The experiment with a plate model shows a good agreement with the numerically calculated calibration curve.

The influence of the nearness of the crack relative to an edge of the test object could be cleared; the calibration curve must be modified.

2.4 Next steps

a) Application of the finite element method to three-dimensional geometries as screw-threads and nozzel surfaces in reactor pressure vessels.

b) Integration of Helmholtz equation in the case of alternating current.

2.5 Relations with other projects

none

2.6 Reference Documents

/1/ Dobmann, G.:
Potentialsondenverfahren. Berechnung und Messung zweidimensionaler Potentialsverteilungen bei homogener Anregung. IzfP-Bericht Nr. 750102-TW.

3. Guided ultrasonic waves electrodynamically excited

3.1 Particular objective

A technique shall be elaborated which allows to inspect heat exchanger tubes for lingitudinal and transverse cracks by guided waves. The guided waves shall be excited and received contactlessly by electrodynamic transducers.

3.2 Experimental facilities

The equipment has been in essential the same as last year with the difference that the electrodynamic receiving coil is located within the tube directly below the transmitting coil outside of the tube. This technique improves the local resolution of flaw echoes.

3.3 Project status

3.3.1 Progress todate

New transducer coils have been developed for the purpose of exciting torsional waves in the tubes. It was found that longitudinal flaws are detected best with torsional modes while transverse flaws give better echoes with longitudinal modes.

3.3.2 Essential results

In addition to transverse flaws longitudinal flaws with a maximal depth of 5 per cent of the tube wall thickness could be detected. But the ratio of signal to coherent disturbances is lower for longitudinal than for transverse flaws.

3.4 Next steps

The prevention of unwanted modes and the suppression of the coherent disturbances have to be more improved, especially for torsional modes. A first trial shall be strated to apply electrodynamic transducers as angle probe, the angle being adjustable by frequency change.

3.5 Relations with other projects

none

3.6 Reference Documents

Mohr, W.:
Elastische Rohrwellen. Erste experimentelle Ergebnisse zur Prüfung dünnwandiger Rohre mit elektrodynamisch angeregten geführten Ultraschallwellen.
IzfP-Bericht Nr. 750513-TW zu RS 102 - 18, Teil II.

Classification: 12.3.	
Title 1 (Original Language): Automatisierung und EDV der US-Impulsechoprüfung (RS 102-17 - II.3.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Gesell-schaft, IzfP
Title 2 (english): Automation of Pulse-Echo-Testing and Electronic Data Processing of the Results	Project Leader: R. Werneyer
Initiated (Date): 15.10.1974 Completed (Date): 31.12.1977 Status: continuing Last Updating (Date): December 1975	

1. General aim

During the production of reactor pressure vessels both, the basic materials and the single components undergo a nondestructive and fully volumetric test. After the completion of the reactor a basic test ist made, and then, at certain intervals, fully automatic in-service inspections take place under more complicated conditions. This is to locate and evaluate defects. Evaluation is effected in a computer after the reconstruction of a three-dimensional flaw picture not dependent upon the different testing methods. These flaw pictures are compared with the specifications and standards.

2. Particular objective

The aim of this project ist the reconstruction of flaw pictures with information on geometry, locus and flaw kind. Only such infomation allows the application of fracture mechanical criteria of empirical rules for the valuation of the defects. To obtain such reconstructions the following steps have to be carried out:

- gathering information on different arrangements and sound pathways for each volume element (tandem, pulse-echo- angle-transmission a.o.);
- utilization of infomation included in echo-dynamics;
- providing for using the infomation of the spectrum at a later date;
- utilization of neighbouring relations;

- correction of transfer fluctuations beside changes of sound pressure caused by sound field characteristics, suppression of form echoes and redisturbances;
- reconstruction of a three-dimensional, completely corrected flaw picture allowing different two-dimensional projections in any direction.

3. Experimental equipment and programme

For the realization of this concept it is necessary to place efficient hardware electronic and software – the pre-condition for constructing and displaying a three-dimensional flaw picture with the help of a computer – at disposal. Software has to be developed under consideration of neighbouring relations. This is realized by the associated institute IITB Karlsruhe. The preparation of the hardware component, running parallel with the realization of experiments to get data records of artificial and natural flaws is done by IzfP.

4. Project status

An experimental set-up has been completed for the development of data records of artificial and natural defects in test specimens. Tests at a specimen have shown that the artificial flaws can be detected with the test methods pulse-echo and tandem with a signal-to-noise ratio not under 30 dB.

The experimental set-up consists of a modefied ultrasonic testing device, a function generator for external triggering, an analog-digital converter and a computer HP 2100 with teletype and tape puncher as connected periphery.

Triggered by the trigger signal the transmitter sends a pulse to the probe. The same trigger signal starts the analog-digital-converter (ADC). The transmitting pulse and the signals received by the probe – the whole image of the ultrasonic testing device – are amplified and rectified and fed to the ADC, which digitizes and stores the signal. The image displayed in 2048 words is read into computer on request. By the aid of a programme amplitudes and transit times are determined and stored for each arriving pulse.

When the test cycle (transmission, pulse-echo on the left, pulse-echo on the right, tandem) is finished, x- and y-coordinates are read into computer, the data block ist made available and displayed on teletype and punched tape.

Arithmetic for correction of amplitude, for determination of size and locus of defects in pulse-echo and tandem testing has been developed.
This arithmetic has been brought into a data processing programme in Fortran IV. By this programme

those data sets are filtered out which represent a flaw echo. Appearing ambiguities are cleared at the same time and the appertaining flaw characteristics and coordinates are analyzed. Documentation of defects has been made in projection on x, y- and x, z-plane.

5. Next steps

Firstly, data recording programmes will be shifted to automatic recording. The necessary devices as automatically running manipulator and hardware components will be developed by the IzfP. The programmes provided for evaluation will be joined in a dialog system under consideration of expansions as display unit, digicoder or intermediate results.

6. Relations with other projects

This programme is partly based on the results of the reactor safety research projects RS 27-1 and RS 27-2 and will be carried trough in coordination with the still running projects RS 27-2 and RS 169.

Classification: 12.3	
Title 1 (Original Language): Holografische Interferometrie (RS 102-22 - II.3.2., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** Fraunhofer-Gesellschaft, IFAM
Title 2 (english): Holografic Interferometry	**Project Leader:** Dr. Jüptner
Initiated (Date): 1.7.1975 **Status:** continuing **Completed (Date):** 31.3.1976 **Last Updating (Date):** December 1975	

1. General Aim

Using holografic interferometry, the field of deformations of workpiece surfaces is registered by interferometric comparison of two states stored in a hologram. The appearing interference fringe system yields quantitative informations about the deformation in three axes. To produce the two stages the unloaded object has to be holographed, consequently it is subjected to the normal load and holographed again.

The field of deformation, obtained in this manner, can be used for calculation of the states of stress even of geometrically complicated workpieces. Furthermore by qualitative and quantitative evaluation weak points become evident. In fracture mechanical tests informations about the stress intensity could be obtained from experimentally determined plastic zones, and thus about the influence of crack displacement.

2. Particular Objective

It appears reasonable to apply the above mentioned advantages of holografic interferometry for safety controls of reactor components. However, before this method can be employed for components with possible interieur cracks or other damages, on the basis of this research project the following problems have to be solved:

a) Can defects be detected in components with thick walls?

b) Which are the conditions for critical stresses to be determined from the holografic fringe system?

c) Is it possible to recognize critical deformations in the region of weld seams?

d) Which equipment is suitable for measurements without vibration protection?

These points shall be investigated in this project.

3. Experimental Facilities and Programme

The research work is done in the following stages:

1. Optimizing the experimental equipment
   - 1.1 Construction of suitable loading equipment
   - 1.2 Installation of the holografic system

2. Measuring of deformations with the normal holografic equipment
   - 2.1 Measuring of deformations
   - 2.2 Comparison with theoretical models

3. Measurements without vibration isolation.

4. Project Status

4.1 Progress todate

4.1.1 Experimental work

In the first experiments it was tried to couple the reference and the object beam very rigid while making the hologram. Although good results were obtained with holograms of vibrating objects, the equipment is not suitable in this form for practical measurements because the holografed area is too small.
The equipment for holografic deformation measurements of thick sheet is in construction.

4.1.2 Theoretical work

To answer the above mentioned question, if defects can be recognized and if the endangerment can be quantified, a first theoretical model was made on the basis of fracture mechanics. By the aid of this model first estimations about the deformations on the surfaces of a thick wall are possible.

4.2 Essential Results

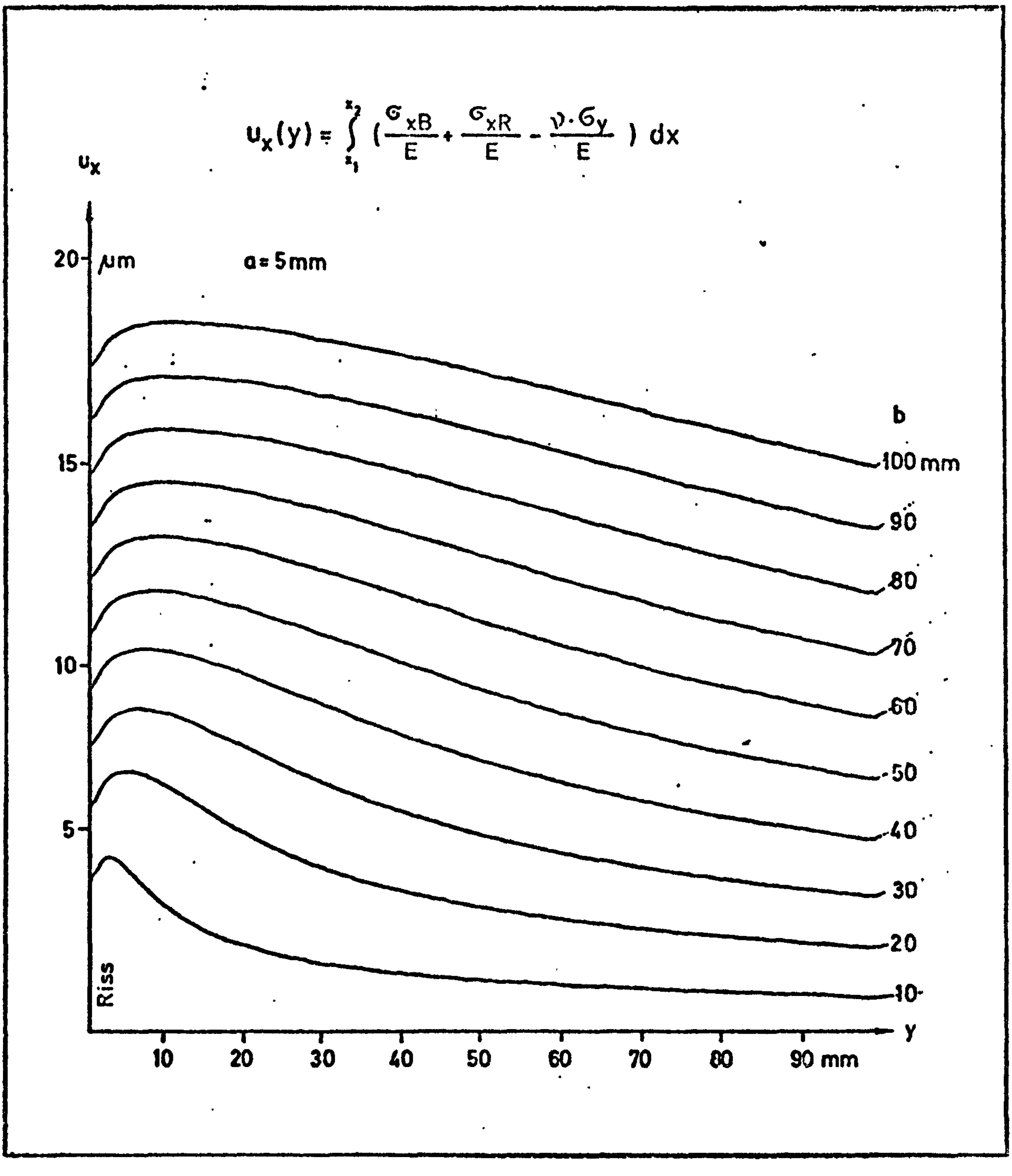
Calculations are made by the following assumptions:

a) The area under consideration is a plain part or pipe wall.
b) It is permitted to use the fracture mechanical model with plain stress.
c) The stresses due to the deformation field of a pipe under internal pressure can be superposed additively with the stresses calculated from the fracture mechanical model.
d) The tangential stress in the pipe is the normal stress for the fracture mechanical calculation.
e) The geometrical distribution of the stress field of a fracture specimen does not change essentially with the specimens' thickness.

Some of these assumptions have to be investigated more precisely in the future. But using the assumptions, a system of equations was set up and the deformation in the axial direction of the pipe u_x along a tangential line (y-direction) was calculated, not considering an additional constant movement, pic. 1. This first estimation suggests the possibility that cracks can be detected by measurements n the surface and the stress intensity factor can be calculated.

5. Next steps

In further theoretical work a better model for the deformation calculation shall be evaluated. Hereby the assumption in the last chapter must certainly be modified. With the calculated deformations, the holografic interference fringe system can be evaluated. This leads to statements about the possibilities of holografic measurements and yields data for experimental investigations.
For comparison deformation measurements will be prepared. Besides this the experiments with equipments which are not vibration isolated will be continued.



Pic. 1: Deformation (u_x) of a pipe under internal pressure in radial direction along a tangential line (coordinate y)

Classification: 12.3		
Title 1 (Original Language): Ermittlung von Schweißeigenspannung mit Hilfe der Röntgenographie im ambulanten Einsatz (RS 13o - II.3.2, Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: ALLIANZ
Title 2 (english): The Determination of Welding Residual Stresses with the Aid of Ambulant X-Ray Diffraction		Project Leader: Dr.Christian Dr.Elfinger
Initiated (Date): .6.1974	Completed (Date): 31.5.1976	
Status: continuing	Last Updating (Date): December 1975	

1. General aim

The object of the work in progress is to clarify to what extent it is possible to discuss, from the aspect of strength factors, the residual stresses determined by non-destructive x-ray diffraction measuring in the surface or in the surface-adjacent zones of welded joints, and to compare the values thus obtained with the results procured by destructive, mechanical measuring procedures. Furthermore, to perfect measuring methods and apparatus in such manner as to render possible their ambulant application in respect of reactor components.

2. Particular objectives

a) Investigation into the evolution of surface stresses and stress gradients in and in the vicinity of welded joints, dependent on welding parameters and heat treatment and their connection with the mechanical properties.

b) Comparative measurements using destructive measuring procedures.

c) Adaptation and reconstruction of the x-ray diffraction measuring equipment for ambulant use and test measuring on reactor components.

3.1. Experimental facilities

In x-ray diffraction measuring, two centre-point free x-ray goniometers are employed, of which one is reconstructed for ambulant use. A hydraulic tension apparatus is available for determining the x-ray diffraction elasticity constants.

3.2. Research program

Residual stress investigations are carried out on eight series of specimens prepared by submerged arc welding from St.52, a filler metal S 2 NiCrMo 1 (4 mm ∅) and granulated flux LW 32o, 9 by 595. As parameters for the individual specimen series, both the welding data and the heat treatment of the weld seams during and subsequent to welding are varied.

Determination of the residual stresses is effected perpendicular to the weld seam and in relation to the depth.

Comparative measurements using destructive measuring procedures are conducted.

A goniometer for ambulant use is being designed and constructed and initial test measurements carried out.

4. Project status

4.1. Progress to date

Determination of the quality values of the welded joint in respect of the first four specimens was effected in line with the AD Leaflets "Manufacture of Welded Pressure Vessels, Series H".

X-ray diffraction stress measurements, perpendicular to the weld seam, in the surface of the heat-affected zone and the weld material, as well as determination of the stress gradients relative to the depth, were carried out on the first four specimens. The measurements in respect of the other four specimens are in progress.

Comparative measurements using the Ring-Kern-Procedure to determine the stress gradients were conducted on two specimen series.

An initial prototype of a goniometer for ambulant use has been constructed and the first test measurements carried out on major components. The final design version of this ambulant goniometer has now been completed.

4.2. Essential results

The course of the residual stresses perpendicular to the weld seam on the surface of the specimen show a clear relationship to the heat treatment of the weld seams during and subsequent to welding.

The stress gradient $\frac{d\sigma}{dt}$, relative to depth, in 3o mm thick weld seams not subjected to heat treatment was practically nil up to a depth of $t \leqq 3$ mm. This result also agreed with comparative measurements using the Ring-Kern-Procedure, which is not a non-destructive method. One reservation which must, however, be made is that mechanically non-treated weld seams are present. If this is not the case, at least o,4 - o,5 mm must be electrolytically removed.

An initial prototype of a goniometer for ambulant huse has been constructed and the first test measurements carried out. The design for a final improved version has been completed.

5. Next steps

a) Construction of the improved goniometer for ambulant use.

b) Completion of stress measurements on the test weld specimens.

c) Test measurements on reactor components.

6. Relation with other projects

7. Reference documents

8. Degree of availability

Classification: 12.3	
Title 1 (Original Language): Hologaphisch-interferometrische Untersuchungen zur Material- und Bauteilprüfung von Reaktorkomponenten (RS 148 - II.3.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Dornier System
Title 2 (english): Holographic-Interferometric Investigations in the Field of Testing Pressure Vessel Materials and Components	Project Leader: Dr. Grünewald
Initiated (Date): 1.10.1974 Completed (Date): 31.1.1976 Status: continuing Last Updating (Date): December 1975	

1. General aim

The aim of the investigations is to establish whether and how the holographic interferometry can be applied to material testing in the field of nuclear safety.

2. Particular objectives

Theoretical and experimental investigations to clarify the possibility of finding faults in pressure vessels with thick walls.

3.1 Experimental facilities

Holographic test facility with a 4 W-argon-laser

3.2 Research program

Theoretical investigations:

- calculation of the deformation of pressure vessels under load and of the influence of local irregularities of different shape and position on the deformations
- calculation of interferograms of pressure vessels with irregularities under pressure load.

Experimental investigations:

- qualitative and quantitative holographic deformation measurements of steel tubes with faults under pressure load (tubes of 50, 100, 203, 419 mm diameter and a wall thickness of 1.6, 2.2, 6.3 and 10.6 mm; the extension of the faults amounts from 6 % to 30 % of the wall thickness), flange-lids and welded steelpieces under thermal load, and plastic deformation measurements around the top of a crack under stress.

4. Project status

4.1 Progress to date

The investigations listed under 3.2 have been carried out. Within these investigations different modes of load have been proved and the sensitivity of the system has been increased by means of varying the optical arrangement and testing procedure.

4.2 Essential results

As one essential result it can be stated that cracks in steel tubes with a diameter-to-wall-thickness relation of about 40 can be detected even when the size of the crack amounts to 6 % of the wall thickness and the crack is located at the inner side of the wall while during testing the outer side of the wall is inspected. The sensitivity was obtained in an advanced testing mode. By optical means the normal interference pattern was suppressed and the high sensitivity for the detection of the irregularities was obtained.

The test pieces with cracks under a stress load led to interferograms with an irregular fringe-shape in the range of plastic deformation near the cracks. The calculation of deformations and strains from the interferograms are not yet completed.

5. Next steps

Evaluation and discussion of the results

6. Relation with other projects

The investigations were carried out in connection to the projects RS 132, executed by the TU Hannover, and RS 102-22, performed by the Fraunhofer-Gesellschaft, IFaM, Bremen.

Classification: 12.3		
Title 1 (Original Language): Entwicklung von zerstörungsfreien Prüfverfahren für Wiederholungsprüfungen an Reaktordruckbehältern (RS 2702 - II.4.2., Jahresbericht A 75)		**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** MAN Nürnberg
Title 2 (english): Development of Non-Destructive Inspection Techniques for In-Service Inspection Tests of Reactor Pressure Vessels		**Project Leader:** J. Lindner
Initiated (Date): 1.10.1972	**Completed (Date):** December 1975	
Status: finished	**Last Updating (Date):** December 1975	

1. General aim

The general aim of this task is the development of inspection equipment with the aid of which in-service inspection tests can be carried out on pressurized components of nuclear reactors in the scope considered necessary by the authorities, such inspection tests to yield qualified test results without subjecting the inspection personnel to excessive irradiation. The inspection tests under this research programme are limited to ultrasonic techniques. In addition to the Federal Institute for Materials Testing (BAM) the task is being handled by the firms of Kraftwerk Union AG (KWU), Krautkrämer GmbH (KK) and Maschinenfabrik Augsburg-Nürnberg AG (MAN).

2. Particular objectives

The task is divided into the following objectives:

1. Basic investigation into means of inspection by ultrasonic techniques and the influence of boundary conditions; inspection studies on primary circuit systems.

2. Development of inspection systems for the various inspection areas.

3. Development, manufacture and testing of manipulating equipment for internal and external tests.

4. Onward development of components of electronic systems and data logging systems for ultrasonic inspection work.

5. Testing of inspection systems.

3. Experimental facilities and programmes

Experimental facilities were set up to simulate internal and external test conditions for the purpose of testing the inspection system embracing the probes, the manipulator, the ultrasonic electronic gear and the data logging system. To this end, a large pressure vessel wall specimen with artificial flaws was devised and subjected to a voluminous inspection programme to establish the influence of various parameters on the detectability of flaws. In addition, there are experimental programmes to examine the influence of the cladding, the inclination of separations, the crack structure and the crack position in the material on flaw detectability.

4. Project status

4.1 Progress to date

A large number of tests were made to establish the influence of the cladding on the probability of fault detection and on fault indication behaviour. In addition, the influence of other parameters was examined, such as the angle and surface structure of planar reflectors and the influence of surface conditions of the test specimen. These investigations were made theoretically and were supplemented by experiments.

Ultrasonic probe systems were improved for better interpretation of the inspection results and higher probability of fault detection.

Ultrasonic electronics were matched to the increased demands on the probe systems. Appropriate computer programmes were set up to evaluate the data stored on tape.

Manipulators for internal and external inspection and for inspection of the spherical closure and closure head of the reactor pressure vessel were advanced to a level where most of the pressure vessel can be tested by remote control.

The whole system was tested by employing the equipment set forth in 3) and improved on the basis of experience gained.

4.2 Essential results

All equipment and processes have been improved to a stage where almost all essential areas of the reactor pressure vessel can be examined with a high probability of fault detection. Limitations need only be imposed in certain areas regarding the interpretation of inspection results.

5. Next steps

Work under this programme will terminate on 31st December 1975. As individual problems have not yet been solved completely, all parties are making efforts to obtain a continuation order covering an extension of the inspection zones of the reactor pressure vessel as well as an improvement in the interpretation of the inspection results. In addition, the parties want to set up a working programme for inspection of the primary circuit.

6. Relation with other projects

Reference is made to what has been said in Report A 74.

7. Reference documents

The participants of the present project issued the following comprehensive reports in German during the period under review:

Author	Title	Date
M.A.N.	Investigations into the inspection ability of inner nozzle radii during external inspection of the reactor pressure vessel	March 1975
KK	Noise level due to grain structure and size at spherical closure of a reactor pressure vessel	April 1975

Author	Title	Date
KK	Ultrasonic inspection of perforated spherical bottom closure of light water reactor pressure vessels	April 1975
BAM	Manual ultrasonic single-probe echo dynamics on artificial reflectors in spherical closure of a reactor pressure vessel	May 1975
M.A.N.	Study of a telescopic mast for the central mast manipulator	May 1975
BAM	Tandem echo dynamics	October 1975
KK	Study of reflection behaviour of various inclin ed notches in connection with the inspection of the spherical closure of light water reactor pressure vessels.	October 1975

8. Degree of availability

A relatively small number of reports were issued. A limited number are still obtainable from the issuer or IRS.

Classification: 12.3	
Title 1 (Original Language): Entwicklung und Bau einer Ultraschall-Prüf- und Auswerteelektronik einschließlich Vorverstärkerkasten (RS 169 - II.3.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Krautkrämer GmbH, Köln
Title 2 (english): Development and Design of an Ultrasonic Testing and Evaluating Electronic System Including Pre-Amplifier Box	**Project Leader:** G. Gutmann
Initiated (Date): July 1, 1975 **Status:** continuing	**Completed (Date):** approx. May 1976 **Last Updating (Date):** December 1975

1. General Aim

The development of an ultrasonic electronic evaluating system which will meet the present day safety requirements and which can be used universally for all types of reactors. Based on past experience and in view of envisaged technical development this electronic evaluating system is to meet the following requirements:

maximum number of channels: 60
maximum speed which can be attained by the manipulating system: 100 mm/s.
rate of scanning: 1 scanning cycle per mm.

In addition the function and stability of the installation are to be continually monitored.

2. Particular Objectives

a) Digital gate cards
b) Summing amplifier
c) Write-read-memory
d) Code converter
e) Miniature transmitter
f) MF-generator
g) Miniature pre-amplifier

3.1 Experimental Facilities

Development and design of the evaluating electronics according to the established requirements.

3.2 Research Program

Development and Desing of the pre-amplifier box.

4. Project Status

4.1 Progress to Date

All the electronic modules especially required for this electronic evaluation have been developed and tested as a functioning model. The requirements regarding function, accuracy and stability of the modules have been met.
The design and development of the pre-amplifier box has been started. Special attention has been given to a compact size and to the prevailing on-site temperatures.

4.2 Essential Results

The complete blockschematic design has been developed and the design and development of the entire installation has commenced whereby the following requirements have been met:

a) Maximum number of channels: 60
b) Maximum possible speed of the manipulating system: 100 mm/s
c) Scanning rate of the testing system: 1 testing cycle per mm.
d) Fully automatic and continuously effective function and stability check.
e) Maximum allowable deviation for stability ± 1.5 dB.

5. Next Steps

a) Carrying on with the work commenced in 3.1 and 3.2.
b) Special research for suppressing interference from multi-conductor systems.

6. Relation with Other Projects

RS 2702: Development of Non-Destructive Inspection Techniques for In-Service Inspection Tests of Reactore Pressure Vessels, MAN Nürnberg

RS 102-17: Automation of Pulse-Echo-Testing and Electronic Data Processing of the Results, Fraunhofer-Gesellschaft, IzfP, Saarbrücken

7. Reference Documents

Semi-annual report in the series IRS-Forschungsberichte
Report period: July 1 - December 31, 1975
IRS-F-28

8. Degree of Availability

The report is available on request from:
Institut für Reaktorsicherheit der Technischen Überwachungs-Vereine e.V., Köln, Abteilung Forschungsbetreuung

13. SYSTEMS OPTIMISATION, STANDARDISATION, NEW CONCEPTS

Classification: 13	
Title 1 (Original Language): Verminderung der Primärkreiskontamination durch Einsatz eines Elektromagnetfilters. Versuche zur prinzipiellen Durchführbarkeit. (RS 171 - II.1.6., Jahresbericht A 75).	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Reduction of the Contamination of Primary System Components by Electromagnetic Filters	Project Leader: Dr. Neeb
Initiated (Date): 7. 75 — Completed (Date): 30. 4. 76 Status: Continuing — Last Updating (Date): 31. 12. 75	

General Aim and Particular Objectives

The possibilities for reduction of corrosion product activity and contamination by means of electromagnetic filters, which have been used in fossile heated plants will be studied. Some tests will be conducted in order to find out, whether electromagnetic filters can be used in the primary system of a PWR.

Experimental Facilities

he experimental studies will be carried out in the Pressurized Water Chemistry Loop (DCK) of KWU-Erlangen. A high-pressure electromagnetic filter has been fabricated and is used in connection with the DCK. Measurements by normal and radio-tracer analytical techniques are done by the KWU laboratories.

Research Program

a) Tests on the effeciency of electromagnetic filters will be conducted, using low concentrations of corrosion products in highly cleaned water at temperatures up to 300 °C

b) Investigation of the influence of boric acid and LiOH with concentrations as used in the coolant system of a PWR

c) Metal release rates of the balls of the electromagnetic filter in boric acid and LiOH-solutions, as far as possible in this experimental facility

d) The possibility of backflushing has to be tested with respect to the corrosion product oxides, seperated on the balls of the electromagnetic filter

Project Status/Progress to Date

After cleaning the Pressure-Water-Chemical-Test-Loop the electromagnetic filter was installed. First tests were run with highly cleaned water and 1 - 2 ppm Li. The corrosion product concentration (Fe, Ni, Cr contents) was determined in the water phase before and after the filter.

Project Status/Essential Results

From the first tests using non-radioactive measurement techniques only very raw data on the decontamination effects in the e. m. f. test device could be obtained. In the lower concentration range (below 20 ppb Fe) obvionsly no deposition of corrosion products was effected, at higher concentrations the results are somewhat erratic. More information is expected from the radiotracer tests.

Next Steps

a) Radiotracer tests with radioactive corrosion products

b) Tests with boric acid and LiOH-water

c) Metal release from different ball materials

d) Backflushing tests

Relation with Other Projects

No relations with other projects.

Reference Documents/Degree of Availability

No reports available.

14. PROBABILISTIC METHODS OF SAFETY ANALYSIS

Classification 14	
Title 1 Vurdering af systemers pålidelighed	**Country** Denmark **Sponsor** DAEC Risø **ORGANIZATION** DAEC Risø
Title 2 Evaluation of the Reliability of Systems	**Project leader** Hans Erik Kongsø
Initiated 1970 **Completed** **Status**: In progress **Last updating** Currently	**Scientists**: Hans Erik Kongsø

1. General aim To develop methods for evaluation of the reliability of systems as an aid in design and safety analysis.

2. Particular objectives Development of computer codes, primarily based upon the simulation method.

3. Experimental facilities Not applicable.

4. Project status

4.1. Progress to date The computer code, RELY 4, has been developed. The code is written in ALGOL, it is of the Monte Carlo type and both direct simulation and simulation with variance reduction can be applied. The code calculates reliability and availability of systems both with and without repair.

A cumputer code, MARKOVA, has been developed. The code is written in FORTRAN, it is an analytic program, and it can be used for solving Markov equations, both the differential equations and the linear equations at equilibrium.

A computer program, REDIS, has been developed. The code is written in FORTRAN, and it is of the Monte Carlo type, based upon direct simulation (no variance reduction). The program calculates a series of reliability characteristics for a given system, and it has been designed with particular emphasis on flexibility and for analysis of details in component- and system performance.

The program can be used for any system in principle, and it is designed for analysis of standby systems, taking into account the various kinds of errors, that can arise during operation as well as standby conditions; in addition the program can take into consideration, that in standby systems certain groups of components will have common operation and standby periods. Dependencies between faults and faults with spread on the data can also be considered, and routine testing of selected groups of components can be taken into account.

4.2. Essential results The REDIS and MARKOVA codes have been used for calculation of the unavailability of the feedpump systems in two conventional power plants.

5. Next steps

Future work will include development of computer codes for very large systems and attempts to apply the Monte Carlo technique for cause-consequence analysis.

6. Relation with other projects The projects has close relations to the Risø-project concerning the cause-consequence method (classification 6).

7. Reference documents RELY 4. A Monte Carlo Computer Program for System Reliability Analysis. Risø-M-1500, H.E. Kongsø. June 6, 1972.

REDIS, A Computer Program for System Reliability Analysis by direct Simulation. IAEA-SM-195/17. H.E. Kongsø. April 1975.

REDIS, A Computer Program for System Reliability Analysis by direct Simulation. Program Description and Manual. Risø-M-1781. Hans Erik Kongsø and Robert Korre Larsen, June 1975.

8. Availability The project information is freely available apart from cases, where commercial interests may be violated.

Classification 14

Title 1 Probabilistisk brudmekanik	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 Probabilistic Fracture Mechanics	Project leader Per Becher
Initiated 1970 Completed Status: In progress Last updating Currently	Scientists Per Becher

1. **General aim** To develop methods for evaluation of the reliability of pressure vessels for the purpose of safety analysis.

2. **Particular objectives** Development of computer codes, based upon probabilistic methods and fracture mechanics, for evaluation of the reliability of pressure vessels.

3. **Experimental facilities**

4. Project status

4.1 Progress to date

Two computer codes was developed, PFM 690 and PEP 706. Both are written in FORTRAN in versions for a Burroughs B 6700 and a IBM 370/160.

PFM 690 is a monte carlo program, which calculates the crack distribution as a function of time, on the basis of an initial crack, crack growth characteristics and stress transients. PEP 706 uses monte carlo with importance sampling and calculates the probability of failure of a pressure vessel, based upon the distribution functions for cracks, stresses, yield strength and charpy-V.

4.2 Essential results

5. Next steps So far the models have been based upon linear elastic fracture mechanics, but attempts will be made to incorporate elastic-plastic fracture mechanics into the models.

In coorperation with other research institutions and power reactor vendors analysis of actual reactor pressure vessels will be carried out, based upon more pertinent data from practical experience. The specific objectivs being assessment of the influence of the neutron embrittlement and development of a new crack growth model.

6. Relation with other projects The project has relations to the Risø projects, dealing with stress analysis of primary steel components and with evaluation of the reliability of systems.

7. Reference documents Risø-M-1650. Application of statistical linear elastic fracture mechanics to pressure vessel reliability Analysis, september 1973. P.E. Becher, Arne Pedersen. (Presented as paper no. M6/4 at 2. Smirt Conf., Berlin 10 - 14th September 1973).

8. Availability The project information is freely available apart from cases, where commercial interests may be violated.

Classification: 14	
Title 1 (Original Language): Die Berechnung der Zuverlässigkeit großer komplexer Systeme nach der Methode der relevanten Pfade (RS 106 - I.1.8., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** TU Berlin
Title 2 (english): Calculation of Reliability Data for Complex Systems Using the Success Paths Method	**Project Leader:** Prof. Dr. Memmert
Initiated (Date): ..4.1973 — **Completed (Date):** 30.9.1975	
Status: finished — **Last Updating (Date):** December 1975	

1. General aim

A computer program is to be written to determine reliability data for complex systems with components which

1. are unrepairable,
2. can be repaired immediately if failure occured,
3. can be maintained in periodical intervals,
4. or whose occurrence probability of failure is given by a time independent value.

The results to be obtained are

a) the occurrence probability of the first systemfailure Q(t),
b) the unavailability U(t) and its mean value $\bar{U}$,
c) the mean time of systemfailure as a function of time,
d) the steady state values, mean time to repair MTTR, and mean time between failures MTBF, if the system is in steady state.

2. Particular objectives

The program should employ analytical methods to find the cut-sets of the system andto compute the reliability data of these sets, which constitute first order approximation to the data of the system. It should be possible to use fault-trees and logical block diagrams and to use a cut-off method for large systems (more than 3.000 cut-sets).

3. Experimental facilities

Not relevant.

4. Project status

4.1 Progress to date

A report on the theoretical background of ARP II was given last year /1/2/3/4/, so that now the application of this analytical reliability program to an example is of interest.
The largely simplified version of a nuclear power plant energy supply is shown in fig. 1 and the corresponding fault-tree in fig.2.

1	:	short circuit in the main circuit connection (self-indicating)
2 and 6	:	transformer failure (maintenanced)
3 and 4	:	switch opens unintentional (self-indicating)
5	:	generator or reactor failure (self-indicating)
7	:	short circuit in the 6 kV supply (self-indicating)
8 and 9	:	switch does not open on demand.

As the top-event (Gate 16) the break-down of the 6 kV supply is defined.
To compute this example with ARP II the following card-deck is required (real values in the format E 10.0 and integer values in the format I 3).

1. card:	9,16,7	max. number of components max. number of components and gates max. number of gate-entrances + 2
2. card:	10,1,1,8	and-gate 10(1) with entrances 1 and 8
3. card:	11,1,5,9	and-gate 11(1) with entrances 5 and 9
4. card:	12,0,1,2	or-gate 12(0) with entrances 1 and 2
⋮		
17. card:	16,0,10,11,15,6,7	or-gate 16(o) with entrances 10,11,15,6 and 7
18. card:	100	the computation is carried out using a hundred time steps

19. card: 100.0 time step length is a hundred hours
20. card: $1.0 \cdot 10^5$ mean time to failure of component 1

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.
.

29. card: $1.0 \cdot 10^4$ mean time to failure of component 9
30. card: 100.0 (first column) mean time to repair of component 1

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.
.

39. card: (second column) 500.0 maintenance period of component 9

4.2 Essential results

The results are presented in fig. 3. 2,5 sec of computing time were required on the CDC 6500 of the TU-Berlin.

5. Next steps

Work was finished at Sept. 1975.

6. Relation with other projects

At present no other work is being done on reliability problems in connection with the reactor safety program of the BMFT. The author's present report is related to the following analytical computer programs:
ARMM-69 and GAMM (both programs use the theorem of Bayes as a basic method),
SICHERHEIT (utilizing the Truth-Table-Method), and
SAP (semi analytical with repair and inspection).

7. Reference documents

1 Quarterly reports in the series IRS-Forschungsberichte

2 Richter, G. and Memmert, G.: Berechnungen von Zuverlässigkeitsdaten komplexer Systeme mit analytischen Methoden. TUBIK 28, Report of the Institut für Kerntechnik, TU-Berlin (Oct. 1973)

3 Richter, G.: Die Berechnung der Zuverlässigkeit großer komplexer Systeme nach der Methode der relevanten Pfade. (Dissertation am Institut für Kerntechnik, TU-Berlin, Febr.1975)

4 Kamarinopoulos, L. and Richter, G.: Vergleichende Untersuchungen verschiedener Methoden zur Berechnung der Ausfallwahrscheinlichkeit komplexer Systeme (Atomkernenergie 26/2),

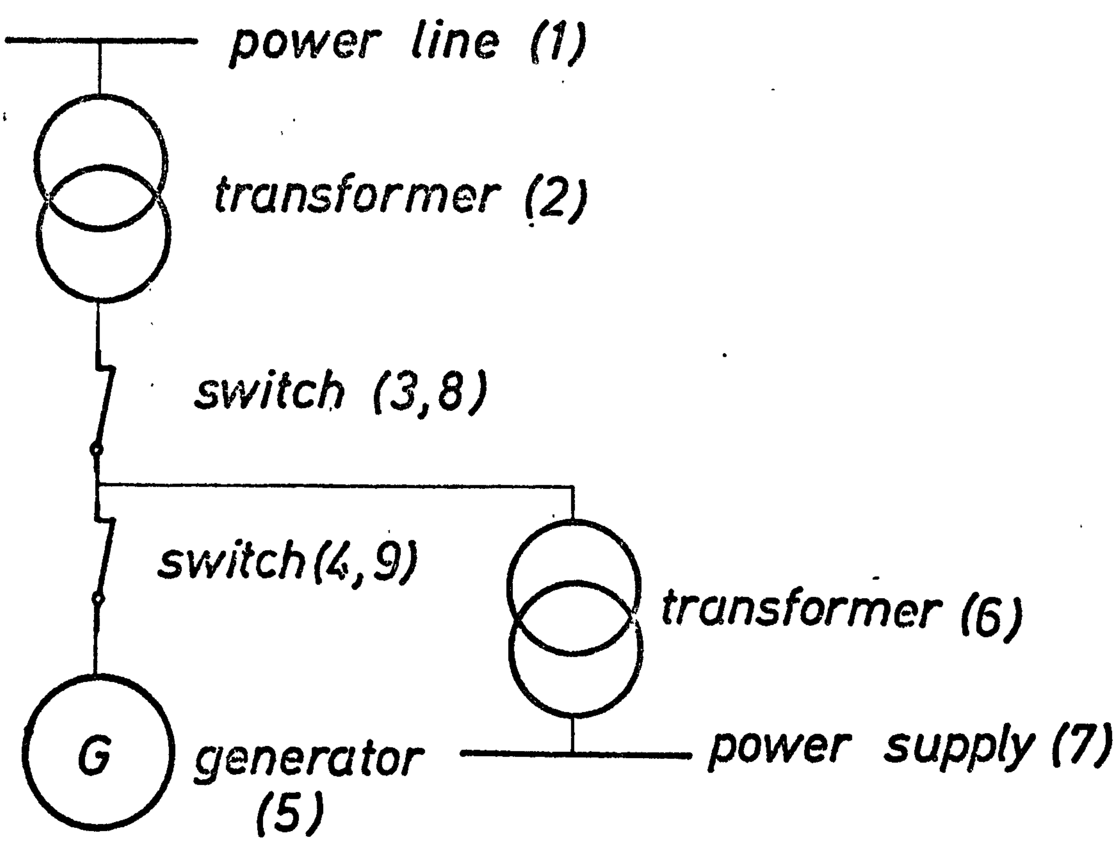

Figure 1

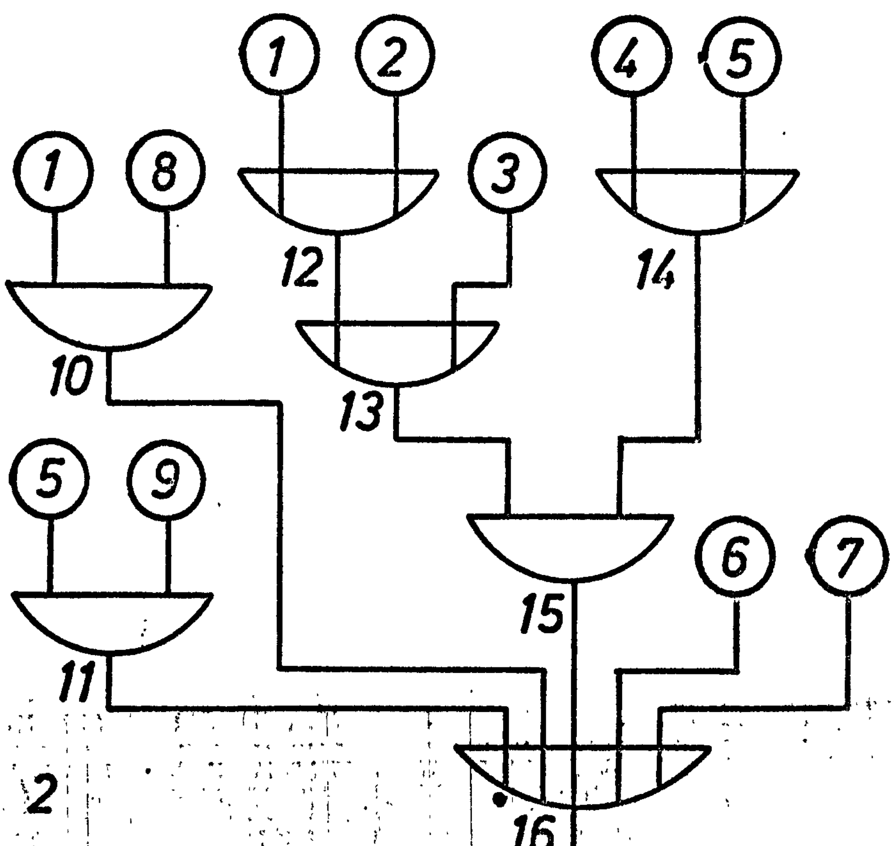

Figure 2

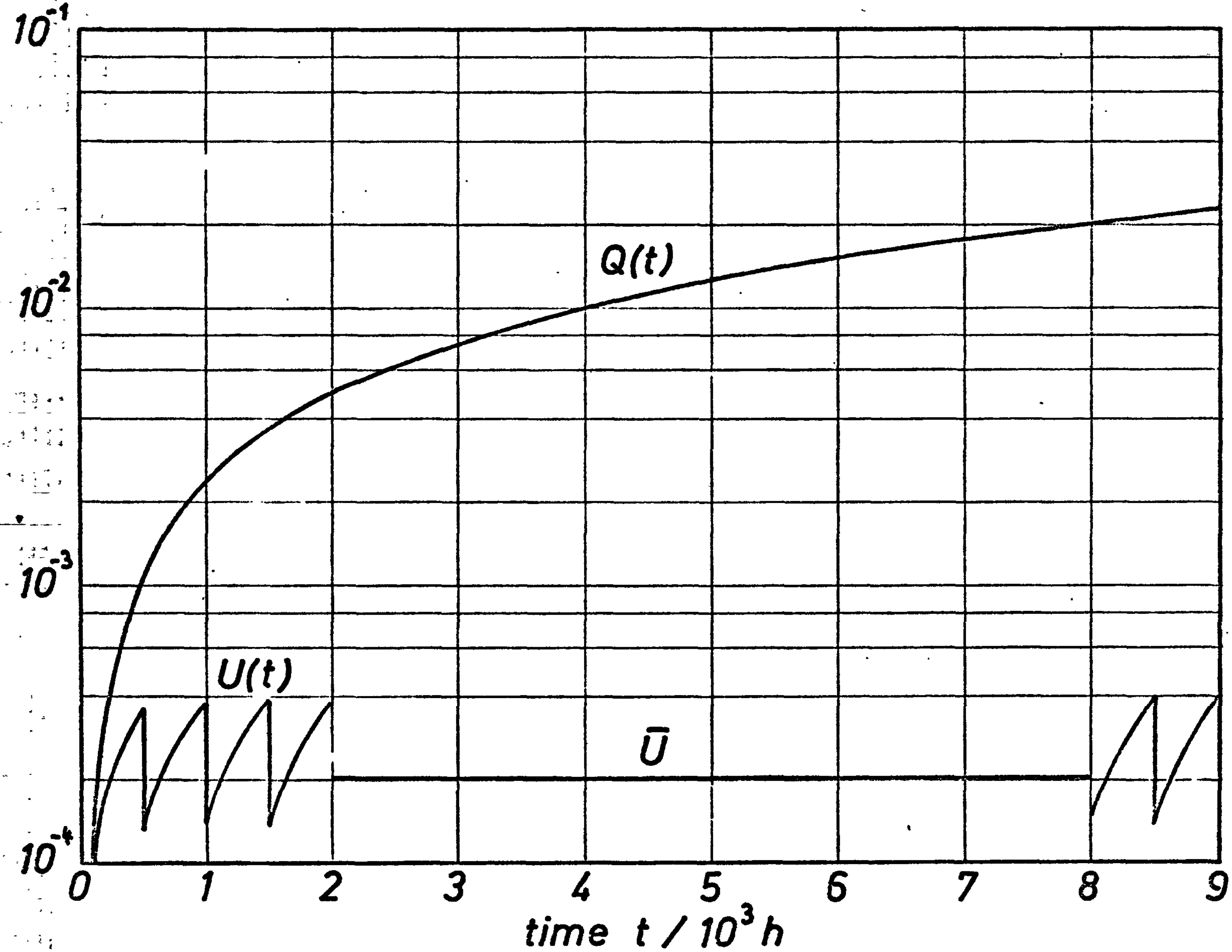

Figure 3

Classification: 14	
Title 1 (Original Language): Sicherheits- und Zuverlässigkeitsanalyse kerntechnischer Anlagen (RS 134 - I.1.8., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Ingenieurbüro Buck
Title 2 (english): Safety and Reliability Analysis of Nuclear Plants	Project Leader: Dr. Buck
Initiated (Date): *.10.1974 Status: continuing	Completed (Date): 30.9.1976 Last Updating (Date): December 1975

1. General aim

This research program concerns the safety and reliability analysis of nuclear plants. In part 1 the so-called long-term effects were investigated, in part 2 it is intended to calculate the short-term effects.

2. Particular objectives

In part 1 of this program the reliability parameters of the emergency core cooling system of the BWR/6 were determined following a large loss-of-coolant accident.

In part 2 it is intended to develop a computer code in order to investigate the reliability parameters of nuclear plants during the short-term period following an accident or during off-normal conditions.

Special objectives are:
- Loss-of-coolant accidents.
- Anticipated transients without scram.
- Reactivity initiated accidents.

3. Project status

3.1. Progress to date

Part 1, i.e. the investigation of the long-term effects, is completed. The first step of part 2 was to outline the possible scope of applications for the new technique. The development of the code was started.

3.2. Essential results

The BWR/6 emergency core cooling system, which was investigated in part 1, contained some hundred components. The particular aim was to determine the influence of the accuracy of the component data on the resulting reliability parameters of the system.

As expected, the results showed that there is considerable influence from the bias while no influence from the precision is detectable.

In part 2 it is expected that the new, highly sophisticated method will yield results which are much more trustworthy than those calculated with the ordinary fault tree method for the short-term period concerned.

4. Next step

The next step will be the completion of the computer program and a first application on one of the problems mentioned above.

5. Reference documents

The latest results of part 1 are contained in report no. 23-2/8.1., with the title:
Reliability Analysis of the BWR/6 Emergency Core-Cooling System.

The first report of part 2, no. 23-1/8.2., gives a description of the scope of applications for the new method. The title is:
Objectives for Short-term Reliability Analysis in LWRs.
The reports are written in German.

6. Degree of availability

The reports may be available from the Bundesministerium für Forschung und Technologie, Bonn.

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Classification: 14	
Title 1 (Original Language): Prozeßmodelle und Reaktorregelung (ATT 085 A - I.1.8., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (english): Process Models and Reactor Control	Project Leader: Dr. W. Bastl Dr. A. Höld
Initiated (Date): Completed (Date):	
Status: continuing Last Updating (Date): December 1975	

1. General Aim

Development of theoretical process models in order to describe the dynamic behaviour of different types of power reactors (together with their conventional parts).

Application of modern control theory, as developed in recent years for complex multiple input/output systems, to on-line control of large nuclear power plants by process computers.

2. Particular Objectives

Since several years the institute is occupied with the development of theoretical process models (and corresponding digital codes) which are able to describe either the frequency response and stability or the transient behaviour of different types of nuclear power reactors (BWRs, PWRs, sodium cooled reactors) and their conventional parts (heat exchangers, steam generators).

In addition, efforts have been made to apply the optimal control theory to nuclear power plants. The work covered two main objectives:

a) Simulation of an optimal core power distribution control loop.

b) Development of an optimal control for a BWR nuclear power plant.

3. Experimental Facilities and Program

Not relevant.

4. Project Status

4.1 Progress to Date

a) Frequency response models:

The development work on frequency response models for BWR, PWR and sodium cooled (both thermal and fast breeder) reactor plants and their computer codes ADYSMO, ADYPMO and ADYMSOR /1 - 5/, and for conventional parts of such nuclear power plants, e.g. for U-tube steam generators /6/ or shell-and-tube type counterflow heat exchangers and its digital code TUMHEX /7, 8/ could be brought in to a final state by concluding the work on a similar frequency response model for U-tube heat exchangers /9/.

Additionally, a combined numerical and analytical method (and, based on it, the digital code FRETI /10/) has been established in order to be able to calculate the transient behaviour of linear or linearized systems after being perturbed by a step or any other system driving function from its pregiven frequency response data. Thus, evidence about the transient behaviour of the above mentioned reactor types and/or their conventional parts can be obtained indirectly by using their frequency response models.

b) Non-linear transient model for U-tube steam generators:

Within the scope of establishing a non-linear transient model for the description of the dynamic behaviour of a PWR nuclear power plant after large excursions (see also project "Dynamics of Large Water Reactors") the derivation of an extensive transient model for U-tube steam generators at natural circulation conditions has been one of the main tasks during this period.

The heat exchanging zone of the generator has been assumed to be represented by a number of identical U-tubes. The primary coolant flows on the inner (having a co-current and a counter-current branch), the secondary on the outer side of the tube. The coolant channels can axially be divided into max. 7, the tube wall radially into max. 3 nodes. The steam dome is characterized by its parameters: system pressure, water volume and water level. The steam removal system takes into account a steam mass flow perturbation which can be caused by isolation-, safety-, bypass-, turbine-trip and/or turbine-control valves, but also feedbacks from the steam turbine. It has been considered that perturbations from the system pressure or from the feed water line propagate through the downcomer thus entering the heat exchanging tubes only after a certain delay-time. The resulting mass flow from natural circulation will be calculated by solving the momentum balance equations along the whole recirculation line.

The theoretical work in establishing the non-linear transient model is finished. The work on the corresponding computer code UTSG ("U-tube steam generator") is done to a large extent.

c) Core power distribution control:

In order to create the possibility of assuming the safety of a digital power distribution control of a reactor core, a simulation of the core control loop has been implemented on the computer. The control program CORECON was taken over from the OECD Halden Project. The program is based on /15/ and has been enlarged and further developed by the Halden Project. CORECON has been coupled with the three-dimensional core simulation QUABOX. To test the control performance on a large core, the input data of QUABOX were chosen to correspond with the core of the Biblis-A PWR. Test runs have shown the correctness of coupling and proved that the control, which was developed for small reactors, is extendable to large cores.

d) Simulation of BWR nuclear power plants:

To test modern control methods there exists the need for a model which simulates the real power plant with sufficient accuracy, but avoiding to cause too heavy a computational burden. For simulating a BWR nuclear power plant a lumped parameter non-linear transient model and its computer code LIMBO ("lumped parameter simulation model for BWRs") has been developed /11/ and - by adding artificial noise - enlarged to the code NIMBO.

A linearization and a strong simplification of the model to only 5 characteristic parameters of a BWR plant yielded finally the code SIMBO which is small enough to be applicable within the modern control methods.

e) Model building and parameter identification of a BWR nuclear power plant for use in optimal control:

For purposes of adaptive optimal control, low-order models are needed which describe the dynamic behaviour of the process to be controlled. Besides the code SIMBO mentioned in the preceeding section, two non-linear model structures have been set up. The models were both derived by simplifying the basic physical equations. The simplifying assumptions differ for the two models, resulting in a different degree of complexity.

The model parameters were identified by two procedures. First, a least-square cost function was minimized by means of a gradient technique. The method yields good estimates when process and measurements are noise-free, but the parameters are biased when the process and the measurements are noisy. For this case, a much more powerful algorithm based on a maximum-a-posteriori (MAP) criterion was employed. The algorithm also estimates the parameter values as functions of time /13, 14/.

f) Optimal control of a BWR nuclear power plant:

Computer codes had to be provided to perform the necessary tasks in an optimal control loop. These tasks consist in optimal filtering to include non-measurable variables, in on-line parameter identification and in calculating the optimal control. For steady-state control and for small deviations from the steady-state, a feedback law has been derived including weights on the velocity of state and input variables. For large deviations, a control program able to handle hard constraints in both states and inputs has been completed.

4.2 Essential Results

a)
The combined application of TUMHEX and FRETI calculating the transient behaviour of a shell-and-tube type counterflow heat exchanger at a special situation showed very satisfactory results. A report about this procedure had been given at the NAMUR-VDI/VDE-GMR.conference at Frankfurt-Höchst /12/.

b)
The digital code UTSG is still in its testing state, only preliminary calculations with encouraging results could be done.

c)
A simulation of the core power distribution control loop has been set up, making possible the analysis of performance and safety of the control system.

d)
The work on the digital codes LIMBO, NIMBO and SIMBO has been completed.

e)
Models of a BWR nuclear power plant have been established for control purposes. The employment of powerful identification procedures has proved to be effective in adapting low-order models to complex systems.

f)
The necessary sub-tasks for an optimal control loop have been programmed. Making use of the identified model, they now allow the coupling of the BWR simulation (LIMBO) and the optimal control.

5. Next Steps

a)
The work on frequency response models is terminated.

b)
The digital code UTSG will be tested. The natural-circulation part, which is not yet built in, will be added to the program and test calculations again undertaken.

c)
Further improvements of the core control simulation have to be considered to achieve a close correspondence to a real case. They comprehend extensions on the core simulation and determination of the best allocation of sensors and controls. An extended control program including xenon transient control and boron as power controller will be taken over from the OECD Halden Project and tested on the QUABOX core simulation. The simulation of the whole loop will be utilized to analyse failures of the control such as in-core detector failures or rod drive failures.

d)
If necessary, smaller improvements on the digital codes LIMBO, NIMBO or SIMBO have to be done.

e)
The models will be improved and enlarged. Each model change has to be followed by an identification run to adapt the new model to the plant simulation.

f)
An optimal steady-state control acting on the BWR-plant-simulation LIMBO will be established, its performance tested and control failures analysed. Further, large load changes have to be considered, leading to an adaptive control able to handle constraints on the main plant variables.

6. Relation with Other Projects

Co-operation with GKSS Geesthacht and the OECD Halden Project on behalf of optimal power distribution.

7. Reference Documents

/1/
A. Höld
Linear analytical model describing the frequency response and stability behaviour of a boiling light water reactor
Nucl. Eng. and Design 16, 1971, p. 103 - 136

/2/
A. Höld
Dynamic and stability behaviour of BWRs with respect to perturbations in reactivity, primary steam flow, water inlet velocity or subcooling. Part II: Computer code ADYSMO
MRR 76, March 1970

/3/
A. Höld
Analytical one-dimensional frequency response and stability model for PWR nuclear power plants
Nucl. Eng. and Design 33, 1975, p. 336 - 352

/4/
P. Ausserer, A. Höld
ADYPMO - Ein Digitalprogramm zur Berechnung des Frequenzgang- und Stabilitätsverhaltens einer leichtwassermoderierten Druckwasserreaktoranlage. Programmbeschreibung
MRR-P-10, Juni 1974

/5/
E. Sädtler
Analytisches eindimensionales Einkanal-Modell zur Berechnung des Fre-

[illegible] A computer code for the calculation of the frequency re-
sponse behaviour of shell-and-tube type counterflow heat exchangers
with respect to primary and secondary inlet temperature and mass flow

/11/
A. Höld, A. Hora
LIMBO - A non-linear lumped parameter simulation model for BWR nuclear power plants. Program description
To be published

/12/
A. Höld
Programme zur Simulation des dynamischen Verhaltens von Geradrohrbündel-Gegenstrom-Wärmeaustauschern
Tagung des NAMUR-VDI/VDE-GMR-Ausschusses, Frankfurt-Höchst, Juni 1975

/13/
D. Beraha, E. Sädtler, K. Volf
Model building and parameter identification of a boiling water nuclear reactor plant
National Systems Conference, Roorkee, India, February/March 1976

/14/
K. Volf
Parameter indentification of a BWR nuclear power plant model for use in optimal control
To be published

/15/
R. Grumbach
Entwicklung und Erprobung eines Verfahrens zur Rechner-Regelung der Neutronenflußverteilung in einem Leistungsreaktor
Dissertation, TU München, April 1972

/16/
Quarterly Reports in the Series IRS-Forschungsberichte

8. Degree of Availability

Documents are available through
Laboratorium für Reaktorregelung und Anlagensicherung
D-8046 Garching
Federal Republic of Germany

Reports MRR-I are confidential and therefore normally not available.

Classification: 14	
Title 1 (Original Language): System- und Zuverlässigkeitsanalyse an Leichtwasserreaktoren (ATT 085 A - I.1.8., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (english): System and Reliability Analysis on Light Water Reactors	Project Leader: Dr. W. Bastl, Dr. Hörtner, Dr. Kafka
Initiated (Date): — Completed (Date):	
Status: continuing — Last Updating (Date): December 1975	

1. General Aim

The detailed investigation of possible accident sequences, especially of their probability and the implementation of the results in a risk concept.

‸. Particular Objectives

- Detailed investigation of typical 1200 MWe-PWR-plants.
- Set-up of cause-consequence diagrams in order to find out the most important accident sequences.
- Detailed analysis of the engineered safety features installed to cope with the associated accidents and estimation of their reliabilities.
- Systematic investigation of important electronic equipments with the aim of finding the failure rates of the relevant failure modes.

3. Experimental Facilities and Program

Not relevant.

4. Project Status

4.1 Progress to Date

For the nuclear power plants considered, the reliability investigations of the engineered safety features needed to control the design basis accident "Large LOCA" were extended to the long-term residual heat removal after the accident. Furthermore, a very detailed reliability analysis of the reactor control rod scram systems for different transients was carried out.

For different electronic modules additional investigations were performed in order to find out the various failure modes and to evaluate the associated failure rates.

On the basis of five test fault trees, a comparison of different reliability computer programs from various institutes was done. The results of our CRESS programs were in good agreement with the results gained from other institutes. A detailed report is under preparation in co-operation with the Institute for Reactor Safety (IRS).

A report with critical comments on the Reactor Safety Study WASH-1400 was prepared /8/. Possible contributions of our institute to a risk assessment of a German PWR plant were specified /17/.

4.2 Essential Results

The carried-out investigations for the engineered safety features were discussed in papers /4, 5, 10/; detailed reports describing the analyses were worked out /13, 16/. The essential results of reliability investigations for electronic devices were described in detail in /1, 14/.

In the framework of the BMI-project "Risk Strategy" the results of the risk seminar that took place in Bonn in February 1975, were compiled in /3/. The present state of the risk concept was discussed /6/. A paper on the problems of risk quantification with respect to reliability investigations was presented /11/.

By means of the results presented in /2/ it is possible to estimate the influence of various parameters on the unavailability of cold-standby systems. Especially, the influence of consecutive testing and of the repair time is discussed in detail.

5. Next Steps

The results of the reliability analysis of the engineered safety features necessary to cope with the design basis accident "Large LOCA" will be discussed more comprehensively in part II of /13/.

When this work is completed, the engineered safety features which are necessary for the control of further important accidents will be analysed. In this context the small LOCA in the primary circuit and accidents in the secondary feedwater steam circuit should be mentioned. It is conceivable, that in addition to the fault tree investigations, accident sequence analyses need to be done. Hence the application of dynamic calculation models to determine the various accident parameters is necessary.

6. Relation with Other Projects

Blow-down and refilling calculations for LOCA's as well as investigations concerning the dynamic behaviour of plant in the case of transient events are done within other sections of the LRA. These investigations are taken as the basis for the reliability analysis.

7. Reference Documents

/1/
S. Goßner
Theoretische Untersuchung des Ausfallverhaltens eines dynamischen Grenzwertmelders
MRR 143, Februar 1975

/2/
E. Dressler, H. Spindler
Die Nichtverfügbarkeit von Bereitschaftssystemen in Anhängigkeit von Teststrategie und Reparaturzeit
MRR 144, März 1975

/3/
H. Hörtner, P. Kafka
Ergebnisbericht zum Risikoseminar am 25.2.1975 in Bonn. Interner Bericht
MRR-I-38, März 1975

/4/
E. Dressler, H. Hörtner, E. Nieckau, H. Spindler
Zuverlässigkeitsuntersuchung von Sicherheitssystemen des Kernkraftwerkes Biblis, Block A
Kerntechnik 17, Heft 4, April 1975, S. 160

/5/
H. Hörtner, H. Spindler
Vergleich der Zuverlässigkeiten von vermaschten und entmaschten Sicherheitssystemen
Reaktortagung, Nürnberg, April 1975

/6/
P. Kafka
Risikokonzept - Ein Beitrag zum Stand und den Entwicklungen auf diesem Gebiet
Reaktortagung, Nürnberg, April 1975

/7/
E. Dressler, H. Spindler
The Influence of Test and Repair Strategies on Reactor Safety
IAEA-Symposium on Reliability of Nuclear Power Plants
Innsbruck, April 1975

/8/
H.P. Balfanz, P. Kafka
Kritischer Bericht zur Reaktorsicherheitsstudie (WASH-1400). Interner Bericht
Interner IRS-Bericht 75, MRR-I-44, April 1975

/9/
W. Bastl, S. Goßner, E. Nieckau
Zuverlässigkeit sicherheitstechnischer wichtiger elektronischer Geräte von Reaktoren
ATM, Blatt V 8238-4, Mai 1975, S. 81

/10/
E. Dressler, H. Hörtner, E. Nieckau, H. Spindler
Ergebnisse einer Zuverlässigkeitsanalyse bei Verwendung unterschiedlicher Ausfalldaten
atw 20, Heft 6, Juni 1975, S. 294

/11/
P. Kafka, W. Bastl
Problems of Risk Quantification Discussed on the Basis of Reliability Investigations of Nuclear Power Plant Systems
Paper presented at Engineering Foundation Conference on Risk Benefit Methodology and Application, Asilomar, California, September 1975

/12/
H. Hoermann, E. Nieckau
Neuerungen in der Leit-, Überwachungs- und Automatisierungstechnik zur Steuerung und Überwachung von Kernkraftwerken
Kerntechnik 17, Heft 9/10, 1975, S. 441

/13/
H. Hörtner, E. Dressler, E. Nieckau, H. Spindler
Kernkraftwerk Biblis, Block A
Zuverlässigkeitsuntersuchung der für die Beherrschung des Auslegungsstörfalls "Bruch einer kalten Hauptkühlmittelleitung" erforderlichen Sicherheitssysteme, Teil I. Vertraulicher Bericht
MRR-V-9, November 1975

/14/
S. Goßner
Zuverlässigkeitsanalyse an dem Impedanzwandler CW 012 (S&F). Vertraulicher Bericht
MRR-V-10, November 1975

/15/
E. Nieckau
Criteria and Requirements for Reactor Protection Systems
Seminar of Rules, Regulations and Laws for Nuclear Installations in Brazil, Comissao Nacional de Energia Nuclear-CNEN, Rio de Janeiro, Brazil, November/Dezember 1975

/16/
Zuverlässigkeitsanalysen der Reaktorschnellabschaltung durch das Reaktorschutzsystem für einen Druck- und Siedewasserreaktor. Interner Bericht
MRR-I-54, Dezember 1975

/17/
W. Bastl, P. Kafka
Spezifikation des Beitrages des LRA zum BMFT-Projekt "Risikostudie". Interner Bericht
MRR-I-53, September 1975

/18/
Quarterly Reports in the Series IRS-Forschungsberichte

8. Degree of Availability

Documents are available through
Laboratorium für Reaktorregelung und Anlagensicherung
D-8046 Garching
Federal Republic of Germany

Reports MRR-I and MRR-V are confidential and therefore normally not available.

Classification: 14		
Title 1 (Original Language): Vergleich von Rechenprogrammen zur Zuverlässigkeits-analyse von Kernkraftwerken (RS 172 - I.1.8., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: IRS, Köln
Title 2 (english): Comparison of Calculation Programs for the Reliability of Nuclear Plants		Project Leader: Otto
Initiated (Date): 25.8.1975	Completed (Date): 15.2.1976	
Status: continuing	Last Updating (Date): December 1975	

1. General Aim

The aim of the project is the comparison of the efficiency of existing codes which are used to evaluate complicated fault trees. The comparison and the assessment of the methodical status reached by the programs, shall serve as basis of decision for future research in fault tree analysis within the research project "Risk and Reliability" which is sponsored by the Federal Ministry of Research and Technology in the framework of the Reactor Safety Research Program

2. Particular objectives

Analytical and simulative programs of various institutions are to be checked, in calculating several fault trees, which were worked in advance by three institutions (1. Institut für Reaktorsicherheit - IRS, 2. Laboratorium für Reaktorregelung und Anlagensicherung - LRA, 3. Gesellschaft für Kernforschung - GfK) which themselves took part in the calculations.

3.1 Experimental facilities

-

3.2 Research program

The fault trees having been set as the task considered e.g. repair work, inspection and switching processes by multiphase calculations. A short description of the fault trees is contained in table 1.

4. Project status

4.1 Progrecs to date

Following instructions and project leaders took part in the project in calculating the fault trees completely or partially:

Ingenieurbüro Buck, Lemförde	(Dr. Buck)
Gesellschaft für Kernforschung, Karlsruhe - GfK -	(Dr. Caldarola)
Industrieanlangen-Betriebsgesellschaft, Ottobrunn - IABG	(Dr. Keller)
Interatom, Bensberg	(Dr. Rosenhauer)
Institut für Reaktorsicherheit, Köln - IRS -	(Dr. Heuser)
Euratom, Ispra	(J. Amesz)
Laboratorium für Reaktorregelung und Anlagensicherung, Garching - LRA -	(Dr. Kafka)
Messerschmitt, Bölkow u. Blohm, München - MBB -	(Fiedler)
Technische Universität Berlin, Institut für Kerntechnik	(Prof. Memmert)
TÜV Rheinland, Köln	(K.R. Hartung)

The results received by the various institutions and the programs used were judged provisionally by IRS, LRA und GfK and a provisional final report had been written up.

4.2 Essential results

The first evaluation had demonstrated that there is a high standard of code development among all institutions, taking part in the calculations. Most of the reports, produced by the single institutions, however, did not show in detail the exact status of the methodical development of the programs. Furthermore in some cases there were big deviations in the results.

As it seemed necessary to solve these problems in discussions with each calculating institution only a provisional assessment could be made till the end of the originally proposed date.

5. Next steps

The status of development of the various programs and differing results shall be discussed with the various institutions having taken part in the project. Then the final report will be written.

6. Relation with other projects

RS 106 Calculation of reliability data for complex systems using the success paths method

RS 134 Safety and reliability analysis of nuclear plants

7. Reference documents

Quarterly reports in the series IRS-Forschungsberichte.

Report period	July - Sept.	1975	IRS - F - 27	(german)
	Oct. - Dec.	1975	IRS - F - 28	(german)

8. Degree of availability

Documents are available through IRS, D-5000 Köln, Federal Republic of Germany

Table 1: Description of Fault Trees

FT	General Problem	Concrete Example	Asked Values
1	Calculation of reliability and availability taking into account inspection and repair	System of components with constant failure rates, regularily inspected, repair times small as compared to duration of time for detection of failures	Reliability and availability of system, mean elapse of time until detection of system failure
2	Testing of simulation routines (changes between service states)	Failure behaviour of an electrical energy supply for a nuclear plant	System reliability and availability, contributions of subsystems to the failure behaviour of the total system
3	Differentiation between standby and operating state	Emergency power supply	Reliability and availability of the emergency power distribution
4A 4B	Reliability analysis of a safety system	Emergency energic system	4A: Mean unreliability of the safety system 4B: Failure Probability in the after cooling phase most important failure combinations
5	Simulation of an investigation from the Rasmussen report	Evaluation of a High-Pressure Injection System of a PWR	Minimal cut sets, unreliability of the system

Classification: 14	
Title 1 (Original Language): Risiko- und Zuverlässigkeitsanalyse von Kernkraftwerken (PNS 4530 - I.1.8., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Risk and Reliability Analysis of Nuclear Power Plants	**Project Leader:** L. Caldarola
Initiated (Date): 1.1.1974 **Status:** continuing	**Completed (Date):** **Last Updating (Date):** December 1975

1. General Aim

General aim of the project is the reliability analysis of systems with respect to nuclear power plants.

2. Particular Objectives

The objective of the program is to develop an analytical computer program as well as a Monte-Carlo-program for fault tree evaluation and to develop methods for the automatic construction of fault trees.

3. Experimental Facilities and Research Program

The project is a mathematical and analytical one. Therefore no experimental facilities are needed. Each program step depends on the results of the step before. (See Essential Results and Next Steps).

4. Project Status

4.1/2 Progress to Date and Essential Results

A computer program has been written and tested. Main features of the program are the following

- Four different types (classes) of components can be handled
  a) Irreparable components
  b) Reparable components, with failures which are immediately detected (revealed faults)
  c) Reparable components, with failures which are detected upon inspection (unrevealed faults)

d) Components characterized by a constant unavailability

- Capability to identify all minimal cut sets in order of importance
- Capability to analyse systems characterized by two phases on following the other in time

- Compatibility test allows one to find out if the two fault trees are logically compatible.

- The following quantities can be calculated as functions of time

  a) System point unavailability

  b) System average unavailability (unavailability averaged over the time)

  c) System failure intensity

  d) System average failure intensity (failure intensity averaged over the time)

  e) System integral of failure intensity (in the present from of the program this quantity is used as system unreliability)

- The system average failure rate (1/MTTF) can also be calculated with all components intact at the initial state /2/.

The computer program needs 480 K in CPU. This allows to analyse fault trees either with a maximum of 256 elements and 200 points on each time axis or with a maximum of 2000 elements and no calculation on the time axis (average and maximum values only).

A new and more sophisticated theory /1-2/ to calculate the unreliability of complex repairable systems has been developed. The method is based on a set of integral equations each one referring to a specific minimal cut set of the system. Each integral equation links the unavailability of a minimal cut set to its failure probability density distribution and to the probability that the minimal cut set is down at time "t" under the conditions that it was down at time "t'" ($t' \leq t$).

Three test problems of the "BMFT Leistungsprüfung" were solved with the Karlsruhe computer program. The results are shown in /3/.

5. Next Steps

- A fifth class of components will be included in the computer program. This will include components which are inspected and repaired when they are demanded to operate (unrevealed faults).

- The theory under section 4 will be incorporated in the computer program.
- The feature to handle systems characterized by many operating states will be built in the program.
- The feature to handle correlated faults (among various components) will be also built in the program.

6. Relation with Other Projects

No direct relation but cooperation.

7. Reference Documents

/1/ L. Caldarola
"A method for the calculation of the cumulative failure probability distribution of complex repairable systems"
(being published in "Nuclear Engineering and Design")

/2/ L. Caldarola,
"Calculation of the mean time to failure of a redundant repairable system"
Bericht Nr. IRE/1/4530/17/75

/3/ L.Caldarola and A. Wickenhäuser,
"BMFT-Leistungsprüfung - Vergleich von Rechenprogrammen zur Zuverlässigkeitsanalyse von Kernkraftwerken, RS 172 - Abschlußbericht" (in English)
Bericht Nr. IRE/1/4530/19/75
PNS 59/75

8. Degree of Availability

/2/, /3/, Internal Reports

PROJECT TITLE : Fault analysis of the conventional island in a LWR nuclear power plant.	CLASSIFICATION 14
SPONSORING COUNTRY :	ORGANISATION : FRANCO TOSI S.p.A.
DATE INITIATED : April 15, 1975 DATE COMPLETED : April 15, 1976	PROJECT LEADER : Vittorio BEDOGNI

Description :

1) Research program

- Fault tree definition of the conventional island of a LWR nuclear power plant
- Fault data collection of the system's components. Analysis and treatment of the fault data.
- Reliability and availability evaluation of the systems.
- Parametric analysis of the systems reliability varying the failure rate of the critical elements and of the components whose fault data are not available or not sufficiently reliable.

2) Facilities

- Computer and computer codes.

3) Reference documents

- R.E. Barlow, F. Proschan "Mathematical theory of reliability" - John Wiley & Son.,Inc., New York
- A.G. Colombo "CADI, a computer code for system availability and reliability evaluation" - Report EUR 4940 e (1973)
- J.B. Fussel "A formal methodology for fault-tree construction" Nuclear Science and Engineering, 52 (1973), pp. 421-432

4) Related projects

None (F.Tosi)

5) The works is done in relation to the design of the ENEL V and VII nuclear power stations.

PROJECT TITLE : Reactor Safety Studies via Noise Analysis	CLASSIFICATION 14 - 8
SPONSORING COUNTRY : Italy	ORGANISATION : C.N.E.N.
DATE INITIATED : 1.1.1975 DATE COMPLETED : .... (in progress)	PROJECT LEADER . N. Pacilio, LFCR

Description :

The research program deals with applications of the theory of stochastic processes to reactor safety studies. They are directed to assessing the correct performance of in-core and ex-core instrumentation, to safety monitoring and early detection of abnormal operating conditions and/or malfunctions. (Preliminary research for early failure detection).

Efforts up to now are :

- set-up of a general theory for multi-detector reactor-noise analysis in ergodic conditions, non-equilibrium conditions and during pulsed experiments ;
- experimental determination of nuclear reactor kinetic parameters and control of multiplying assemblies that must be kept subcritical ;

The general theory will soon be expanded to include problems related to heat transfer (fluctuations in temperature, pressure and void volumes) and to a preliminary analysis of significance of acoustic noise analysis.

Facilities

- four light-water reactors (ROSPO, RANA, RITMO, TRIGA) and a copper-reflected highly-enriched fast reactor (TAPIRO)

Reference list

- N.PACILIO, V.M. JORIO - Review of International Noise Conference held in Rome (October 21-25, 1974) CNEN Report. RT/FI(74)47

5

RCN	CLASSIFICATION: 14
TITLE: Faalweg en faalkansvoorspellingen van reactorsystemen.	COUNTRY: NETHERLANDS SPONSOR: Ministry of Social Affairs ORGANIZATION: Reactor Centrum Nederland
TITLE (ENGLISH LANGUAGE): Failure mode and failure rate prediction of reactor systems.	PROJECTLEADER: W.F. Heshuysen
INITIATED: June 1974 COMPLETED: STATUS: progressing LAST UPDATING: March 1976	SCIENTISTS: -

1. General aim

   To predict the chance of failure of systems on basis of possible failure modes and component reliability, taking into account accidental as well as systematic failures due to external causes or inherent faults of components.

2. Particular objectives

   The first stage of the programme consists of:
   - making an inventory of the general available methods of reliability analysis and "data banks" on failure rates
   - establishing a standard procedure to collect data on failures in nuclear power stations in The Netherlands
   - developing procedures to evaluate the reliability of reactor systems for specific cases applicable to Dutch power reactors.

3. Experimental facilities

   Not applicable.

4. Project status: -

5. Next steps: -

6. Relations with other projects: -

7. Reference documents: -

8. Degree of availability

   The reports will be written in the English language. They will be submitted to the Ministry of Social Affairs, Postbus 69, Voorburg.

Classification 14	
Title 1 Faalkans-analyse met behulp van gebeurtenissen- en foutenbomen	Country The Netherlands Organization KEMA
Title 2 Failure analysis by application of event- and fault trees	Projectleader R.W. van Otterloo

1. General aim

To analyse new systems or changes in existing systems. And by application of this technique to make decisions concerning these new systems or changes in existing systems in the right way.

2. Particular objectives

- Analysis of the unavailability of reactor safety systems and of the unreliability of reactor systems.
- Analysis of the probability of different groups of radioactive releases of a nuclear power reactor.

3. Experimental facilities

Not applicable.

4. Project status

Methods and computer codes have been compared.

5. Next steps

Not applicable.

6. Relation with other projects

This project was started to do the "Risk analysis of the fuel cycle in the Netherlands" (RASIN-study) which was finished in June 1975.

7. Reference documents

See 6.

Several applications of this method are written in the Dutch and the English language.

8. Degree of availability

Through the organization KEMA.

15. INTERRELATION BETWEEN REACTOR PLANT AND OPERATING PERSONNEL

Classification
15

Title 1	COUNTRY Denmark
Anlægsbeskyttelse	SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 "Study of the Process Operator".	Project leader L.P. Goodstein
Initiated Approx 1966 - in its present form 1973. Status: progressing Completed: Last updating	Scientists: L.P. Goodstein O.M. Pedersen C.D. Grønberg P.Z. Skanborg J. Rasmussen

1. General aim

2. Particular objectives

To study the process operator - his work situation and procedures - together with methods for supporting him, especially in abnormal situations.

3. Experimental facilities and programme

- Experimental computer-controlled date acquisition and CRT display system.
- Interactive CRT graphics terminal.

4. Project status

- Experiments have been run to demonstrate the feasibility of supporting the operator by means of computer-controlled displays. Ideas from these experiments have been incorporated in several of the newest Danish power stations.
- Information from the analysis of tape recordings made by operators at Danish power stations have been used in the formulation of models of the operator as a system component. Monitoring of operator performance at the DR 2 and DR 3 reactors at Risø is also currently active. Other studies made in cooperation with Danish power stations have been concentrated on the control room tasks of the operator.

5. Next steps

- A program to continue with the testing and validation of these models of the operator is being planned. Improved methods for the evaluation of risks which include human error will be evaluated.

6. Relation with other projects

7. Reference documents

Goodstein, L.P., "Operator Communications in Modern Process Plant" IEE Conference on Display, Loughborough 1971.

Rasmussen, J., "On the Communication between Operators and Instrumentation in Automatic Process Plants. Risø-M-686 (1968).

Rasmussen, J., "The Human Data Procession as a System Component - Bits and Pieces of a Model - Risø-M-1722 6/74.

1. Budget

2. Personnel

- Approx 4 Engineers.

805

Classification: 15.1	
Title 1 (Original Language): Ermittlung und Analyse menschlicher Funktionen beim Betrieb von Kernkraftwerken (SR 100 - II.5.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: TÜV Rheinland, Köln
Title 2 (english): Identification and Analysis of Functions of the Human Operator in the Operation of Nuclear Power Plants	Project Leader: Prof. Dr. Kuhlmann
Initiated (Date): ril 1973 **Status:** finished	**Completed (Date):** 1975 **Last Updating (Date):** December 1975

1. General aim

Operation experiences in nuclear power plants indicate that the human operator considerably contributes to the system output on normal as well as on faulty operating conditions. Thus an effective power plant design and development has to take into systematic account the possibilities and limitations of the human element. In view of this the study aims at identifying and analyzing the functions of operating ıd maintenance personnel.

2. Particular objectives

The objective of the project is to find out to what extent the above requirements are met in existing plants, i.e. to identify what the human operators are required to do and how they achieve it. It is expected that these analyses will result in basic Human Factors recommendations for the design of safe and effective operating and maintenance procedures.

3. Program

3.1 Experimental facilities

None.

3.2 Research program

(1) Search, compile and annotate the literature of both technical and Human Factors origin in order to give a survey of the present state of the art.

(2) Analyze functions of the plant pèrsonnel in terms of tasks and responsibilities assigned to them by operating procedure manuals, work regulations, etc.

(3) Analyze incident reports to better take into account random events that cannot be observed directly.

(4) Administer interviews to operating and management personnel in order to obtain informations on the functions as well as on their subjective evaluation.

(5) Observe directly personnel carrying out routine and, if possible, non-routine work. This is considered a major source of information for identifying and analyzing tasks.

4. Project status

4.1 Progress to date

Functions of operating and maintenance personnel in three nuclear power plants were investigated by consideration of both nonfunctional and dynamic factors. The latter were established by observations of tasks carried out, by interviews administered to personnel, and by analyzing reported operations of all kinds. The former ones included recording and evaluating equipment, facilities, job aids, regulations, etc., as well as interviews. Besides, pertinent literature references and incident reports were compiled and evaluated. The data collection activities have been terminated.

4.2 Essential results

Results were obtained on the role of static characteristics such as the design of displays, controls, and communication facilities, structural and technical design, job aids, and personnel and work organization. Further results concerning the performance of tasks include the use of information transmission equipment and of manuals, the analysis of individual activities and the structure of tasks, the impact of characteristics of work and personnel organization, and of particular working modalities such as the physical environment.

5. Next steps

Organize and report the results of the study.

6. Relation with other projects

The study is related to projects RS 70 (completed September, 1973) and SR 36 (in progress).

7. Reference documents

Quarterly reports (in German) in the series "IRS-Forschungsberichte":

Report period covering Jan.-Mar. 1975
Report period covering Apr.-Jun. 1975
Report period covering Jul.-Sep. 1975

8. Degree of availability

The "IRS-Forschungsberichte" can be obtained from the organization issuing these research reports.

Classification: 15.2		
Title 1 (Original Language): Entwicklung und Aufbau eines Ausbildungssystems im Medienverbund zur Intensivierung der Schulung und Ertüchtigung von Betriebspersonal von Kernkraftwerken (RS 152 - II.5.2., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Frankfurt
Title 2 (english): Development of a Training System for the Staff of Reactor Plants		Project Leader: H. Martin
Initiated (Date): 1. 1. 1975	Completed (Date): 1. 12. 1976	
Status: Continuing	Last Updating (Date): 31. 12. 1975	

General Aim and Particular Objectives

Information systems will be investigated which are suitable for training the staff of operating reactors. Criteria for the choice of an optimal schooling program are to be developed, considering several information carriers and media. An example of a training program will be produced and tested.

Experimental Facilities and Research Program

Work is concentrated on the following program:

a) Investigation of the applicability of various information systems

b) Development of a suitable combination of several media

c) Production criteria for schooling programs

d) Build-up of training programs for the staff

e) Production of special training examples

f) Documentation

Project Status/Progress to Date

Information about professional hardware: sound film automates, daylight-projectors, overhead-projectors, magnetic tape recorders, audiovisional equipment.
Rules were worked out for the improvement of animation, using polarization materials of various structure and raster. The effect

of recognition of different letters (shape, size) was studied.

Trick-filming by double-negative exposure and with the help of polarization foils was improved technically. For comparison several automatic projectors, combined with tape recorders were tested. A multiplan-trick technology, which operates on various photographic levels, was investigated, based on turning trick phases.

Project Status/Essential Results

The study of prospect material has shown, that only few apparates are suitable for group schooling purposes. Specific informations for power plant staff were not available on overhead foils.
Trick-filming with the help of polarization foils reduced the animation time about 30 - 40 %. The photografic technique has to be improved (accurate cover of negatives).

First tests have shown that magnetic recording is suitable for representing flow and turbulent effects, using polarization filters and foils.

Next Steps

Diapositive-projectors with higher power and tape recorders with better sound quality will be compared. The multiplan-photografic technique will be developed for practical use. Synchronization problems will be cleared up.

Relation with Other Projects

No relation with other Projects.

Reference Documents/Degree of Availability

No reports available.

Classification 15.2

Title 1	Country
Specificatie van eisen voor bedienend personeel in kern(energie) centrales.	Nederland **Sponsor** Ministry of Social Affairs **Organization** Reactor Centrum Nederland
Title 2	**Projectleader**
Specification of requirements for operating personnel at nuclear (power)stations.	F.M. de Meulemeester

Initiated August 1974 Completed July 1975 (intended)

Status Progressing Last updating n.a.

1. General aim

To advise the authorities on the training requirements for operating personnel at nuclear power stations.

2. Particular objectives

The programme will consist of:

- an inventory and evaluation of requirements imposed on the operating personnel bij the Dutch utilities and the requirements prescribed by the authorities in other coutries,
- preparing a report to the Dutch authorities on the basis of the results obtained from the forementionned study.

3. Experimental facilities)
4. Project status)
5. Next steps) not applicable
6. Relation with other projects)
7. Reference documents)

8. Degree of availibility

The reports will be written in the Dutch language. They will be submitted to the Ministry of Social Affairs, Postbus 69, Voorburg.

16. ENVIRONMENTAL PROTECTION

812

Classification: 16.2

Title 1 (Original Language): Stereobildübertragungssystem (RS 113 - I.2.3., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Forschungslab.Kleinwächter, Lörrach
Title 2 (english): Stereo-Television System	Project Leader: Prof. Dr. Kleinwächter
Initiated (Date): ıril 1974 — Completed (Date): June 1975 Status: finished — Last Updating (Date): June 1975	

General Aim

The aim of this investigation was constructing a three dimensional picture transmission system for observation tasks in radioactive contaminated surroundings. Television equipment can give spatial view to man, attribute of remotely controlled saving and repairing operations - especially in telemanipulator applications.

Particular Objectives

The objective of the project's first step was the conception of an elemantary stereo-tv-system for demonstrating that you can effect a spatial impression of transmitted movable pictures in man's mind. The principle is to transmit the signals of two cameras that are arrenged in man's eye distance beside each other. These pictures are reproduced in two different colours on a colour monitor. The pictures are separated by coloured spectacles. Subsequentely, stage 2 and later on stage 3 provided the progressive improvement of a fadeless colour stereo-tv-system.

Project Status

Progress to Date / Essential Results

Several stereo-tv-systems have been realized. After having completed all works on the complementary colour system, the elements for a chopper system were selected. The most suitable material is ferroelectrical ceramic because of its short rise and fall times. These elements are still in the experimental status and some probes arrived with Klera at the end of the project time, so that not all works could be completed in time. All electronical units were prepared.

To reach the aim explained above - especially in colour - a twoway-system with colour-tv-equipment was planned and realized. The camera head consisting of two colour-tv-cameras was optimized, all degrees of freedom are remotely controlled. These degrees are focus, zoom, convergence - pitch and azimuth-axis of the camera head. The aperture is controlled automatically.

The complementary system of stage one as well as the colour-tv-system of stage two and three give a very good spatial impression to the viewer if the cameras are optimally adjusted.

Next steps

The chopper system should be completed. A prolongation of the project was proposed at the end of stage 3.

Relation with other Projects

The development is done for the KTH of the GfK-Karlsruhe and coordinated with RS 21 (Synchronous Telemanipulator System).

Reference Documents

Quarterly reports in the IRS-Forschungsberichte (German).

Report period	April 1974 - June 1974	V 74/2
	July 1974 - Sept. 1974	V 74/3
	Oct. 1974 - Dec. 1974	V 74/4
	Jan. 1975 - March 1975	V 75/1
	April 1975 - June 1975	V 75/2
Annual report (Engl.)	April 1974 - Dec. 1974	IRS-F-24

Degree of Availability

Request necessary.

Classification: 16.2	
Title 1 (Original Language): Ferngesteuerte Arbeitsgeräte und mobile Systeme zur Schadenserfassung (PNS 4422 - I.2.3., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Remotely Controlled Working Gear and Mobile Systems for Damage Assessment	**Project Leader:** G.W. Köhler
Initiated (Date): 1971 / **Completed (Date):** December 1975	
Status: finished / **Last Updating (Date):** December 1975	

General aim

For the damage assessment after emergency cases in nuclear facilities it can be necessary to dispose remotely controlled mobile systems with manipuators.
The development and improvement of those systems is the aim of this project.

Progress to date and essential Results

The chassis of the lightweight MF3 manipulator vehicle for the Nuclear Emergency Brigade was completely assembled and subjected to works trial runs and acceptance tests.
All funktions of the MF3 chassis with its variable geometry chassis were verified in accordance with specifications.
Two electric MasterSlave EMSM II manipulators and the action control system for the whole MF3 system including command transmission, transmission of information and power supply were subjects of a tendering procedure after the end of the project design phase and following completion of the design. Fixed price bids have now been submitted by industries on the EMSM II manipulators and the action control system.
The possibilities of employing "MF3" have been studied in depth and covered in an internal report.

The Expert Committee on "Equipment for Emergencies and for Removing the Consequences of Incidents" of the Federal Ministry of Research and Technology has not been able to recommend the application for funding the development projects of electric MasterSlave manipulators, "EMSM II" and of the action control system for "MF3". This is due to the responsibilities for the Nuclear Emergency Brigade, which presently are in need of clarification.
The additions for test rigs to the electric EMSM I MasterSlave prototype manipulator have been finished.
The "EMSM" manipulator has been tested in detail. In the light of the experience accumulated, the load carrying capability and the cooling system were improved upon and the elasticity of the system was reduced.

Next Steps

The "EMSM II" project and the action control system will again be submitted to the responsible body of experts as soon as the responsibility for the Nuclear Emergency Brigade has bee clarified.

Reference Documents

"Manipulator vehicle system MF2 and its possibilities of application", Kerntechnik (1975) No.12 (german and english)

Report KFK 1859 (1973) p. 215 (german)
Report KFK 1908 (1973) p. 235 (german)
Report KFK 2050 (1974) p. 268 (german)
Report KFK 2130 (1974) p. 336 (german)
Report KFK 2195 (1975) p. 446 (german)
Report KFK 2262 (1975) p. (german)

Semiannual reports in the series IRS-Forschungsberichte
Report period Jan.-June 1974 IRS-F-21 (german)
July-Dec. 1974 IRS-F-23 (german)
Jan.-June 1975 IRS-F-26 (german)

Degree of Availability

The distribution of the KFK-reports is restricted.

17. NUCLEAR ACCIDENT RECOVERY AND DECOMMISSIONING

Classification: 17.1	
Title 1 (Original Language): Entwicklung von Dekontaminationsverfahren (PNS 4411 - I.2.4., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Development of Decontamination	**Project Leader:** T. Dippel S. Kunze D. Hentschel
Initiated (Date): ıuary 1973 — **Completed (Date):** December 1976 **Status:** continuing — **Last Updating (Date):** December 1975	

1. General aim

Development of methods and technics for the treatment and removal of surface contaminations by fission products, actinides and activated corrosion products.

2. Particular objectives

- classification of paints and floorings used in nuclear facilities with respect to their decontamination behaviour depending from the composition and fabrication parameters;
  the development of a pastetype cleanser
- the testing of molten salts as a decontaminant for metall surfaces.

3. Experimental facilities and program

3.1 Laboratories are in operation with the equipment necessary to handle radioactive materials. To some extend lab scale results can be used and checked with a technical installation.

3.2 This work is part of the R + D program of the Nuclear Safety Project: confinement and elimination of the consequences of incidents in nuclear facilities.

4. Project status

4.1 Progress to date

Experiments and classification of flooring materials and paints have been finished. The cleansing pastes and molten salts are tested in laboratory experiments.

4.2. Essential results

Floorings

PVC Floorings, fabricated by mixing of the basic components, showed no relation between content of fillers and decontamination results. Decontamination results are partly poorer, if the flooring contains a high concentration of the filler, especially if the latter consists mainly of hydrophilic materials. The coloring of the floorings seems to have no influence on the decontamination.

Rubber Floorings, fabricated by chemical reactions between polymeres, vulcanization materials and fillers, show decontamination results depending definitely from the proper choice of the filler. Flooring types, containing lampblack, graphite, kaoline, barium sulfate and titanium oxide are easy to decontaminate. Increasing contents of hydrophilic filler cause a fall off in the decontamination results.

Clearsing Pastes

The decontamination effectiveness and the homogenity of clea sing pastes based an hydrochloric acid, nitric acid, titanium oxide and polyethelene powders is strongly dependend on the content of hydrochloric acid. Reduction of the content of this component to less than 2 w/O remains the effectivness unchanged only if the titanium oxide-polyethelene powder mixture is substituted by a high density, highly surface active powder material. This type of paste containing no hydrochloric acid shows nearly the same decontamination effectivness as standard pickling pastes containing about 30% hydrochlorid acid.

Molten Salts

Properly prepared salt powder turn out to be easily and sucessfully aplied to metall surfaces by a flame spray technique. The thin layer of molten salts is a very effective decontaminant also in application to samples contaminated in the primary loop of a PWR.

5. Next steps

Equipment for paste and molten salts coating will be developped and constructed. These tools have to work reliably, automatically and under remote controll.

6. Reference documents

Report KFK 2130 (1975) german and english abstracts
Report KFK 2195 (1975) german and english abstracts

7. Degree of availability

The distribution of the KFK-reports is restricted.

Classification: 17.3	
Title 1 (Original Language): Entwicklung von Methoden und Verfahren zur Stillegung und Endbeseitigung nuklearer Anlagen (PNS 4421 - I.2.4., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Development of Methods and Procedures for the Decommissioning and Final Disposal of Nuclear Facilities	**Project Leader:** G.W. Köhler
Initiated (Date): nuary 1974 **Completed (Date):** 1978 **Status:** continuing **Last Updating (Date):** December 1975	

General aim

The aim of this project is the development of methods to enable the decommissioning and final disposal of nuclear plants either at the end of working life or after an accident.
This providence is required in the comprehensive frame of a systematic use of nuclear facilities.

Progress to date

Because of the greater urgency involved, studies were begun on the project design of a plant for decontamination and processing fit for ultimate storage of radioactive, heavy reactor components (conditioning plant); work on decommissioning of nuclear facilities and especial on clearance of a reactor core damaged by a major incident has been suspended for the time being.
The most important sub-areas investigated were these:

- Operating experience in large hot cell facilities in dismantling work and treatment of radioactive wastes.
- Analysis of the process flow diagrams of six treatment techniques applicable to heavy radioactive reactor components and determination of most important process steps and their sequences.
- Estimation of the importance and the probability of the different kinds of treatment.

- Basic design of the plant and design of the cells to fit different purposes.
- Introduction of a reactor component without causing any contamination.
- Methods of separation and disassembly for different materials and combinations of materials and wall thicknesses and dimensions of components.
- Packing and fixing wastes and bagging them out without causing any contamination.
- Remotely controlled transport and handling, respectively, of reactor components, tools, waste parts and waste containers in the plant.
- Visibility in remotely controlled operations.
- Maintenance of mechanical equipment in the disassembly cell.
- Building design features of a facility built of concrete bricks and prefabricated components so as to be capable of dismantling in most of its parts.

Essential Results

Design drafts of the medium active part of a conditioning plant have been elaborated for a central facility, one facility each to be built up in the close vicinity, e.g., of a nuclear power station with vertical and horizontal loading, and one standby solution. The underlying design component covers all the components of a nuclear power station ot the category of Biblis A.
A description of the building design features and the sequences of operational steps and the data and cost estimates are included in PNS Report 61/75.
In view of the lengths of the cuts and the sometimes excessive material thicknesses, the disassembly of large components can be carried out within reasonable periods of time and under both practically and economically tolerable conditions only by thermal cutting techniques. Depending upon the material or the combinations of materials concerned, autogenous flame cutting or plasma fusion cutting or a combination of the two techniques must be applied in preference to any of the numerous other techniques. The results of the studies carried out so far are summarized in PNS Working Report 60/75.

Next Steps

For a conditioning plant to be set up in the immediate vicinity of the nuclear power station, the part designed for the treatment of low level components and the whole facility, including the auxiliary systems, will be designed and the problem of transferring a large component from the reactor building into the conditioning plant will be clarified. The suitability of different plant concepts should be investigated with respect to the safety of the personnel. Because of the results to be expected from parallel studies carried out elsewhere (e.g., on transport, activities, amounts of material), the design drafts must perhaps be modified.

Underwater flame cutting techniques will be thoroughly investigated with respect to their suitability, capability and limits of applicability and also with a view to the further developments necessary.

Reference Documents

Report KFK 2050 (1973) p. 267 (german)
Report KFK 2130 (1974) p. 335 (german)
Report KFK 2195 (1975) p. 437 (german)
Report KFK 2262 (1975) p. (german)
Semiannual reports in the series IRS-Forschungsberichte
Report period Jan.-June 1974 IRS-F-21 (german)
July-Dec. 1974 IRS-F-23 (german)
Jan.-June 1975 IRS-F-26 (german)

Degree of Availability

The distribution of the KFK-reports is restricted.

18. FUEL CYCLE

Classification: 18	
Title 1 (Original Language): Kritikalitäts-Studien (ATT 085 A - I.4.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (english): Criticality Studies	**Project Leader:** Prof. Dr. Birkhofer W. Thomas
Initiated (Date): **Completed (Date):** **Status:** continuing **Last Updating (Date):** December 1975	

1. General Aim

Problems of criticality safety are investigated and solved under this topic for the nuclear fuel cycle. Safety criteria, critical and safe parameters will be established for fissile material in fuel manufacturing and reprocessing, storage and transportation. Risks and consequences of criticality accidents in processing nuclear fuel outside reactors will be evaluated.

2. Particular Objectives

Special problems of neutron interaction and isolation will be treated. The influence of concrete structures of hot cells on neutron reflection will be considered. Problems of homogeneous and heterogeneous neutron poisoning are investigated especially for reprocessing facilities. Assessments of radiological consequences of criticality accidents or other severe accidents in plutonium and uranium fabrication facilities are included.

3. Experimental Facilities and Program

No experimental work.

4. Project Status

4.1 Progress to Date

During last year criticality data for uranium and plutonium nitrate solutions with homogeneous poison, including gadolinium have been computed. A study of plutonium accumulations in vessels containing boron raschig rings has been performed. These calculations are significant for nuclear safety in reprocessing facilities. A series of calculations investigated the influence of various concrete reflectors on criticality of UO_2 containing slabs. The Alize-core has been calculated using various computer codes as a benchmark problem for heterogeneous plate poisoning.

In the field of risk analysis in the nuclear fuel cycle two studies have been performed for the air transport of plutonium and the impact of UO_2 fuel manufacturing to the environment. A computer code has been written for the calculation of radiological consequences of criticality accidents (computer code RADCA).

4.2 Essential Results

A new supplement has been issued to the "Criticality Handbook" containing data for UO_2F_2, ADU and AUC. Data for homogeneous gadolinium poisoning of U-Pu-solutions have been collected in a report. For the computer code RADCA an input description is available.

5. Next Steps

Our "Criticality Handbook" will be completed by one or two supplements dealing with the following topics: neutron interaction, reflector effects, new data for plutonium carbide, U-Pu-mixtures and homogeneous poisoning of ^{233}U-solutions. A study of heterogeneous poison will be performed especially for storage of spent fuel elements and the first extraction cycle of PUREX reprocessing scheme.

6. Relation with Other Projects

Relations exist with design studies for a new reprocessing plant performed at INR, Karlsruhe. Similar work is under way in the UK, France and USA, as documented in:

- Handbook of Criticality Data, J.H. Chalmers et al., AHSB 1965/7
- Guide de Criticité, CEA, CEA-R-3114, 1967
- Criticality Handbook Vol. I-III, R.D. Carter et al., ARH-600, 1968/72.

7. Reference Documents

/1/
W. Heinicke, W. Thomas, W.J. Weber
Handbuch zur Kritikalität
LRA Garching, Januar 1975, 6. Teillieferung

/2/
W. Heinicke
Nachrechnung der kritischen Experimente der ALIZE-Versuchsanordnung.
Interner Bericht
MRR-I-42, Mai 1975

/3/
W. Thomas
Gefährdung aus der Verarbeitung von niedrig angereichertem Uran zu Leichtwasserreaktor-Brennelementbündeln. Interner Bericht
MRR-I-49, Juni 1975

/4/
W. Thomas
RADCA - Ein Rechenprogramm zur Berechnung der radiologischen Auswirkung von Kritikalitätsunfällen. Programmbeschreibung
MRR-P-18, Oktober 1975

/5/
W. Thomas
Kritikalitätssicherheit durch Gadoliniumvergiftung und Konzentrationsbeschränkung für Spaltstofflösungen bei der Wiederaufarbeitung von Kernbrennstoffen
MRR 142, Februar 1975

/6/
W. Thomas
Kritikalitätsprobleme durch Spaltstoffablagerung in Behältern mit Borglas-Raschigringen
ATW 7/8, Juli/August 1975, S. 361-363

8. Degree of Availability

Documents are available through
Laboratorium für Reaktorregelung und Anlagensicherung
D-8046 Garching
Federal Republic of Germany

Reports MRR-I and MRR-V are confidential and therefore normally not available.

PROJECT TITLE : Evaluation of chemical and physical properties of active solidified wastes.	CLASSIFICATION 18
SPONSORING COUNTRY : ITALY	ORGANISATION : CNEN
DATE INITIATED : 1975 DATE COMPLETED : 1977	PROJECT LEADER : F. GERA

Description : The purpose of the project is to control the main chemical and physical properties of active solidified wastes to reduce possible long-term environmental impacts.

Radioactive solid wastes of nuclear reactors are fixed in beton or plastic compounds. On the resulting solid wastes many tests will be carried out to evaluate the following characteristics: leaching, thermal stability, radiation damage, compression, etc......

PROJECT TITLE : Research and development on fuel casks	CLASSIFICATION 18
SPONSORING COUNTRY : ITALY	ORGANISATION : CNEN-ENEL-AGIP NUCLEARE - FIAT NUCLEARE - NUOVO PIGNONE - UNIV. PISA
DATE INITIATED : 1974 (present phase) DATE COMPLETED : in progress	PROJECT LEADER : CNEN-Divisione Ric.Sicurezza

Description : Preliminary design, experimental researches, calculations (shielding, mechanics, heat transfer), model elaboration, tests, optimization studies on type of casks and means of transport related to power plants and reprocessing facilities, for the development of spent fuel shipping casks.

Related project

18 (Pisa University)

PROJECT TITLE : Safety Problems in Design of Packagings and Transportation of Radioactive Materials.	CLASSIFICATION 18
SPONSORING COUNTRY : Italy	ORGANISATION : Pisa - University
DATE INITIATED : 1968 DATE COMPLETED : 1980	PROJECT LEADER : G. Forasassi

Description :

Present research activity refers to determination of energy absorption characteristics of steel structures and tests on models and components to support the design of spent fuel shipping casks.

Facilities in Scalbatraio Center of University of Pisa can be utilized for experimental studies as well as to carry on IAEA standard test series on packagings for radioactive materials.

The following equipments are operable:

- Drop test tower for casks and models up to 2000 Kg of weight.
- Thermal test station (open fire).
- Spray test facilities.
- Vessel for hydraulic and tightness tests.
- Instrumentation for measures of acceleration, displacements and temperatures.

Program sponsored by CNEN.

REFERENCES

1. G. ELETTI, R. ZAMBRINI, G. FORASASSI, P. SACCOMANNO
   Sicurezza del trasporto del materiale radioattivo.
   Atti del convegno sulla sicurezza negli Impianti Nucleari
   Pisa 21 - 26 Settembre 1970

2. P. CITTI, G. FORASASSI, B. GUERRINI
   Rilievi su prototipi di un contenitore per il trasporto di
   materiale radioattivo.
   Atti del 1° Convegno AIAS - Palermo 1972.

3. G. ELETTI, G. FORASASSI
   Italian testing station following IAEA Regulation; CF6 a
   fissile Class II Type B packaging.
   4° International Symposium on packaging and transportation
   of Radioactive materials.
   22 - 27 Sept. 1974.

Classification 18	
Title 1 Ontwikkeling van een computercode voor het schrijven van een éénduidige en optimale herladingsprocedure	**Country** The Netherlands **Organization** KEMA
Title 2 Development of a computer code which writes an unambiguous and optimum reload procedure	**Projectleader** K.P. Termaat

1. General aim

The programme "Reload" will provide the operator of a nuclear power plant with a stepwise written optimum reload procedure. The programme is based on octant symmetric reload patterns, however non symmetric fuel element movements can be included.

2. Particular objectives

The programme is developed to be applied in the Dodewaard nuclear power plant. The objective is to minimize the number of refuelling steps and the quantity of time to reload the core. The programme will prevent errors which can possibly be made by handwriting the elaborous procedure, especially when a large number of reload elements, shuffle elements, dummy elements and inspection elements are involved in one reload scheme.

3. Experimental facilities

Not applicable.

4. Project status

The programme is in the development status.

5. Next steps

Complete development and test with previous handwritten reload procedures.

6. Relation with other projects

Physics reload scheme, fuel inspection programme, fuel test programme.

7. Reference documents

To be made.

8. Degree of availability

Through the organization KEMA.

19. ECONOMICS OF SAFETY

20. OTHER TOPICS

Classification: 20

Title 1 (Original Language): HDR - Sicherheitsprogramm (RS o123 A - III, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): HDR Safety Program	Project Leader: W.Müller-Dietsche
Initiated (Date): ..4.1974 / Completed (Date): 1981	
Status: continuing / Last Updating (Date): December 1975	

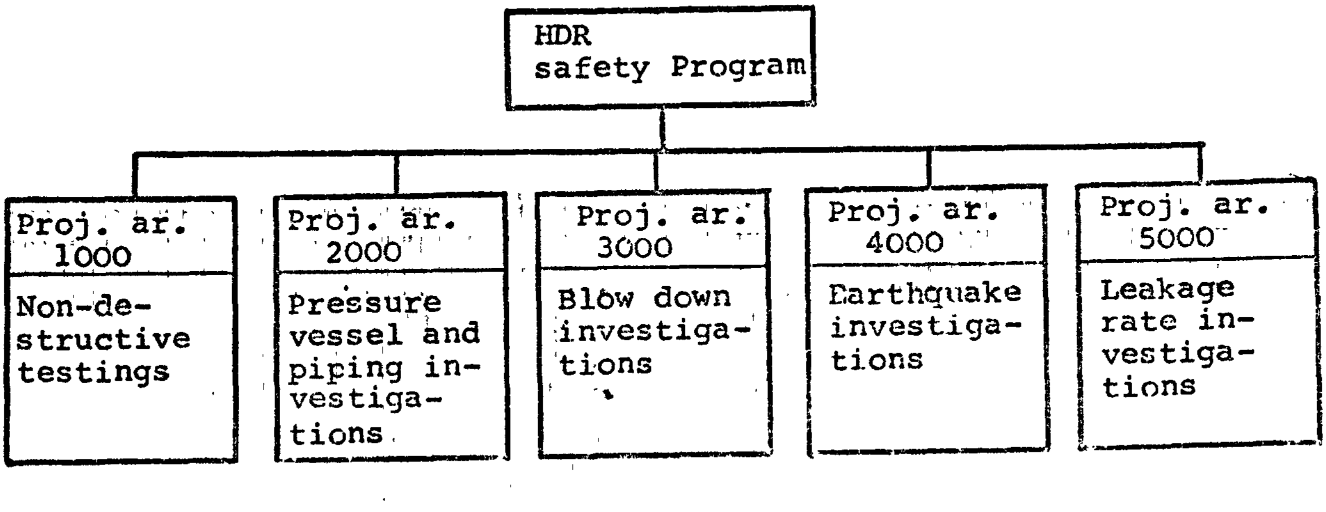
1. General aim

The safety program which will be performed at the deco missioned HDR-plant near Frankfurt/Germeny, serves for clarification of important problems within the framework of the light water reactor safety research program of the Federal Minister of Research and Technology (BMFT).

The unrestricted availability of this reactor for investigations concerning safety technology offers the rare possibility to check on a complete plant, the validity of engineering theories , applied design methods and the results achieved in current research projects up to the actual limits of stress bearing capacity of specific reactor components.

2. Particular objectives

The HDR safety program is devided into five project-areas:



These five project-areas were determined due to the main objectives presently persued under the Reactor Safety Research Program of the Ministry after harmonization with the involved authorities.

3. Experimental facilities and program

3.1 Experimental facilities

The HDR with 100 MWth power was built to demonstrate the possibility of nuclear superheating. Typewise, it largely corresponds to the German water boiling reactor designed around 1967/1968. The main data can be seen from Table 1. Particular importance is attached to a full pressure containment and the reactor pressure vessel. No nuclear operation is envisaged for the experiments. The experimental conditions will be obtained with an electrically heated boiler.

3.2 Program

3.2.1 Project-area 1000: Non-destructive Testings

Non destructive Testings serve for controlling the reactor pressure vessel and the piping under pressure tests and blow down-tests and for testing in-service inspection systems under development. The following objectives will be investigated.

- Assesment of record of the initial conditions
- Detection of failure generation and failure propagation
- Testing of inspection methods; Proof of the suitability of new test methods.
- Comperative evaluation of non-destructive testing systes such as: Ultrasonic impulse testing, magnetic particle testing, penetration testing, eddy current method, ultrasonic scattering, acoustic emission , acoustic holography , radiography

3.2.2 Project-area 2000: Pressure vessel and piping investigations

Investigations of pressure vessels and piping should be a further step of intensifying and securing the basic knowledge of their safe design. The most important aspect is a study of the effective safety margin above the design ratings of components. The following objectives are pursued:

- material properties at the beginning of the experiments
- component behavior under special loading conditions p.e.: operating conditions, blow down, thermoshock, earthquake

- component behavior under specific weakening - fabricational and operational defects and crack formations - to determine boundary strengths or critical defects
- reliability of the employed methods of calculation.

3.2.3 Project-area 3000: Blow down investigations

In the blow down investigation it is possible to measure radial stresses of reactor pressure vessel internals, containment structures and full scale valves by means of reactor typical impulse and mass-flow exitations in their realistic dimensions. The investigations comprise the following objectives:

- stress behavior of the reactor pressure vessel internals an containment structures
- behavior of full scale valves under mass-flow exitation
- testing and further development of the various fluid and structural dynamic computercodes.

3.2.4 Project-area 4000: Earthquake investigations

Vibration-investigations serve to improve and correct the basic knowledge about the safe design of building structures and pipe-systems with respect to the effects of earthquake. This implies the following objectives:

- verification and optimization of available analytic methods
- influence of material and soil characteristics and desigtn conditions
- testing of experimental techniques
- advancement of analytical procedures for the non-linear range.

3.2.5 Project-area 5000: Leakage rate investigations

These investigations serve to determine the previously unsettled parameters with the aim to standardize the leakage rate procedure and the in-service investigations of nuclear power stations. This involves the following objectives:

- leakage behavior of a cold plant
- leakage behavior of a plant at operating temperature
- verification and optimization of existing analytic procedures.

4. Project Status

4.1 Progress to Date

Central Preparation of Investigations

The total HDR-safety program was drawn up on the basis of five detail specifications to the project areas. The test loop, a central data acquisition system and a documentation system were conceived.
The conversion was planned of the HDR facility allowing accommodation of the experiments. The preparations to this effect included dismantling above all of radoactive plant components which are no longer required.

Project area 1000: Non-destructive Testing

Non destructive tests were performed at the pressure vessel in order to determine exactly the actual state of the pressure vessel by

- ultrasonic impulse testings, ultrasonic manual-tandem-testings, ultrasonic testings with focusing heads.

The measurements yielded some small defects in the base material and the circumferential welding joints, which, howewer, are so low that the safety is not endangered by the planned load tests.

Project area 2000: Pressure Vessel and Piping Investigations

Specimens are taken at the wall of the pressure vessel and at the piping system and used to determine in mecano-technological and metallographic investigations the characteristic material properties, strength, ductility, structural state. In the earthquake tests performed at low excitation (cf. project area 400) extensive strain measurements were made at the reactor pressure vessel, the reactor piping system and the containment. The stresses occurring at different vibration excitations were determined. A new theoretical stress analyses for the pressure vessel was completed which relies on the finite element method. It will serve as a basis of the comparison between measurement and calcualtion.

Project area 3000: Blowdown Investigations

Preliminary calculations were made relative to the loads and stresses of the containment and of the pressure vessel internals and a first coarse theoretical analysis was carried out of the fluid dynamic processes taking place during blowdown. The test components, especially the pressure vessel internals, and the extensive measurement technology for testing have been conceived. Preliminary experiments started to test the measurement technology.

Project area 4000: Earthquake Investigations
The experiments at the HDR facility were completed in September 1975. They had been performed at low vibrational excitation and on the basis of preliminary calculations. The reactor building and single plant components were excited systematically at different frequencies and intensities by shaker, snap back and explosion, and the exited vibrations were measured by means of accelerometers. Specific resonance ranges were investigated. The evaluation of results will be terminated in January 1976 and the results will form the basis of comparison with comprehensive calculations planned for 1976.

Project area 5000: Leakage Rate Investigations
The prgram for the investigations was set up and preparations started of plant and measurement technology for the first measurements in the cold facility (to be performed approximately in April 1976).

4.2 Essential Results
The total test program was completed in December 1975. The conception of the test facility and of the measurement technology has been largely worked out and preparations are in progress.

The new theoretical stress analysis for the reactor pressure vessel, the results of non-destructive tests and the knowledge of the exact state of the component-material provide an important basis for the implementation of the experiments.

The earthquake tests allowed evaluation at low excitation of the searched vibrational behavior (time history and local distribution of frequency and amplitudes) of the facility as a whole and of specific components (pressure vessel, pipings, containment). The strain measurements performed in parallel yielded an insight into the stresses occurring during vibrations.

5. Next Steps

Installing on electrically heated vessel the test loop will be completed allowing simulation of operating conditions and for realization of the blowdown tests in 1976.

Main points of further investigations include

- detailed experimental stress analysis (project area 2000) in the "cold" pressure test (143 bar, 60° C) with acoustic emission measurements (project area 1000)
- leak rate investigations at the containment with the facility in the "cold" state (project area 5000)
- automatic ultrasonic examination of the pressure vessel (project area 1000)
- calculations of the vibrational behavior in the scope of earthquake examinations (project area 4000)
- preliminary calculations and preparation of blowdown tests (project area 3000)
- laboratory experiments in the test rig on the failure of piping sections (project area 2000).

6. Relation with other projects

1. Investigation into the Phenomena Involved in the Depressurisation of Water-Cooled Reactors — RS 16/2

2. Investigation of the Phenomena Occuring within a Multicomparment Containment after Rupture of the Primary Cooling Circuit in Water-Cooled Reactors. — RS 50

3. Reaction - and Impingement Forces on Components and Structures Caused by a Two Phase Jet Discharged at primary pipe breaks. — RS 93

4. Joint Reactor Safety Experiments in the Power Station of Marviken, Sweden. RS 33

5. Ultrasonic Pulse-Echo Spetroscopy in Ultrasonic NDT. RS 54

6. Investigations on a Continuously Operating System for Crack Growth Surveillance in Pressure Vessels, Part IV: Further Development of Accoustic Emission with Regard to the Application at the Reactor. RS 31/3

Nondestructive Inservice Inspection for Reactor Pressure Vessels with Eddy Current Methods. RS 89

7. Development of Non-Destructive Testing Methods for In-Service Inspections of Reactor Pressure Vessel. RS 27/2

7. Reference documents

- Paper for: IEA - SLT - ERD - Expert Group on Reactor Safety "Safety Investigations in the decommisioned HDR-Plant" April 1975 (english)
- Quarterly reports: In the series IRS-Research Reports (german) Reports IRS - F - 21, 23, 25 - 27
- Annual report: IRS - F - 24 (english)

8. Degree of availability

Unrestricted distribution

Table 1 — Main Plant Data

Safety Containment:	Dimensions	
Inside height	m	60
Inside diameter	m	20
Test pressure	atm.g.	7,12
Design pressure: Overpressure	atm.g.	5,60
Underpressure	mm water	350
Design temperature	°C	155
Wall thickness of cylindrical part	mm	30
Material spezification of steel cladding	–	FB 50 S fine grained steel
Free containment volume	m^3	10,853

Pressure Vessel:	Dimensions	RDB	HDU	SDU	UK
Number	–	1	1	1	2
Inside diameter	mm	2960	1765	1978/1508	700
Overall height	mm	12 700	14 110	10 920	4365
Wall thickness (average)	mm	112	45	61/46	28
Design pressure	atm.g.	110	110	110	110
Design temperature	°C	360	550	400	320
Material	–	23 NiMoCr 36	BHW 38	BHW 38	BHW 39 S
Flow: primary	t/h	–	170	170	170
secondary	t/h	–	130	130	130

Water Circuits:	Dimensions	Primary circuit between RDB and HDU	Primary circuit between HDU and SDU	Primary circuit between SDU and RDB	Secondary circuit up to reducer station	Recirculation loop
Design pressure	atm.g.	110	110	112	110	112
Design temperature	°C	550	400	320	400	320
Material	–	No. 4961	No. 4550	No. 4550	No. 7335	No. 4550
Inside diameter	mm	250 – 300	250	200	300 – 350	350 – 450

Notes:

Reactor pressure vessel = RDB
Superheated steam converter = HDU
Saturated steam converter = SDU
Subcooler = UK

Classification: 20.		
Title 1 (Original Language): Auswirkungen von Kühltürmen großer Kernkraftwerke auf ihre Umgebung (PNS 4152 - III., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Environmental Impact of Cooling Towers of Large Nuclear Power Plants		Project Leader: K. Nester
Initiated (Date): anuary 1974	Completed (Date): December 1976	
Status: continuing	Last Updating (Date): December 1975	

1. General aim

The aim is pursued under this task to study the impact of emissions from large cooling towers on the environment.

2. Particular objectives

A three-dimensional simulation model should be developed which allows calculation of the rise and of the dispersion of heat, humidity and droplet emissions from cooling towers as a function of ambient conditions.

3. Experiments

Experiments have not been performed (compare 6).

4. Project status

The already available three-dimensional model has been extended so that also the vortex-pair structure of the cooling tower plume can be simulated now. The comparison between the calculations and the measurements (compare 6), which can be made but qualitatively for the time being now gives a good agreement as to the profile and structure of temperature, humidity and vertical velocity distribution in the plume.

5. Next steps

It is intended to adapt to the real situation the program parameters by test calculations based on the conditions during the measurements and by comparison with the respective measured data. For this adaptation of the computer program the detailed data are supposed to be available in 1976.

6. Relation to other projects

The measurements in cooling tower plumes required for testing are carried out by DFVLR and Deutscher Wetterdienst within the framework of the HHT Project of the Jülich Nuclear Research Establishment.

7. Reference documents

Report KFK 2195 (1975) p 141 (German)
Report KFK 2155 (1975) p 118 (German)
Semiannual report IRS-F-26, 1975 (German)

8. Degree of availability

Unrestricted distribution

Classification: 2o	
Title 1 (Original Language): Untersuchungen über den Wärmeaustausch Fluß/Atmosphäre am Rhein (PNS 4151-III, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Investigations of the Heat Exchange River/Atmosphere at the Rhine River	Project Leader: Dr.Schikarski
Initiated (Date): .972 Status: continuing Completed (Date): 1977 Last Updating (Date): December 1975	

1. General Aim

Evaporation, convection and heat radiation as well as conventional meteorological and hydrological datas are to be measured directly over the water surface. Available heat loss models are to be tested and advanced. The results will give improvements to temperature profile models and cooling strategies.

2. Particular Objectives

The three important heat fluxes and their variation with microclimatic situations shall be determined directly. Investigations on best regional heat loss models, using general available input data, are to be performed on the basis of the in-situ-data. By means of statistical analyses the most relevant situations, as far as waste heat pollution is concerned, shall be revealed and characterized.

3.1. Experimental Facilities

A floating meteorological mast with one single degree of freedom in vertical motion is installed in the Rhine river, 60 m off shore, 5 km downstream of the Philippsburg Nuclear Power Station now under construction. Probes are installed in water and in .8, 2 and 8 m over the water surface as well as 40 m above ground on shore (10m above the top of trees) at a second mast. The observed parameters are: temperature, humidity, wind speed, water surface roughness and oxygen content. A supersonic anemometer-thermometer combined with a miniature thermistorized psychrometer serves for direct on-line computation of evaporation, convection and turbulence, making use of the eddy-correlation method. All data input, linearisation and averaging (10 min. intervals) is done by a PDP 11 computer. Output is available on teletype, papertape punch or cassette recorder as well as on analogeous multichannel printers. The data gathering system is housed in a 3.50 x 3.50 m steel hut on shore.

3.2. Research Program

Microclimatical and hydrological data are to be gathered. The site shall be representative for the upper Rhine valley. The minimum period of time is regarded to be about one year before and one year during discharge from the power station. Results of eddy correlation computations are to be set into relation with different heat exchange calculations in order to find the best description of the effects of rather rapidly flowing water surfaces. Of specific interest is a good generalisation of the results to obtain a reliable instrument for doing better heat loss calculations for the whole river.

4. Project Status

4.1. Progress to Date

Since November 1975, the station has been working with a few interrupts. Installation activities are finished.

4.2. Essential Results

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5. Next Steps

Computer codes for data evaluation have to be programmed. Usual maintenance of the station has to be continued.

6. Relation with Other Projects

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7. Reference Documents

PNS-Halbjahresbericht 2/1974 , KFK 2130
PNS-Halbjahresbericht 1/1975 , KFK 2195

8. Degree of Availability

No restrictions, LA-GfK

Classification: 20	
Title 1 (Original Language): Programmentwicklung zur mehrdimensionalen Kontinuumsmechanik (RS 139 - III., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** Interatom, Bensberg
Title 2 (english): Code Development in Multi-Dimensional Continuum Mechanics	**Project Leader:** H. Banasch
Initiated (Date): 1.9.1974 — **Completed (Date):** 31.12.1976	
Status: continuing — **Last Updating (Date):** December 1975	

1. General aim

The aim of the project is the development of methods in the field of multi-dimensional continuum mechanics in order to describe

- shock and pressure wave propagation together with one and two-phase large scale fluid flow in a general geometry (compressible as well as non-compressible viscous hydrodynamics in an Eulerian coordinate system)
- and coupled with that the elastic-plastic deformation, penetration and failure of complex structures (in a Langrangian coordinate system).

2. Particular objectives

The experimental verification of all important aspects of the methods developed is a particular objective of the project:

- evaluation of available tests
- participation in the vessel explosion programme being performed at Euratom/Ispra (pre- and post-shat theoretical analysis)

3. Research programme

- literature survey of available methods for solving problems in continuum mechanics
- development of procedures for the hydrodynamics calculation in discrete Eulerian meshes

- description of the solid structures in a Lagrangian system (finite elements, thin-shell theory, multi-axial stress and strain conditions, general materials laws)
- coupling of the Eulerian and Lagrangian procedures for solving the integral hydrodynamic and structural dynamics problem
- development of a theory describing the high pressure gas bubble through an Eulerian coordinate system
- inclusion of energy dissipation due to friction (e.g. flow through perforated plates) into the description of hydrodynamics
- verification of the theoretical methods by analysis of appropriate tests.

4. Project status

Separate code routines have been written for the compressible hydrodynamics (application of the conservation equations for mass, momentum and energy for discrete meshes in a general Euler-Lagrange-formulation) and structural dynamics (theory of thin shells multi-axial elastic-plastic stress conditions are determined by the Prandtl-Reuß-law in combination with Miese's yield criterion). These routines have then been coupled and successfully tested in a working version of a code (KOELSCH) representing the geometric features of the SNR-3oo reactor vessel for stationary conditions. The separate routines have been checked out by various test cases.

5. Next steps

The working version of the established code will be tested for transient conditions and the verification will be performed by analysis of appropriate tests.

Classification: 20	
Title 1 (Original Language): Störfallanalyse von Natrium-Wasser-Reaktionen im Dampferzeuger unter Berücksichtigung der Bildung von Zwei-Phasen-Zwei-Komponenten-Gemischen (RS 140 - III., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** Interatom, Bensberg
Title 2 (english): Accident Analysis of Sodium-Water-Reactions in Steam-Generators and the Associated Generation of Two-Phase, Two-Component Mixtures	**Project Leader:** H. Banasch
Initiated (Date): .9.1974 **Completed (Date):** 30.9.1977 **Status:** continuing **Last Updating (Date):** December 1975	

1. Objectives

The objective of this work is to improve methods of the accident analysis of sodium-water-reactions due to water tube failures in steam-generators of sodium cooled breeder reactors. A closer inspection of basic experiments carried out by Interatom on sodium water reactions has led to the suspicion that the large hydrogen bubble that follows the sodium will be filled entirely with small sodium droplets.

Due to this phenomenon deviations have occured between computed and measured data of the sodium-water-reaction and subsequent events. The objective ot this work is to develop computer code POOL, version GT, in which the sodium is not replaced by pure hydrogen but by hydrogen filled with sodium droplets. This computer program should replace the older version G in which a computer model by Salmon has been used.

2. Basic work and results

In the first place, results of the following basic tests have been more closely inspected: ASB, step 1, straight tube steam generator, coiled tube steam generator model. The results of these tests have been reworked and compared with the new computer model. The computer model has been brought into a form that can be used as subroutine in other computer programs dealing with accident analysis.

Initially, the basic test series and ASB, step 1, have been dealt with by using computer program POOL, version G.
In the course of this work, computer program PARA III has been used in order to get some values for otimum parameters. When evaluating the calculations it could be shown that the computer model of Salmon is not in agreement with reality. It could be shown that the sodium droplets in the gas do not change speed as quickly as the gas itself. Therefore, all accelerations of the sodium droplets are accompanied by an additional flow resistance. At the same time, it should be considered that extremely large difference in speed between gas and sodium droplets eventually result in a further breakdown of these sodium droplets into smaller ones. (critical Webernumber = 6). A very intensive heat exchange will also occur between gas and sodium due to the large surface area of the sodium droplets. Hydrogen can also be expected to diffuse into the sodium and, therefore, gaseous hydrogen will be lost. Due to the high rate of change of this phenomenon, the surface area of the sodium can be expected to be the decisive quantity for the diffusion of the hydrogen gas. The above observations have led to the introduction of a point model to describe the phenomenon. The increase in pressure-loss due to the presence of sodium droplets in the hydrogen is correctly accounted for in this computer program. The approach results in ordinary, non-linear differential equations that can be solved using iterative procedures. A numerical difficulty has occured in that at some points three solutions were found instead of one. By introducing adequate logical elements it will be possible that correct solutions are selected.

The computer program POOL has been further developed to a stage in which the increase in pressure-loss, due to the sodium droplets can be correctly accounted for. The influences of the temperature changes and of the diffusion of hydrogen have not yet been introduced into the computer program.

The computer program PARA III is useful for the following reasons:

- The sodium mass in hydrogen can be determined by accounting for the separation of a film from the inside of the tube or the vessel.
- The thickness of the film changes with the sodium speed and decreases with decreasing speed.
- It will take some time to reach the eventual film thickness. Information on this subject in the open literature is scarce.
- When using computer program PARA III the velocity and the change in film thickness on the wall can be determined. At the same time the temperature of newly formed hydrogen can be controlled. It appears to be important, that small leaks are accompanied by much bigger sodium quantities in the hydrogen than larger leaks. This leads to differences in the overall behaviour.

3. Further Work

Computer program PARA III will be further refined. In computer program POOL the influence of the temperature changes of hydrogen and of the diffusion of hydrogen into the sodium will be introduced. It will be considered whether the generation of sodium hydrides should be introduced into the program. At the same time the results of basic test series with different boundary conditions should be more closely inspected and it should be checked, whether a close agreement between experiment and calculation has now been reached. Results obtained by using computer program PARA III should be reasonably close to experimental ones.

4. Reference documents

- Salmon, M.A. Mc Donald, I.S. Effects of tube leaks in sodium heated steam generators, NAA-SR-814o, Atomic International, Canoga Park, 15. April 1963

- K. Dumm, H. Mausbeck, W. Schnitker, Status of Sodium-Water-Reaction Test Work at INTERATOM, ANL-752o, Part I, 1969, P. 374-383

- W. Schnitker, The peculiarities of the reaction of sodium and water in vessels, Atomkernenergie, Bd. 24, 1974/75, P. 225 - 232.

Classification: 20	
Title 1 (Original Language): Entwicklung fernbedienter Ultraschallprüftechnik für Schnellbrutreaktoren (RS 143 - III., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** Interatom, Bensberg
Title 2 (english): Development of a Remote Controlled Ultrasonic Testing Device for Fast Breeder Reactors	**Project Leader:** H. Banasch
Initiated (Date): 1.9.1974 **Completed (Date):** 30.6.1976	
Status: continuing **Last Updating (Date):** December 1975	-

1. SCOPE

The scope of this task is the development of an ultrasonic testing method for austenitic welds as well as of remote controlled manipulators for testing instruments applied for out-side inspection of the reactor vessel.

For the development work of the ultrasonic testing heads and of the manipulators, the ambient conditions expected for the repeating inservice inspection of the reactor vessel of a sodium cooled fast breeder reactor plant have to be taken into consideration:

- Temperature of the reactor vessel wall and the ambient atmosphere
- γ-radiation level about 10^4 rad/h
- Nitrogen as a medium surrounding the reactor vessel
- Reactor vessel loaded with fuel.

2. PERFORMED WORK AND OBTAINED RESULTS

2.1 Construction and production of ultrasonic testing heads for austenitic weld joints

For the calculation of the testing heads the 60 mm thick weld was devided into 4 test zones of each 15 mm extension in depth and 15 mm width (Scanning pattern). The test zones were placed among one another. On the basis of the obtained calculation data an ultrasonic testing head for the test zones 1-3 was designed, manufactured and tested. The following demands were met:

- Partly damp out respectively fade out of simultaneously proceeding transverse waves occurring at SEL-testing heads.
- Highest sensitivity of the testing head at the middle of the test zone.
- Generation of a wall stabilized impuls.

These first SEL-angle-testing head-series used for longitudinal defect testing lead to satisfactory results with the "cold" testing. The design work of those heads for "hot" testing was started. For the 4th zone a testing head will be developped which will work differently than the testing heads with transversal waves used for the other test zones. After the concept for the transverse defect testing was found the trial fabrication was startet by the end of 1975. Because the problems with the coupling medium are not yet solved, the dry coupling will be developped further - some experiments took already place.

2.2 Design and construction of the manipulator for installation of the testing instruments

Altogether 12 inspection manholes with an inner diameter of 49o mm, above which the testing system is put into position, are distributed on the vessel periphery in order to cover the welds of the vessel wall at the periphery.

Working position is the annulus between reactor vessel and reactor guard vessel with a gap width of min. 28o mm. The testing installation will be moved to the test place by a rail-chain-system. The rail-chain-system renders the vertical testing motion possible. The horizontal motion as well as the necessary rotary motion for circular seam weld testing is provided by an extentable installation.

The design development of the test carriage with jib system for testing of welds within the cylindrical range is nearly completed. Difficulties are being encountered in the range of the spherical vessel bottom as well as for the testing of circular seam welds and stud welds for the nozzels.

Because the problems for the liquid coupling have not been solved - test results about removing the coupling medium from the vessel wall after testing are not yet available - further design work had to be reduced considerably by the end of 1975. By the end of the first quarter 1976 the further procedure will be decided.

Classification: 20.	
Title 1 (Original Language): Code Validation (RS 162 - III., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: IRS, Köln
Title 2 (english): Code Validation - Design Basis Accident Modelling Theory	Project Leader: Dr. Scharfe
Initiated (Date): March 1975 Status: finished	Completed (Date): December 1975 Last Updating (Date): December 1975

1. General Aim

To understand the effects of overprompt-critical power excursions of fast reactors chemical charges are fired in well-instrumental tank models. At the same time it is tried to verify these experimental data with computer codes. Thus the reliability of the codes used for hypothetical accidents should be augmented.

This report concerns a participation at this program over a period of one year.

2. Particular Objectives

The topics of this program are the

- tank models
- the instrumentation
- charge
- analysis of tests by computer codes

3.1. Experimental Facilities

The models to be used are of loop (SNR) and pool (CFR) reactor proportions. The model system is sufficient large to enable the components to be well made. The models consist of a base plate and a top plate held together with tie-bars and with the base firmly fixed to the large mass (~47 tons) of the bottom of a new bunker.

The general aim of the instrumentation to be used in these experiments is to place it at a standardized set of measuring locations that can be used in all experiments performed for validation. Strain gauges as well as pressure transducers are mounted at the walls and the latters in the fluid too.

To check the accuracy of all transducers, they are calibrated statically and dynamically before and after every shot. Permanent elongations can be measured additional by putting a rectangular grid over the whole of the outer cylinder area.

The experiments except the first four ones will be carried out using a low density high explosive charge (LD HE) in order that the stress levels reacted in the various components will be comparable with the stress levels achieved by a UO_2 vapour explosion or UO_2 - Na interaction. The properties of the LD charge which is developed in the UK represent a compromise between the demand of a start pressure as low as possible (o,5 -1,0 kbar) and the necessity of reproducibility and the independence of confinement.

3.2. Research Program

The safety assessment of fast reactor design has emphasized the need for validated codes. Therefore it is decided to start the series of experiments with very simple rigid tanks without internals filled with water. For every test a preshot calculation is made and if necessary a postshot calculation. If all relevant details of the experiment are also represented by the calculation the next more complicated test will be performed. A time step of about six weeks for each experiment is planned. At the end of the program a complex model of a reactor tank with flexible inner and outer walls, curved bottom, neutron shields, diagrid and dipplate shall be fired and calculated.

To reach this goal a parallel development and improvement of the computer codes is necessary.

4. Project Status

4.1. Progress to Date

At the begin of the reference period the first three (of 22) experiments were already performed. These tests should be identical and were made to enable a cross check concerning the instrumentation with two other laboratories in the UK where similiar experiments are carried out. A further test with a different height of the water level should give informations about the influence of this parameter on the roof impact. By reason of the big failure rate of pressure transducers (up to 50 %) at high pressures, new transducers had to be ordered from the UK. Additional an improved type was developed and manufactured in Ispra. The new pressure gauges arrived in december, so that no further test could be performed. Analysing the results of the first three tests, they showed an excellent agreement with each other. Im comparision with the results of the calculation the agreement was good, except some few points where the discrepancies could be explained by the numerical treatment in the calculation or the insufficient measurement. Nevertheless even this simple test showed the necessity of code validation experiments.

At the present, the following codes are available at the JRC Ispra and provided for validation: REXCO - H Release 2, SURBOUM Version VD 7 and VD 8, ARES 3 and ASTARTE. The CDC-Version of ARES 3 was adapted to the IBM-machine. In early 1975 it was decided to introduce compressibility into the incompressible eulerian SURBOUM-code. This work is almost finished i.e. simple experiments with rigid walls can be treated. The possibility of the graphical output of the results is given at all available programs.

4.2. Essential Results

The first experiments with a high explosive high density charge in a rigid tank are performed. It is proved that the experimental equipment, the instrumentation and the data recording system work satisfactorily. An experimental and a theoretical test report is produced. All relevant and available european computer codes are established at the JRC Ispra and are running on the IBM-machine. The limited code development of a compressible version of SURBOUM is encouraging. The LD charge and the according facilities as well as new transducers are arrived from the UK so that the next tests are prepared.

5. Next Steps

not relevant for this contract

6. Relation with other projects

none

7. Reference Documents

Quarterly reports in the series IRS-Forschungsberichte

Report period	April 1975	- June 1975	[illegible]
" "	July 1975	- Sept. 1975	IRS - F - 27
" "	Oct. 1975	- Dec. 1975	IRS - F - 28

8. Degree of Availability

No restrictions