COMMISSION OF THE EUROPEAN COMMUNITES

Directorate-General for Research, Science and Education

XII/D/3

NUCLEAR SCIENCE AND TECHNOLOGY

European Community
Water reactor
Safety Research Projects

VOLUME I

July 1978

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INTRODUCTION

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

The following guidelines were used in compilation of the index which is now in two volumes for convenience:

1) the first page relevant to each project is always on the right hand side of the document when opened.
2) All pages have a number, even if blank.
3) Within each class (chapter) the formats are assembed in the following order of country: Belgium
   Germany
   Denmark
   France
   Ireland
   Italy
   JRC Ispra
   Luxembourg
   Netherlands
   United Kingdom
4) Updated formats will be inserted in the relevant replacement position. When additional pages have to be inserted they will be numbered with the proceeding page number plus an oblique and an extra number (for example page 53/1 will be inserted following page 53).
5) Formats for new projects will normally be inserted following the last format of the relevant country within that class (chapter).
6) If a project is entered under more than one class (chapter), the full format is given only once in the most important position (indicated by a square box around the class number, for example $\boxed{1.1.1}$).
   Copies of the titles only are entered under the other class numbers.

I

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

IV

IV

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

20. Other topics

1. BLOWDOWN AND EMERGENCY 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

2

2

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.

Classification: 1

Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: NORCOOL. A Model for Analysis of a BWR under LOCA Conditions.	Organization: Risø National Laboratory
Initiated date: September 1976 Completed date: Status: under development	Scientists: J.G.M. Andersen P.S. Andersen P. Astrup N. Bech J. Eriksson M. Eget P. Hansen R. Holt J. Miettinen H.V. Larsen

1. General aim

Development of a model for analysis of a BWR under LOCA conditions.

2. Particular objectives

NORCOOL is a model for analysis of a BWR during LOCA conditions and for the evaluation of the performance of the ECC system.

NORCOOL is based on a detailed mechanistic modeling of the individual phenomena during a LOCA for a BWR. The two-phase flow model is based on a fully independent and multi-dimensional description of the phases, which allows counter current flow and thermodynamic non-equilibrium. The heat transfer accounts as well for the wall heat transfer as for the interfacial heat transfer and contains conduction, convection and radiation heat transfer. The heat conduction model is based on the one-dimensional Fourier equation, and two-dimensional conduction at quenching fronts is considered through correlations.

NORCOOL consists of two projects NORCOOL-I and NORCOOL-II. NORCOOL-I is a further development of RHC and thus contains only one fuel element, and the rest of the primary system is scaled accordingly. NORCOOL-II, however, contains an arbitrary number of parallel fuel elements in the core and the whole primary system inside the vessel is represented.

3. Experimental facilities and programme

4. Project status

   1. Progress to date

      NORCOOL-I is in the testing phase.
      NORCOOL-II is under development.

   2. Essential results

5. Next steps

6. Relation with other projects

The NORHAV project includes:

a) The core heat-up programme RHC, Risø.
b) A one-dimensional blow down computer program for reactor systems under development at IFA, Norway.
c) The Danish transient subchannel computer program TINA and the one dimensional blow down code RISQUE under development at Risø.
d) Updating of COBRA 3-C and RELAP 3 by STF, Finland and AE, Sweden.
e) A 64-rod (electrically heated) core heat-up experiment by AE, Sweden.

7. Reference documents

J.G.M. Andersen, P.S. Andersen, P. Astrup, N. Bech, J. Eriksson, R. Holt, H.V. Larsen, J. Miettinen, A. Olsen, NORCOOL, A Model for Analysis of a BWR under LOCA Conditions, NORHAV-D- 47, August 1977.

Jens G. Munthe Andersen, The Modeling of the BWR in NORCOOL-II, NORHAV-D-37, February 1977.

8. Degree of availability.

Titre Etude et réalisation d'une installation (PHEBUS) pour tester le comportement d'une grappe d'éléments combustibles PWR en cas de dépressurisation.	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) Study and construction of a nuclear facility for the testing of the behaviour of a PWR fuel under depressurization condition.	Organisme exécuteur : CEA/DSN/SES Cadarache Responsable : (SES)
Date de démarrage :01/06/71 Date prévue d'achevement: 31/03/78 E : actuel : en cours Dernière mise à jour : 13/12/77	Scientifiques :

1 - Objectif général :

Réalisation d'une installation d'expérimentation permettant de :

- reproduire les conditions d'environnement d'un réacteur PWR pour une grappe expérimentale d'une taille de 25 crayons.
- simuler les différentes conditions de dépressurisation étudiées dans l'analyse de sûreté d'un réacteur PWR.
- simuler les différents types d'injection de sécurité qu'il est prévu d'utiliser à l'occasion d'un tel accident.

2 - Installation expérimentale :

L'installation comporte :

- un réacteur source d'une puissance thermique de 60 MW, capable de produire les flux neutroniques et réajuster leur évolution pour que l'énergie dégagée dans l'assemblage d'essai soit représentative de celle qui existerait dans un réacteur de puissance même pendant la séquence expérimentale (chute des barres et action du refroidissement de secours).

- le coeur nourricier composé de 36 assemblages à crayons d'UO_2 faiblement enrichi, contrôlé au moyen de 6 barres de contrôle-sécurité à crayons de hafnium.

- le circuit d'eau déminéralisée qui refroidit le coeur à partir d'une réserve de 500 m3 servant de volant thermique et permettant de conduire un essai à pleine puissance pendant 20 à 30 minutes.

- une boucle d'essai avec :
  - une cellule en pile dans l'axe vertical du coeur, contenant la perche d'essai qui abrite la grappe de crayons combustibles à tester.
  - un ensemble de circuits hors pile contenus dans un caisson de 450 m^3 pouvant tenir 1,5 bars de surpression et rassemblant les fonctions de :
    - maintien des conditions nominales de fonctionnement d'un PWR.
    - déclenchement de la dépressurisation et du renoyage.
    - mesure des paramètres, et prélèvement d'échantillons.

- un ensemble de recueil et traitement des informations, comportant :
  3 enregistreurs magnétiques assurant au total 36 voies.
  1 multiplexeur -convertisseur analogique digital de 96 voies - de 12 bits.
  2 calculateurs MITRA-15-35 pour l'acquisition et le pré-traitement.

4 - Etat de l'étude :

A ce jour, la construction est terminée.
Les essais de sous-ensembles sont en cours.
Le transfert de la responsabilité de l'installation se fait progressivement de Technicatome vers le DSN/SES.

5 - Prochaines étapes :

Essais d'ensemble du réacteur nourricier et de la boucle expérimentale.
Chargement du réacteur au 2ième trimestre 1978.
Divergence du réacteur.
Premiers essais de dépressurisation avec combustible à la mi 78.

6 - Relation avec d'autres études :

Ce type d'installation répond à des soucis équivalents à ceux qui ont motivé le programme actuel des installations LOFT et PBF (USA).
Il permet d'effectuer un test global, complémentaire de toute une série d'expériences analytiques menées au CEA sur la sûreté des PWR.
Réf. fiche 147-1 -01

7 - Documents de référence :

disponible:

Plaquette "PHEBUS" (version en français et en anglais)

147-1 -01		1
Titre Programme PHEBUS : Méthodes de calcul et calculs préliminaires.		Pays FRANCE Organisme directeur CEA/DSN
Titre (anglais) PHEBUS experiment preliminary calculations.		Organisme exécuteur CEA/DSN-SETSSR Responsable DSN/SETSSR DSN/SES/Cad
Date de démarrage 1/06/76	Etat actuel terminée	Scientifiques
Date prévue d'achèvement 31/12/77	Dernière mise à jour 1/78	

1 - Objectif général :

Utilisation des codes d'accident de lère génération (RELAP 4, CERES...) pour définir le programme expérimental.

2 - Objectifs particuliers :

1) Prise en main des codes (essentiellement RELAP 4) ; étude des différentes options du code ; corrélations d'échange, débits critiques, équations des moments...

2) Etude de sensibilité de différents paramètres de la boucle : pertes de charge de la vanne représentant les composants de boucle, pertes de charge des creusets destinés à recevoir le combustible fondu, découpage de la boucle, injection de secours.

3) Etude du scénario d'initiation d'accident.

4) Etude des paramètres physiques de la boucle devant permettre de définir la grille des essais.

.../

4 - Etat de l'étude :

1) Avancement à ce jour :

Les quatre phases de l'étude (nommées ci-dessus) ont été réalisées.

2) Résultats essentiels :

1) La prise en main de RELAP 4 mod 3 a permis de se familiariser avec ses différentes options et de connaître ses possibilités.

2) L'étude de sensibilité a permis :

- de définir un certain découpage optimum de la boucle.
- pour la vanne représentant les composants, de montrer que sa définition actuelle ne permettait pas d'obtenir, au niveau du combustible, des débits représentatifs de l'accident LOCA ; d'où la nécessité dans la configuration actuelle d'adopter une dépressurisation double brèche.
- de montrer la nécessité de prendre en compte les échanges avec les structures.
- pour l'injection de secours : de montrer la nécessité d'une injection au moyen des accumulateurs.

3) Un scénario d'initiation d'expérience a été défini :

a) fermeture du circuit de refroidissement de brèche (t=0,3 s).

b) isolement de la boucle de dépressurisation et chute de barres (t=1,2 s).

c) ouverture de la vanne de dépressurisation (t=1,55 s).

4) Les paramètres physiques étudiés sont :

a) la taille de brèche.

b) l'emplacement de la brèche.

c) la puissance linéique.

d) la pression interne des crayons.

.../

Trois valeurs de paramètres ont été étudiées dans chaque cas afin de couvrir les possibilités actuelles de la boucle. Cela a permis de connaître l'influence relative de ces différents paramètres. En particulier nous avons constaté que la pression interne des crayons (compte tenu des modèles utilisés) n'avait pas d'influence sur les principales grandeurs thermohydrauliques et que le temps de dépressurisation était déterminé par la taille de brèche (légère influence de l'emplacement de la brèche).

Cette étude des paramètres physiques nous a conduit à faire la proposition de programme d'essais

6 - Relation avec d'autres études :

Etude de l'accident de perte de caloporteur primaire. Programmes expérimentaux et calculs prévisionnels de ces expériences (OMEGA).

Modèles et codes de calcul de 2ème génération pour l'étude de l'accident de perte de caloporteur dans les réacteurs à eau pressurisée.

7 - Documents de référence : rapports internes non disponibles

Titre Expériences de dépressurisation et de renoyage d'une grappe combustibles PWR : Programme PHEBUS.	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) Experiments on depressurization and reflooding of a PWR fuel assembly : PHEBUS project.	Organisme exécuteur : CEA/DSN-SES Cadarache Responsable : (SES)
Date de démarrage : 01/01/76 Date prévue d'achevement: 31/12/82 Etat actuel : en cours Dernière mise à jour : 13/12/77	Scientifiques :

1 - Objectif général :

Le programme expérimental a pour but d'étudier :

1) le comportement d'une grappe de crayons dans les conditions d'un accident de perte de réfrigérant primaire incluant l'intervention des systèmes de refroidissement de secours.

2) le comportement thermodynamique de la boucle et l'efficacité des systèmes de secours dans les conditions particulières à PHEBUS.

Associé aux programmes hors pile OMEGA et ERSEC, il doit permettre de préciser le domaine dans lequel un combustible demeure correctement refroidi pendant et après accident.

2 - Objectifs particuliers :

a) Validation des programmes de calcul décrivant la dépressurisation et le renoyage en vue de leur application au calcul des accidents des réacteurs de puissance;

b) Etude de l'injection de secours. Il s'agit de s'assurer que dans les conditions propres à la boucle PHEBUS les modèles de calcul décrivant le remplissage peuvent rendre compte des phénomènes observés.

c) Etude du comportement physico-chimique et mécanique des crayons combustibles et de l'assemblage pendant l'accident.

d) Etude du comportement des crayons combustibles lorsque les limites fixées par les critères de protection du coeur sont atteintes.

La plus grande partie du programme concernera des grappes de 25 crayons, mais les premiers essais seront effectués sur des dispositifs ne comportant qu'un seul crayon. L'installation a été étudiée pour tester des combustibles vierges et irradiés.

3 -Installations expérimentales et programme :

Ce programme sera entrepris sur l'installation PHEBUS, construite au CEN/Cadarache et actuellement en cours d'essais. Cette installation permet de réaliser les conditions thermodynamiques typiques d'un réacteur PWR sur une grappe de 25 crayons de 0,80 m de longueur active. (1,20 m environ au total). En fonctionnement en palier à ces conditions, elle permet de provoquer une dépressurisation avec une brèche de dimension et de position variables puis une injection de secours dont on peut faire varier : l'instant de mise en route, la pression, le débit.
Divers points d'injection sont possibles.

Pour réduire le programme à des dimensions acceptables et tenir compte de l'importance respective des diverses variables, on a classé les paramètres en plusieurs catégories :

- la catégorie I se compose des paramètres que l'on fera varier indépendamment : la taille de la brèche, sa localisation, la puissance linéique maximale des crayons combustibles et leur pression interne.

- la catégorie II se compose des paramètres qui seront étudiés en faisant varier un seul paramètre de la catégorie I, à savoir : la pression d'injection, le débit d'injection, la température de l'eau d'injection.

- enfin des essais réalisés dans des conditions bien précises pour étudier des phénomènes particuliers et notamment les phénomènes aux limites concernant l'intégrité des éléments combustibles seront appelés essais de la catégorie III : l'influence des points d'injection, le comportement des crayons combustibles dans les conditions limites permises par les critères et dans la mesure du possible, l'état des crayons en fin de dépressurisation.

On prévoit 7 à 8 essais par an à partir du début de 1979.

4 -Etat de l'étude :

1) Avancement à ce jour :

L'installation est construite. Les essais de recette sont en cours.
Les dispositifs d'expérimentation sont en voie de réalisation. Des calculs préliminaires permettant de préciser quantitativement le programme sont faits à l'aide des codes RELAP, CERES, FRAPT, FLIRA.
Le programme général des essais préliminaires est arrêté.
Les calculs neutroniques sont achevés et ont permis d'obtenir une distribution de puissance dans la boucle.

5 -Prochaines étapes :

Le chargement du coeur nourricier doit se faire en Août 1978.
La fin des essais en configuration neutronique définitive, interviendra en octobre 1978.

Les essais préliminaires de mise au point d'instrumentation auront lieu à la fin de l'année 1978.
Le programme expérimental pourra débuter en 1979 .

6 -Relation avec d'autres études :

Ce programme est entrepris en étroite connexion avec les programmes ERSEC et OMEGA réalisés au Service des Transferts Thermiques de GRENOBLE.

7 -Documents de référence :

disponible. :

Plaquettes PHEBUS IPSN/DSN 1977 .

145-1 -06		1.
Titre Méthodes de mesure en double phase		Pays FRANCE
		Organisme directeur CEA /DgCS
Titre (anglais) Development of two phase flow instrumentation		Organisme exécuteur CEA/DTCE-STT (GRENOBLE)
		Responsable STT - Grenoble
ate de démarrage 1974	Etat actuel en cours	Scientifiques
Date prévue d'achèvement 31/12/79	Dernière mise à jour 1/78	

1 - Objectif général :

Mesures des paramètres de l'écoulement double phase au cours de l'accident de dépressurisation.

2 - Objectifs particuliers :

Developpement de débitmètres pour écoulement diphasique (venturi et moulinets)
Développement de méthodes de mesure de taux de vide par neutronographie.

3 - Installations expérimentales et programme :

DEDIF : Circuit eau-argon d'étalonnage en régime permanent.
CANON : Test en régime transitoire.

.../

4 - Etat de l'étude :

Avancement à ce jour

Appareils de mesure par neutronographie étalonnés.
Moulinet et venturi étalonnés en régime permanent.
Développement technologique en cours des paliers de moulinet.
Utilisation d'un moulinet (Ø = 20 mm) avec roulement à billes effectuée avec succès.
Moulinet avec palier fluide en cours d'essai.

5 - Prochaines étapes :

Etude des transitoires de dépressurisation avec un moulinet de 80 mm de diamètre.
Venturi à l'étude sur CANON.

6 - Relation avec d'autres études :

Expérience OMEGA : Détermination des débits aux extrémités des sections d'essais de la Boucle OMEGA.

Expérience PHEBUS : Mesure des débits en branche chaude et en branche froide de la boucle d'essai PHEBUS.

7 - Docuements de référence disponibles :

- "Void Fraction Measurements by Neutron Attenuation and Neutron Scattering Method", J-C. ROUSSEAU, J. CZERNY, B. RIEGEL . Communication au Transient Two Phase Flow Meeting - Toronto (3-4 août 1976) .

- R. FRANCK et J. MAZARS
  "Determination of mass flow rate and quality using a venturi and a turbine meter, Europan two phase flow" - June 77 - Grenoble .

- R. FRANCK et J. MAZARS et R. RICQUE
  "Determination of mass flow rate and quality using a turbine meter and a venturi, Conference on heat transfer and heat flow in water reactor safety", MANCHESTER, Sept. 77 .

Titre Thermohydraulique du LOCA : Etude expérimentale du refroidissement de secours des réacteurs à eau : Programme ERSEC		Pays FRANCE Organisme directeur CEA/DgCS - EDF/SEPTEN
Titre (anglais) LOCA Thermohydraulics : Experimental investigation of water reactors safety injection : ERSEC project.		Organisme exécuteur CEA/DTCE-STT (GRENOBLE) Responsable STT - Grenoble
Date de démarrage 01/01/72	Etat actuel en cours	Scientifiques
Date prévue d'achèvement 31/12/78	Dernière mise à jour 1/78	

1 - Objectif général :

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

- Objectifs particuliers :

Développement de modèles physiques pour l'interprétation des expériences.

3 - Installations expérimentales et programme :

Boucle ERSEC :

- Expérience de renoyage à débit constant en tube,
- Expérience de renoyage à charge constante en tube,

Expérience de renoyage à débit constant en grappe 36 barreaux 17 x 17.

.../

4 - Etat de l'étude :

1) Avancement à ce jour

- Une première campagne d'essais de renoyage à débit constant sur une grappe 36 barreaux PWR 17 x 17 a eu lieu en 1975 mais a été interrompue par suite d'une perte d'isolement électrique des éléments chauffants.
  Essais de renoyage à débit constant sur une section d'essais tubulaire très instrumentée et avec isolation thermique par enceinte à vide en cours.

- Essais de renoyage à débit constant sur tubes de différentes longueurs chauffants

- Interprétation en cours des expériences : progression du front de trempe avec le modèle PSCHITT et corrélation d'échange en aval avec le modèle FLIRA.
  Etude de l'écoulement amont.

2) Résultats essentiels

Développement de modèles physiques représentant le rayonnement en aval du front de trempe.
Une corrélation des coefficients d'échange au niveau du front de trempe a été développée permettant de retrouver le profil axial des températures pour une large gamme des vitesses de progression du front de trempe.

5 - Prochaines étapes :

- Essais en grappe

- Expériences fondamentales en vue d'étudier l'écoulement amont (mesures de la ligne piezomètrique et du profil axial de taux de vide) pour l'ajustement du modèle théorique "codes avancés".

7 - Documents de référence disponibles :

" Heat Transfer during the Reflooding Phase of a Tubular Test Section ", D.ANDREONI, M.COURTAUD, R.DERUAZ - European Two Phase Flow Meeting - Harwell 1974.

" Echanges thermiques lors du renoyage d'un coeur de réacteur à eau ", D.ANDREONI - Thèse de Docteur Ingénieur, 28/11/75.

.../...

- " FLIRA : Reflooding calculation model following an accidental primary fluid loss ", JP. L'HERITEAU and D. MENESSIER.
European two Phase Flow Meeting Haifa (June 1975)

- " Modélisation thermohydraulique des écoulements et des échanges de chaleur au cours du renoyage d'un coeur de réacteur à eau pressurisée ", P. RAYMOND -
Thèse de 3ème cycle (Nov. 77)

- " Modeling of quench front progression and heat transfer by radiation during reflooding of a tubular test section ", P. CLEMENT, R. DERUAZ -
European two phase flow meeting - Erlangen (Juin 76)

- " Modeling of heat transfer by radiation during the reflooding phase of LWR ", R.DERUAZ, B. PETITPAIN - Specialist's Meeting on the behaviour of water reactor fuel elements under accident conditions - SPATING (Norway) (Sept. 76)

- " Interprétation des essais ERSEC sur le refroidissement de secours des réacteurs à eau pressurisée au moyen du code FLIRA ", C. REVIGLIO -
Thèse de 3ème cycle (Nov. 77)

- " Some aspects of reflooding studies en FRANCE (1977) ", N. TELLIER -
4 th meeting of the CSNI-ECCS ad hoc group - GRENOBLE, Juin 1977

- " Application of FLIRA code to fit ERSEC experiments ", P. RAYMOND , JP L'HERITEAU -
European two phase flow meeting - GRENOBLE (Juin 1977)

- " Development of reflood codes FLIRA and PSCHIT.
Physical modeling and interpretation of ERSEC experiments ", P. CLEMENT , R.DERUAZ, JP L'HERITEAU , P. RAYMOND, P. REGNIER, M. REOCREUX -
Fifth annual water reactor safety research information meeting -
(WASHINGTON Nov. 1977);

145-1 -01/4151-10	
Titre Accident de perte de caloporteur dans les réacteurs à eau pressurisée : codes 1ère génération.	Pays FRANCE Organisme directeur CEA/DSN
Titre (anglais) LOCA and ECCS studies on PWR : first generation codes.	Organisme exécuteur CEA/DSN - SETSSR Responsable SETSSR / Fontenay
Date de démarrage 01/01/71 — Etat actuel : en cours Date prévue d'achèvement 12/80 — Dernière mise à jour 1/78	Scientifiques

1 - Objectif général :

En s'appuyant sur les expériences françaises et étrangères, étude des possibilités (et qualification) d'ensembles de codes décrivant tout l'accident dans le but de les utiliser pour l'évaluation de sûreté des réacteurs de puissance.

2 - Objectifs particuliers :

1) Etude des options du code RELAP 4 mod 5.
2) Qualification des codes sur expériences OMEGA et problèmes standards CSNI.
3) Précalculs de l'expérience EURATOM LOBI.
4) Calculs préliminaires PHEBUS.
5) Calculs de sensibilité pour les réacteurs de puissance.

.../

4 - Etat de l'étude :

1) Avancement à ce jour :

- Les calculs préliminaires PHEBUS sont terminés (voir fiche PHEBUS) ; ils ont permis de mesurer la sensibilité des différents paramètres essentiels à la conduite des essais ; ces études serviront de base à l'établissement de la grille des essais.

- L'interprétation des essais OMEGA est en cours, on a surtout fait porter l'effort sur un essai sans chauffage et un essai avec chauffage.

- Les premiers calculs préliminaires LOBI ont été réalisés et leurs résultats composés avec les calculs similaires d'autres pays de la Communauté.

- Le problème standard CSNI N° 5 (dépressurisation sans chauffage de LOFT avec injection de sécurité) a été réalisé mais avant l'expérience. Les résultats seront comparés avec l'expérience vers le milieu de 1978.

- Des études de renoyage relatives à PHEBUS ont été réalisées avec les codes CERES et RELAP FLOOD.

- Le 1er calcul concernant l'accident de dépressurisation d'un réacteur de puissance a commencé. Etablissement des données RELAP à partir des données brutes fournies par le constructeur.

2) Résultats essentiels :

- Les calculs préliminaires PHEBUS ont montré l'importance relative des paramètres puissance linéique maximale, taille de brèche, localisation de la brèche, pression interne initiale des crayons. Dans l'état actuel des modèles ce dernier paramètre semble n'avoir que peu d'influence.

- Les calculs ont montré la nécessité de bien prendre en compte les fuites thermiques dans toute la boucle. Enfin un certain nombre de paramètres, propres à la boucle, se sont révélés comme très importants: citons principalement la perte de charge de la vanne VA EP 14 qui permet de by passer le circuit de charge au début du transitoire et la perte de charge des creusets en débit direct et inverse. La norme de ces pertes de charges devra être effectuée hors pile avant le début des essais.

- L'interprétation des essais OMEGA a pour but de faire un choix dans les différentes options de RELAP afin de rendre compte au mieux des résultats expérimentaux. Ceci permet ensuite d'utiliser ce code dans des calculs réacteurs avec un meilleur niveau de confiance quand à la validité des résultats.
  Les premiers calculs ont été effectés avec la version RELAP 4 mod 3. Il s'est alors avéré que pour bien rendre compte des essais sans chauffage il était nécessaire :
  1°/ d'utiliser le début à la brèche de MOODY avec un facteur C_D=0.6
  2°/ d'utiliser le modèle de séparation de phases dans le plenum opposé à la brèche.

.../

- Les calculs ont été ensuite repris avec la version améliorée RELAP 4 mod 5.
  Il faut avec cette version :
  1°/ utiliser le modèle homogène à la brèche.
  2°/ utiliser le modèle de glissement entre phases dans la section d'essais et dans le plenum opposé à la brèche.
  Les résultats de cette version semblent plus proches de l'expérience que ceux de la version mod 3.
  L'interprétation des essais avec chauffage a montré des déficiences quand au calcul des transferts de chaleur.

- Les études de renoyage concernant PHEBUS ont montré la possibilité d'utiliser le code RELAP FLOOD. Mais on est toujours sans moyen pour calculer l'évolution fine des températures de gaine pendant le renoyage : il nous faudra pour cela attendre que le code FLIRA soit opérationnel et validé par les essais ERSEC. On attend également sur ce sujet l'arrivée du code américain RELAP 4 mod 6, mais il semble qu'il y ait là des retards.

5- Prochaines étapes :

- L'interprétation des essais OMEGA devra maintenant faire porter l'effort sur les transferts thermiques entre gaine et fluide.

- La poursuite des calculs LOBI sur la base d'un programme expérimental proposé par la France, devra permettre l'introduction dans la 2ème partie du programme allemand (A2) d'inclure des essais intéressants pour la communauté.

- Le calcul de l'accident pour le réacteur de puissance sera mené à son terme et servira de base pour les calculs de sensibilité ultérieurs (sensibilité du système, sensibilité aux modèles du code).

- Problèmes standards ultérieurs.

6 - Relations avec d'autres études :

Expériences OMEGA, ERSEC, CANON, MOBY DICK, EDGAR, PHEBUS, code FLIRA 2 et code POSEIDON.

7- Documents de référence : - rapports internes non disponibles .

145-1 -02		1
Titre : DEVELOPPEMENT DE MODELES ET CODES DE CALCUL DE 2^{e} GENERATION POUR L'ETUDE DE L'ACCIDENT DE PERTE DE CALOPORTEUR DANS LES REACTEURS A EAU PRESSURISEE : Programme général coordonné .		Pays: FRANCE Organismes directeurs: CEA-DSN - EDF-SEPTEN
Titre (anglais) DEVELOPMENT OF ADVANCED MODELS AND ADVANCED CODES FOR THE STUDY OF THE LOCA IN PWR : Coordinated general program .		Organisme exécuteur: CEA - EDF Responsable: DSN-SETSSR FONTENAY-aux-ROSES
Date de démarrage : 01/01/77	Etat actuel: en cours	Scientifiques
Date prévue d'achèvement : 31/12/81	Dernière mise à jour 1/78	

1 - OBJECTIF GENERAL -

Elaboration de codes avancés amenés à remplacer les codes de 1ère génération pour le calcul des accidents de perte de caloporteur dans les réacteurs à eau pressurisée .

2 - OBJECTIFS PARTICULIERS -

2.1. Ecriture de modèles physiques

Ces modèles doivent décrire l'ensemble des phénomènes physiques intervenant au cours des diverses phases de l'accident (les plus importants sont soulignés)

Les modèles mis au point pour CLYSTERE sont en partie utilisés

- Ecoulement diphasique 1 D
- Singularités géométriques
- Brêche 1 D
- Piquage pressuriseur
- Injection de secours
- Ecoulement et transfert de chaleur en dépressurisation
- Ecoulement et transfert de chaleur en amont du front de trempe
- Ecoulement et transfert de chaleur en aval du front de trempe
- Conduction 2 D - Modèle de front de trempe
- Volume 0 D
- Downcomer
- Volume 3 D
- Piquage sur volume

- Thermomécanique des crayons
- Singularités complexes
- Cross flow
- Brêche 3 D
- Neutronique
- Ecoulement air eau vapeur
- Séparation sur les structures
- Modèle de flooding
- Modèle de pompe

2.2. Validation des modèles physiques

Chaque modèle doit être validé sur des expériences françaises (voir liste au paragraphe 3) ou étrangères .

2.3. Elaboration d'un système informatique (POSEIDON)

Cette élaboration comprend les tâches suivantes :

- mise au point d'un langage et d'un système de gestion de mémoires (ESOPE, OTOMAT)
- écriture des fonctions de l'eau
- écriture de banque de données, de bibliothèques
- écriture de moniteurs
- assemblage du système .

Ce système doit conduire à un code de type modulaire .

2.4. Méthodes numériques

Ces études ont pour but d'optimiser le temps de calcul tout en assurant le maximum de souplesse possible . Ces études comprennent les tâches suivantes :

- Développement de méthode numérique d'assemblage de modules :
  . méthode par fonction de transfert
  . méthode SOR
- Mise au point d'algorithme permettant d'avoir :
  . des pas en temps variables selon l'abscisse
  . des pas en espace variables selon le temps
- Organisation des méthodes numériques utilisées dans la résolution des équations de l'écoulement, de la conduction ... .

2.5. Ecriture des modules

Ces modules doivent être écrits dans le cadre du système POSEIDON . Ils doivent être ensuite assemblés au fur et à mesure des besoins : dépouillement d'expériences, problèmes standard, calcul réacteur .

Les modules recensés sont les suivants :

- Ecoulement

.../...

- Singularités
- Brèche
- volume OD
- tuyau non chauffant
- tuyau avec piquage type piquage pressuriseur
- tuyau chauffant en dépressurisation
- tuyau chauffant en renoyage
- tuyau avec piquage injection de secours
- pompe
- plenum supérieur
- plenum inférieur
- downcomer
- générateurs de vapeur
- crayon combustible
- canal en dépressurisation
- canal en renoyage
- neutronique
- coeur en dépressurisation
- coeur en renoyage
- coeur en renoyage de type simplifié
- accumulateur
- pressuriseur
- couvercle
- baffles

Confinement : - écoulement entre casemates
- enceinte

2.6. Réalisation d'assemblages

Les assemblages seront réalisés pour interpréter les expériences données ci-dessous et dans l'ordre suivant :

OMEGA et CISE
MARVIKEN
AQUITAINE
ERSEC système
PKL
SEMISCALE renoyage
PHEBUS
LOFT - SEMISCALE
LOBI

L'assemblage réacteur est l'étape finale de réalisation du code avancé.

3 - INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Ces expériences servent de base à la validation des modèles physiques.

MOBY DICK	Ecoulement diphasique Ecoulement critique (basse pression)	(voir fiche 145-1-03)
SUPER MOBY DICK	Ecoulement diphasique Ecoulement critique (haute pression)	(voir fiche 145-1-03)
CANON SUPER CANON	Ecoulement diphasique en dépressurisation adiabatique	(voir fiche 145-1-06)
MARVIKEN-CFT	Ecoulement diphasique critique à grande échelle	(participation au programme international voir fiche 145-1-02)
OMEGA	Ecoulement et transfert de chaleur en dépressurisation	(voir fiche 145-1-06)
ERSEC	Ecoulement et transfert de chaleur en renoyage modèle de front de trempe	(voir fiche 145-1-07)
EPIS	Injection de secours	(voir fiche 145-1-05)
EVA, EPOPEE	Pompe en diphasique	(EVA programme FRA-CEA) (EPOPEE programme EDF)
EDGAR	Thermomécanique de la gaine du combustible	(voir fiche 145-2-03)
PHEBUS	Thermomécanique du combustible	(voir fiches 147-1-01, 147-1-02)
Cuve LNH	Downcomer en monophasique	(programme EDF)
REBECA	Ecoulement diphasique air-eau-vapeur entre les casemates	(voir fiche 145-1-09)
ECOTRA	Condensation dans les enceintes	(voir fiche 145-1-08)
AQUITAINE	Interaction comportement mécanique écoulement	

4 - ETAT DE L'ETUDE

4.1. Avancement à ce jour

4.1.1. Elaboration du système informatique

- ESOPE-OTOMAT

Le travail de mise au point du langage ESOPE et du système OTOMAT est terminé (sur IBM). Des améliorations sont en cours.

- Fonctions de l'eau

Un premier jeu de fonctions de l'eau est actuellement réalisé.

- Banques - Bibliothèques

Des premières versions de bibliothèques ont été écrites. La grille de RELAP 4 mod 5 a été introduite.

- Moniteurs

L'écriture des moniteurs est en cours.

4.1.2. Méthodes numériques

- Méthodes numériques d'assemblage

La méthode par fonction de transfert est en cours de test.

La méthode SOR a été testée dans le code CLYSTERE sur différents assemblages allant jusqu'à l'assemblage réacteur. Son introduction dans le système POSEIDON est en cours.

- Optimisation des méthodes numériques

Un travail d'amélioration du temps de calcul du module écoulement est en cours.
Un travail de comparaison de méthodes numériques va être démarré.

4.1.3. Ecriture des modules - Développement et validation des modèles physiques

Module écoulement

Un modèle d'écoulement (HEXECO) prenant en compte le glissement et les déséquilibres thermodynamiques a été écrit (modèle à six équations). Ce modèle est une "extension" du modèle SERINGUE (code CLYSTERE) qui ne prenait en compte que le déséquilibre thermodynamique.

Les lois de transfert du modèle HEXECO ont reçu une première validation sur les expériences MOBY DICK eau-air et eau-vapeur.

Des tests sur l'inversion de débit, sur l'apparition de la vapeur.... sont en cours.

Une version HEXECO 001 du module d'écoulement a été réalisée et figée. Son introduction dans le système POSEIDON est en cours.

Module tuyau chauffant en dépressurisation

Ce module est en cours d'écriture.

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Module tuvau chauffant en renoyage

- code FLIRA

  Une version synthétique des différentes versions précédentes de FLIRA est en cours de test.

- Modèle d'écoulement et de transfert de chaleur en amont du front de trempe.

  Des premiers ajustements sur les expériences ERSEC ont été effectués.

  Des essais de modélisation avec le modèle drift flux sont en cours.

- Modèle d'écoulement et de transfert de chaleur en aval du front de trempe.

  Des premiers ajustements sur les expériences ERSEC ont été effectués.

- Modèle de front de trempe

  La corrélation de coefficient d'échange au niveau du front de trempe est mise au point à l'aide du code PSCHIT.

  Deux modèles (A et B) de coefficients d'échange ont été testés.
  Le modèle B a donné des résultats assez satisfaisants.
  Une corrélation avec ce modèle a été établie pour une pression de 1 bar.
  Cette corrélation est en cours d'extension aux pressions supérieures.

  Des tests de sensibilité lorsque l'on transpose les calculs au combustible ont été effectués.

Module pompe

Le travail a porté sur le module pompe de CLYSTERE. On effectue actuellement les ajustements du modèle en écoulement monophasique dans les 4 quadrants.

4.2. Résultats essentiels

- Dans la description des écoulements diphasiques il est apparu indispensable de prendre en compte à la fois le déséquilibre thermodynamique et le déséquilibre mécanique (glissement).

- Les tests de sensibilité dans la transposition au combustible suivant le modèle d'échange de chaleur au niveau du front de trempe montre la possibilité d'erreurs assez importantes si le modèle, même ajusté sur les expériences, ne correspond pas au phénomène physique réel.

5 - PROCHAINES ETAPES

5.1 Elaboration du système informatique

ESOPE-OTOMAT	amélioration - implantation sur CDC
Banques- Bibliothèque	réalisation de versions améliorées
Moniteurs	réalisation d'un ensemble complet de moniteurs
Système	mise au point d'un mini système permettant la réalisation d'un assemblage OMEGA

.../

5.2 Méthodes numériques

Méthodes numériques d'assemblage

Suite des tests sur la méthode par fonction de transfert.

Fin de l'introduction de la méthode SOR dans POSEIDON.

Optimisation des méthodes numériques

Suite du travail de comparaison et du travail d'amélioration du temps de calcul du module HEXECO.

5.3 Ecriture des modules - Développement et validation des modèles physiques

Module écoulement

- introduction du module HEXECO 001 dans POSEIDON.
- amélioration des modèles de transfert aux interfaces :
  reprise des dépouillements des essais MOBY DICK eau-air (frottement interfacial) et calculs d'écoulements dans des gammes différentes de paramètres.

Module singularités

Démarrage de l'écriture.

Module tuyau chauffant en dépressurisation

Fin de l'écriture du module (opérationnel informatiquement mais non validé physiquement)

Module tuyau chauffant en renoyage

- Code FLIRA : Fin des essais
- Modèle d'écoulement et de transfert de chaleur en amont du front de trempe.
  Suite de l'ajustement du modèle drift flux.
- Modèle d'écoulement et de transfert de chaleur en aval du front de trempe.
  Définition d'une stratégie de dépouillement systématique des essais ERSEC.
- Modèle de front de trempe.
  Extension du domaine de validité de la corrélation du modèle B.
  Continuation des tests de sensibilité dans la transposition.

Module pompe

- Fin de l'ajustement en écoulement monophasique.
- reprise de la modélisation compte tenu des résultats obtenus dans la modélisation des écoulements diphasiques.

Module tuyau avec piquage injection de secours

Démarrage de la modélisation.

.../

5.4 Assemblage

Réalisation dans le cadre du système POSEIDON de l'assemblage OMEGA .

6 - RELATIONS AVEC D'AUTRES ETUDES

- Voir paragraphe 3 .

7 - DOCUMENTS DE REFERENCE DISPONIBLES

G. HOUDAYER, G. LE COQ, B. PINET, M. REOCREUX, J-C. ROUSSEAU

Modeling of two phase flow with thermal and mechanical non equilibrium .

Rapport DSN 166 e - Présentation at the 5 th Water Reactor Safety Research Information Meeting - Washington 1977 .

P. CLEMENT, R. DERUAZ, J-P. L'HERITEAU, P. RAYMOND, P. REGNIER, M. REOCREUX .

Development of reflood codes FLIRA an PSCHIT. Physical modeling an interpretation of ERSEC experiments .
Rapport DSN 167 e - Présentation at the 5 th Water Reactor Safety Research Information Meeting - Washington 1977 .

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/C
Initiated 1973	Completed 1975	Scientists M. SUREAU
Status	Last updating : 20.01.75	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.

...

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, JG. 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, JG. 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.

Title 1 (Original language) Statistical analysis of randome signals	Classification II - 3 - 4 - 8 IO - I4
Title 2 (English)	Country ITALY Sponsor Organisation } CNEN
Date initiated I966 Date completed in progress Last updating April I977	Project Leader A. Federico

I. - General aim — Concern the developments of statistical methods for acquisition and elaboration of experimental data coming from nuclear power plants and experimental loops.

2. - Particular objectives — Apply statistical methods to study: Reactor physics, Thermohydraulic and mechanical effects, Acoustic noise, Fuel coolant interaction etc.

3. - Experimental facilities — Transducers, d.c. amplifiers, filters, magnetic records.
A package of programmes for statistical analysis.

4. - Project status — The main efforts are now devoted to the LMFBR reactors.

5. - Relation to other projects — These studies are made in collaboration with responsibles of experimental facilities

6. - Reference documents —

I) L. Cimorelli - A. Federico
Applications of spectra analysis techniques to examine natural and superimposed neutronic flux fluctuations in a nuclear power reactor.
Rapp. CNEN -ING(69)3 - Marzo I969

2) A. Federico - S. Taglienti
Frequency and time-domain systems for statistical signal elaboration developed in CNEN laboratories. IAEA specialist meeting on Analysis of Measurements to Diagnose Potential Failures.
Roma, Aprile IO-II, I972

3.8
38

Title 1 (Original language)	Classification
Statistical analysis of random signals	I - 3 - 4 - 8 IO - I4

7. - Degree of availability Know-how and facilities for statistical analysis are available.

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:

blow 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.

N.V. KEMA

CLASSIFICATION : 1. 7.1. 7.2

TITLE :

COUNTRY: THE NETHERLANDS

SPONSOR :
KEMA

ORGANIZATION :
KEMA

TITLE (ENGLISH LANGUAGE):

Calculations of the consequences of pipe breaks in reactor systems

PROJECTLEADER :
R.M. vam Kuijk

INITIATED : –	LAST UPDATING : 1978
STATUS : progressing	COMPLETED : 1980

SCIENTISTS :
Kloeg
Oppentocht
Talens

General aim

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

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.

Experimental facilities and programme

- Participation in the Marviken project
- Main computer programmes:
  slow down : BRUIS
  core heat up: CHEMLOC-5, BUBBLE
  containment : RIS, DRUKSTUK, ZOCCO, RISLIFT
  forces : MARC

Project status

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

Next steps

- Calculation of Marviken II and III results.

Relation with other projects

Not applicable.

Reference documents

Internal KEMA reports.

Degree of availability

Free on basis of exchange with other programmes and results.

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Classification: 1.1

Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: DINO - Core heat-up during blow down	Organization: Risø National Laboratory
Initial date: February 1971 Completed date: September 1972 Status:	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.

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
Heating in a reactor fuel element rod under transient
conditions. Part II.
Risø-M-1533 (1972)

8. Degree of availability

Available.

Classification: 1.1

Title:	Country: DENMARK
	Sponsor: Risø
Title: NORHAV- P(B)WR blow-down computer program (TINA)	Organization: Risø National Laboratory
Initiated date: 1973 Completed date: Status: progressing	Scientists: Peter S. Andersen Niels Bech

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 distribution of coolant mass, -flow, -enthalpy and pressure as well as fuel rod temperature in a P(B)WR core during the blow-down phase of a loss-of-coolant accident. The model which is based on the sub-channel approach includes slip and thermal non-equilibrium between steam and water.

3. Experimental facilities and programme

4. Project status

The program is operational. It has been tested against the LOFT Semiscale blowdown experiments.

5. Next steps

Completion of program layout and documentation.

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 core heat-up programmes NORCOOL-I and NORCOOL-II at Risø.

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

d) A three-dimensional computer program for the analysis of PWR core transients (TINA + nodal theory neutronics).

7. Reference documents

8. Degree of availability

Available on exchange basis when completed.

PROJECT TITLE : Blowdown code assessment	LWR $\boxed{1.1}$ - 1.2
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : JRC Ispra Establishment
DATE INITIATED : Jan. 1974 DATE COMPLETED :	PROJECT LEADER : L. Larsen

Description :

1) General aim
- To acquire a working knowledge of the scope and limitations of the major accessible blow down/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 blow down programme.

3) Project status - Progress to date
A library of blow-down programmes are kept updated at the JRC Ispra computing center. Calculations are mainly performed with the Relap 4/MOOD 5 and Relap-UK, but various other programmes are used for special analysis of blow-down transients.

Standard Problem 3 has been analysed by the programmes Relap-4/MOOD 2, Relap-UK and Relap-3.
Differences and similarities between predictions of the various models and between predictions and measurements have been analysed. Critical flow two phase pressure drops play an important role in the investigated transients. This problem has led to development of a special programme Relap-3/MODA which calculates critical flow pressure drops for blow down transients.
The one dimensional models of heat conduction and heat transfer taking place in a nuclear fuel rod and its surrounding channel during a blow down transient have been analyses in search for mathematical methods faster and more accurate than the traditional finite difference methods. Polynomial spline functions in connection with variational methods have been shown to be promising tools, even in presence of non linearities and strong power transients. The work is in its reporting stage.

A litterature study of blow down experiments has been started. Emphasis is made on the Idaho Nuclear Corporation Semiscale experiments series 600-1000 and on the Semiscale MOD-1 experiments. The work is in its initial stage.

Reference documents

51.719

1. 1976 Safety progress report of the Ispra Establishment ACS 94e
2. SINDOC (76) 30 Calculations for Standard Problem 3 using Relap-3, Relap-4 and Relap-UK. W. Kolar, M. Lolk Larsen and L. Piplies. December 1976.

Classification
$\boxed{1.1.}$
(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 1/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

Reference documents :

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

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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.

PROJECT TITLE : Loop Blowdown Investigations (LOBI)-Project : Influence of PWR primary loops on blowdown.	LWR $\boxed{1.1}$ 1.2
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : J.R.C. Ispra
DATE INITIATED : January 1974 DATE COMPLETED : December 1976	PROJECT LEADER : W. Riebold

Description :

1. General Aim

Design and construction of a large scale two-loop blowdown test facility.

Performance of loss-of-coolant experiments (LOCEs) by simulating tube ruptures of different sizes at several positions within a PWR primary cooling circuit system.

2. Particular Objectives

Experimental investigation of the role of the different components of a PWR primary cooling circuit during a blowdown by the measurement of the main thermohydraulic quantities, especially those which influence the core cooling, i.e. the flow and heat transfer conditions and the pressure differences.

The experimental results will be applied to check and improve the blow-down codes and associated theories used for the safety analysis of LWRs.

3. Research Programme

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

Programme A, to be performed for the BMFT-Bonn in the framework of the R&D contract RS-109/143-73 PIHOD, concluded between the BMFT-Bonn and the C.E.C., will be concerned with the investigation of the influence of the following parameters on the blowdown:

- rupture size and position
- pumps operation performances
- initial power level
- heating-power time-funtion during blowdown
- strength of heat sink (steam generator secondary side conditions)
- downcomer resistance and volume
- ECC water injection positions

An appropriate test matrix A comprising 60 tests has already been defined by a German Expert Group at the very beginning of the project work; this test matrix has still to be revised for being adapted to the final parameter situation and test facility configuration resulting from several modifications to be applied during the revision and construction phase of the project.

Programme B, to be performed for the Commission of the E.C. after conclusion of Programme A, will be concerned with the

- performance of some reference tests (repetition of tests of programme A) which at the same time constitute reproducibility tests
- performance of component studies, to be done with this test rig after having modified certain components; the purpose of these tests is to investigate the influence of the geometrical shape or the elevation of these components on the blowdown.

Seven such modifications of the programme A test rig have already been agreed upon by an ad-hoc Working Group of experts of the Community member countries:

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

The funds from the Commission's budget, necessary for these modifications, had been allocated to the LOBI-project budget in the beginning of 1975 and enabled orders to be placed for these modifications together with the orders for all mechanical loop components.

An appropriate test matrix B is actually being elaborated by the experts of the before mentioned ad-hoc working group on the basis of a first proposal submitted during the last session in October 1976.

4. Experimental facilities, computer codes

A 4-loop primary cooling 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 representing the brok n "reactor" loop. Both experimental loops are active loops containing a pump and a steam generator each.

Tube ruptures of various rupture sizes(from double ended down to small leak) are to be simulated at three different positions within the broken loop (hot leg, cold leg, loop seal).

The scaling factor of 712 for power, mass flow and volume led to

- 5 MW heating power input to a 64 heater rod bunle as reactor core simulator,
- 21 kg/s and 7 kg/s fluid mass flow in the intact and broken loop respectively
- about o,7 m^3 volume content of the primary loop test system.

The loop system and component design has been done for 16o bar and 325^{o} 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 the components
- the elevations of the components
- the lengths of the heat transfer surfaces (core rod bundle, steam generators)

equal to the corresponding reactor values.

Two accumulators (6o bar and 3o^{o} C operating pressure and temperature respectively) of different volume content (28o and 95 dm^3) for the two loops are providing ECC water for both, separate and combined cold leg and hot leg injection into both loops.

A secondary loop system provides heat removal from the primary loops in the steam generators and operates at 52 bar system pressure and in a temperature range between 21o^{o} and 27o^{o} C (steam generator secondary side inlet and outlet temperature respectively).

The measurement of fluid absolute pressure and pressure differences, absolute temperature and temperature differences, mass flow and density will be done at the boundary of all loop components, where special tube inserts, called spool pieces, being instrumented with appropriate measuring devices, are mounted into the loop tubing. The same fluid quantities will be measured also within the reactor model region (downcomer, core, lower and upper plenum).

Furthermore the outer surface temperature of the heater rods will be measured at 192 positions equally distributed over the heated bundle region.

Test facility design calculations are done by the "theory and experiment" group of the project staff with the RELAP4-MOD 2 code and by the LRA-Garching (FRG) with BRUCH-D code.

Pre-prediction and results evaluation calculations will be done by the same groups with blowdown computer codes of the same code family.

5. Progress to Date

During the report period the project activities were concerned mainly with works of phase III of the project planning: mounting of the LOBI test facility.

More in detail the following works have been performed:

- Completion of a new building containing two rooms for housing the data acquisition and signal processing system, and the loop regulation and control instruments and panel
- Completion of the construction works in the laboratory hall: concrete cavity in the floor for housing the reactor model of the test facility, foundations for the loop scaffolding and for the 5,5 MW rectifier system, concrete bunker for simulating the reactor containment
- Mounting of the loop scaffolding, and of a special scaffolding for assembling and disassembling of the reactor model
- Mounting of the 5,5 MW rectifier system
- Mounting of the big components of the primary loops, except reactor model and pumps, of the secondary loop and of the tertiary plant.

- First part of factory technical acceptance tests of the LOBI pumps and their electrical drive system allowing impeller speed time control
- Installation works for extending, modifying and adapting the existing electrical power supply, control and switch gear system to the LOBI test facility requirements
- Technical acceptance tests and commissioning of the data acquisition and signal processing system
- Preparation and testing of computer programs for data acquisition, handling and evaluation and for process control
- Long-time behaviour tests of the signal processing system (amplifiers, filters)
- Testing of prototypes for components of the various measuring channels (e.g. pressure transducers, dragbodies etc.)
- Technical acceptance tests and calibration of the various measuring channels delivered (pressure, temperature etc.)
- Fabrication of fluid temperature probes
- Experimental investigation of signal disturbances and theoretical considerations on signal analysis
- Theoretical considerations and code calculations on two-phase pump characteristics , containment back-pressure simulation, two-phase break nozzle calibration, thermal stresses in reactor model pressure vessel, forces on primary loop structures
- Evaluation of downcomer flow resistance tests and preparation of LOBI-loop characteristics data set
- Code calculations for CSNI standard problem 3, survey calculations for programme B, development of untility programs for RELAP4.

6. Results and project status

Fabrication difficulties (e.g. shrink holes in the pump housing, etc.) and delivery delays from subcontractors led to a 4 months delay in the completion of the pump fabrication. Their factory technical acceptance tests started in December 1976 and showed the unobjectionable operation of the pumps itselves up to the maximum admissible impeller speed of 8500 rpm. The tests had to be interrupted for eliminating disturbances, the source of which could finally be localized in the pump drive control. The acceptance tests will be concluded in January 1977. The final and precise electrical adjustments of the pump drive plants has to be done later under real operation conditions.

Fabrication difficulties have been encountered for both, the reactor model pressure vessel and the upper power connecting plate. High pressure vessel material strength properties connected with too high brittleness required change of material, which led to larger wall thickness and thereby higher thermal stresses. Delivery delay for the new material caused a corresponding fabrication delay for the pressure vessel.

Soldering difficulties due to the sandwich design of the upper power connecting plate necessitated design modifications and led also to strong fabrication and delivery delays.

Several differential pressure transducer types have been tested before the choice has been made for the one best suited for our purpose and to be purchased.

The two first dragbody prototypes, designed and fabricated for us by the Battelle Institute, Frankfurt (FRG), in the framework of a R&D contract from the BMFT-Bonn, have been successfully tested under steady-state operation conditions and shown that the required specifications are satisfied well with exception of one case, where the temperature influence on the zero stability was inadmissibly high, and therefore this dragbody had to be re-shipped for repair.

A prototype of the water-cooled stand-off pipes, for connecting the pressure transmission line from the differential pressure transducers to the pressure taps, has been tested under operation conditions previous to the release for fabrication of the total number of stand-off pipes required.

The absolute pressure measuring channels, consisting of transducers, signal lines and amplifiers with filters, have been delivered and tested. The results obtained confirm the required specifications and the total error of a complete channel amounts to 1,2 % full scale at most.

The resistance thermometers have been calibrated after delivery; the measuring precision is of o,15° C at 3oo° C.

The amplifiers for all temperature measuring channels with thermocouples have been subjected to long-time tests for determining their zero drift, which amounts to o,1° C and o,5° C for fluid and heater rod temperature channels respectively during 7 days; these are admissible values, they require however a calibration before each blowdown experiment.

The experimental investigation of disturbances on the heater rod thermocouple signals under operation conditions have shown, that the electrical screening of the thermocouples has to remain closed over whole the length of the signal line; only then the total interference voltage corresponds to about $\pm$ 1° C at constant heater rod current and increases to about $\pm$ 4° C at stepwise decrease of the heater rod current.

Theoretical studies on the measuring signals from the γ-densitometers were concerned with the setting up of physical models required for the interpretation of these signals with respect to the influence of different two-phase flow regimes.

Code calculations had to be done for determining the thermal stresses in the reactor model pressure vessel wall under transient temperature conditions; these calculations became necessary, after the wall thickness had to be increased due to the change of material. The maximum thermal stresses to be expected during ECC water injection amounts to less than twice the yield strength and is therefore admissible (ASME regulation).

A physical model has been developed for using the break nozzles, which have to be inserted into the break tube of the test facility for adjusting the break cross section to various sizes, for the determination of the two-phase break mass flow during blowdown. For the calibration of these nozzles, the test parameter ranges and a test matrix have been established.

Code calculations (by the LRA-Garching) have been started for determining the containment back pressure history during blowdown in both, the reactor and the experimental containment. The results are required for determining the amount of apparatus for pressure regulation and control in the experimental containment.

Theoretical studies were concerned with the pump behaviour under two-phase flow conditions, with the aim to set up an appropriate test matrix for the forthcoming investigations to establish the two-phase LOBI pump characteristics.

The evaluation of the downcomer flow resistance test results has shown, that this flow resistance will be lower in the experimental plant than in the reactor plant. Therefore special inserts will be necessary for adjusting this flow resistance.

The LOBI loop characteristics data have been determined, including single phase pump characteristics. This data set is used as base for the blowdown code survey calculations to uncover the most sensitive test parameters. These calculations have been started by the project staff with the RELAP4 code. The final data set will be established on the basis of the results from the preliminary LOBI tests.

DAPSY code calculations have been started by the LRA-Garching to determine the hydraulic forces on the loop structures during the early blowdown phase. These forces are required for calculating afterwards the mechanical stresses within the loop structures with the STRUDL code.

RELAP4 code calculations have been done for the CSNI standard problem 3 in the framework of a workshop exercise, where the calculation results of different participants and codes have been compared with each other.

Several utility programs for the RELAP4 code have been developed with the aim to facilitate the use of this code; they are concerned e.g. with an easy adaption of the program size to an actual task case or with very versitile plotting possibilities of the results.

The actual status of the project work can be summarized as follows:

The previous planning of the mounting works had completely to be changed due to several and considerable delivery delays for various components.

The mounting works have been started with three months delay due to delay in the completion of the construction works in the laboratory hall on one hand and to delay in the mechanical loop components delivery on the other hand.

The six months delivery delay of control and regulation components (valve, etc.) caused a second step in the loop mounting.

Considerable difficulties in the fabrication of the reactor model pressure vessel, of the upper power connecting plate and of the pumps led to strong delivery delays of these parts and caused the introduction of a third step in the loop mounting, to be done during April and May, 1977.

Therefore the completion of the mounting phase of the project planning is now scheduled for July, 1977. The commissioning of the loop system will be started thereafter.

7. Next Steps
- Completion of test facility mounting
- Commissioning of the test facility.

8. Relation with Other Projects
See previous annual report.

9. Reference Documents
- Quarterly Reports of 1976, IRS-F-3o to 34
- W. Riebold: Two-Phase Measuring Techniques in Depressurization Experiments. Conference paper to the 1976 Meeting of the European Two-Phase Flow Group, Erlangen, 31st May - 4th June, 1976
- W. Riebold, W. Hufschmidt, M. Larsen: Ispra Studies in the Field of LWR-LOCA. Conference paper to the ANS/ENS International Conference on World Nuclear Power, Washington D.C., November 14 - 19, 1976. Transactions of ANS, Vol. 24; 438 - 439, (1976)
- W. Kolar, W. Brewka: REL4UPD and REL4AUTO - two utility programs for RELAP4. External Report EUR-5689, 1976
- W. Kolar, M. Lolk Larsen, L. Piplies: Calculations for the Standard Problem 3 using RELAP3, RELAP4, and RELAP-UK. Conference paper to the Second CSNI workshop on LOCA Standard Problems, Paris, Dec. 6 - 9, 1976
- F. Wind: Fehleranalyse für eine Cs-137-Strahlenabsorptions-Dichtemeßanlage zur Bestimmung der Dichte in einem Wasser-Dampf-Zweiphasengemisch. Externer Bericht EUR-5645 d, 1976.

1o. Degree of Availability of the Reports
- Quarterly Reports: from IRS-Köln, Glockengasse 2, 5 Köln 1.
- All Conference Papers and External Reports: from authors

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.77 - 31.12.77	1.1.1	RS 0016 B

Vorhaben/Project Title	
Vorgänge bei der Druckentlastung wassergekühlter Reaktoren. Modellversuche mit einem 11,2 m hohen Stahlbehälter mit Einbauten Investigation into the Phenomena Involved in the Depressurization of Water-Cooled Reactors. Experiments Using a Steel Vessel 11.2 m in Height with Internals.	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: BATTELLE-INSTITUT E.V. Frankfurt am Main Abt. Energietechnik

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
July 15, 1972	April 30, 1979	Dr. T.F. Kanzleiter
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	December 1977	

1. **General Aim**

The experimental blowdown program is aimed at integral large-scale experimental simulations of loss-of-coolant accidents in water-cooled reactors of PWR and BWR type. All experimental results are to 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**

In the main experiments the loads on reactor vessel internals under BWR and PWR conditions and the phenomena in the discharge nozzle during the initial phase of blowdown are to be investigated. Preliminary experiments without internals are to be performed to show the influence of the internals on the discharge process.

3. **Research Program**

3.1 Preliminary LOCA experiments with a pressure vessel without internals under PWR and BWR conditions.

3.2 PWR experiments part I with "flexible" internals of PWR type.

3.3 BWR experiments with internals of BWR type.

3.4 PWR experiments part II with "flexible" and "inflexible" internals.

4. **Experimental Facilities, Computer Codes**

The experimental facilities consist essentially of

- a pressure vessel (5.2 m^3, 140 bar, 300 °C) with an electric heater (600 kW max.)

- PWR internals with flexible and inflexible components
- BWR internals
- a loop with piping, pumps and heat exchangers to realize the same differences in enthalpy as inside an original vessel
- measuring instruments (approximately 110 channels for pressure, differential pressure, temperature, density, mass flow, force, strain, acceleration, displacement and water level
- a data collecting and processing system with 120 channels and a threshold frequency of 5 kHz.

For comparison with the experimental data, several computer codes are used by Battelle (see research project RS 312) and by external institutions. Some of these computer codes are also being used for licensing procedures.

5. Progress to Date

Ad 3.1 Four preliminary BWR and PWR experiments without internals (Nos. SWR 1R, SWR 2R, DWR 1R and DWR 2R) were carried out in 1974 and 1976.

Ad 3.2 Seven PWR experiments with flexible internals (Nos. DWR 1 to 5, DWR 2L and DWR 5A) were carried out in 1975. Evaluation and model calculations of these experiments were completed in 1977.

Ad 3.3 Eleven BWR experiments with internals were planned and partly prepared in 1976 and 1977.

6. Results

Ad 3.3 For the planned BWR-experiments with internals expecially developed differential pressure transducers for direct installation inside the pressure vessel have to be provided. Quality inspection tests, carried out with a newly developed type of transducer, showed that this transducer is suitable for the specified pressure and temperature conditions (80 bar/ 290 °C). These tests indicated also a tolerable measuring error caused by pressure and temperature transients anticipated for BWR blowdown experiments. On the other hand, the tests showed unsuitable dynamic characteristics of the transducers:

- undamped oscillations after initiation by acceleration or pressure shocks
- eigenfrequencies at values far below the specification.

Summing up, the quality inspection tests lead to the conclusion that the tested transducers are not suitable for measurements inside a blowdown vessel and that alternative measuring systems must be provided for the BWR blowdown experiments.

7. Next Steps

Ad 3.3 Providing other differential pressure transducers or planning an experimental BWR-blowdown program for experiments without differential pressure transducers.

Ad 3.4 Preparing PWR blowdown experiments.

8. Relation to Other Projects

RS 312

9. References

(1-4) Quarterly Reports in the Series "GRS-Fortschrittsbericht. Bericht über die vom Bundesministerium für Forschung und Technologie geförderten Forschungsvorhaben auf dem Gebiet der Reaktorsicherheit" (in German).
- January to March 1977
- April to June 1977
- July to September 1977
- October to December 1977

(5) BF-RS 0016 B-10-1: "Ergebnisse der ersten DWR-Versuche mit Einbauten DWR1 bis DWR5)", September 1977

(6) BF-RS 0016 B-21-5: "Beschreibung des Programmpakets RS DV zur Meßdatenverarbeitung", January 1977

(7) BF-RS 0016 B-33-1: "Versuchsergebnisse zum Gemischspiegelverhalten in einem 11,2 m hohen Behälter ohne Einbauten nach einem Dampfleitungsbruch (Versuch SWR 2R)". March 1977

10. <u>Degree of Availability of the Reports</u>

Reports are available through GRS-FB.

Documents (5) to (7) can be made available only by special agreement.

Title 1 (Original language) PROGRAMMA P.I.P.E.R.: esperienze di blow-down in presenza di strutture interne.	Classification $\boxed{1.1.1,}$ 1.1.2
Title 2 (English) Blow-down Tests by Piper apparatus-experiments with internal structures.	Country ITALY Sponsor CNEN-CNR Organisation University of Pisa
Date initiated 1972 Date completed 1978 Last updating 1977	Project Leader P. VIGNI

1) General aim
The program is intended to study basic blow-down problems and to analyse causes of possible disagreements between experimental results and RELAP calculations, with particular reference to the transfer of model data to full scale plants.

2) Particular objectives
Tests are intended to reproduce in real time the thermohydraulic transient in a vessel during blow-down, without internals or with structures simulating the internal geometry of a B.W.R.
The tests have also the purpose of investigating the mechanical effects of the pressure transients on structures assembled inside the test vessel.

3) Experimental facilities and program
PIPER apparatus is a pressure vessel equipped with an electric device, rupture disk assembly and instrumentation for measurements of pressure and temperature transients. Six measurement points are available along the height of the vessel.
The main design features of vessel are:
- pressure: 110 Kg/cm^2
- temperature: 310 °C
- internal height: two values can be used (1,8 and 3 m)
- internal diameter: 0,194 m
- outlet nozzles: two, both of which have diameter of 50 mm and length of 400 mm
- electrical device: it consists of three heater rods, designed to produce a total power of 24 Kw, located at the vessel bottom.

4) Project status
Up to date a set of about 40 blow-down tests were carried out.
Starting pressure was varied from 2o up to 70 Kg/cm^2 with the same initial water level. Blow-down was operated from either water or steam zone through openings sharp and rounded-edged of 13,6 mm, 14,8 mm and 50 mm diameter with or without internals.

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Title 1 (Original language)	Classification
PROGRAMMA P.I.P.E.R.: esperienze di blow-down in presenza di strutture interne.	1.1.1, 1.1.2

5) Next steps
The tests in the next future concern blow-down with internal plate provided with a sharp-edged circular orifice, at starting pressures from 20 up to 70 Kg/cm^2, and measurements of transients of liquid level and stresses on internal plate.

6) Relation to other projects
The research is strictly connected to the program SOPRE 1 (Pressure suppressione LOCA).

7) 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.

4. P. VIGNI et alii
Programma di ricerca sull'incidente di perdita di refrigerante nei reattori nucleari ad acqua (P.I.P.E.R.).
Efflusso di miscele acqua-vapore, inizialmente allo stato saturo, attraverso piccole aperture.
Istituto di Impianti Nucleari, RL 205(75).

8) Degree of availability
The previous references are free; the next ones may be available with the autorisation of the CNEN.

Title 1 (Original language) Analisi dei transitori termici ed idraulici a seguito di LOCA nei reattori ad acqua leggera.	Classification $\boxed{1.1.1,}$ 1.1.2, 1.1.4, 1.2
Title 2 (English) Analysis of thermal and hydraulic transients following a LOCA in Light Water Reactors	Country ITALY Sponsor CNEN and CNR Organisation University of Pisa
Date initiated 1974 Date completed 1978 Last updating may 1977	Project Leader N. CERULLO

1) General aim

The program has the purpose of investigating thermal and hydraulic transients following LOCAs on Light Water Reactors. The aim is to achieve a keener understanding of some aspects of the blow-down physical phenomena, and to improve some features of blow-down codes.

2) Particular objectives

Extensive work has been carried on regarding the WREM codes and the blow-down and heat-up codes.
These codes have been used to analyze:
- LOCA Standard Problems, proposed by NEA-CSNI; the results obtained have been presented at the second NEA-CSNI workshop;
- results of experimental programs performed by the "Istituto di Impianti Nucleari" at the Scalbatraio Center, University of Pisa.

3) Facilities

IBM 370/158 and 370/168 Computer belonging to CNUCE, Pisa.
The experimental small scale facility PIPER of Scalbatraio Center, University of Pisa.

4) Next step

The next step will be the use of the RELAP4-Mod.5 computer program and its application on some of problems mentioned above.

5) Relation to other projects

Blow-down tests by PIPER APPARATUS - Project Leader P. VIGNI.

6) References documents

1. N.CERULLO,F.ORIOLO, U.ROSA, R.SANI, P.VIGNI
Blow-down activity performed at the Scalbatraio Center of the Pisa University: comparison between experimental results and RELAP3 calculations.
Meeting on Computer program for the analysis of certain problems in thermal reactors safety. NEA C.P.L. ISPRA, 23-24-25 October 1974.

Title 1 (Original language)	Classification
Analisi dei transitori termici ed idraulici a seguito di LOCA nei reattori ad acqua leggera.	1.1.1, 1.1.2, 1.1.4, 1.2

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 miscela acqua-vapore inizialmente allo stato saturo.
29° Congresso Nazionale A.T.I. - Firenze 25-27 Settembre 1974.
3. K.V.MOORE,W.H.RETTIG
RELAP 4 - A Computer Program for Transient Thermal Hydraulic Analysis, ANCR 1127 Rev. 1, March 1975.
4. WREM: Water Reactor Evaluation Model - NRC - May 1975.
5. 1½ LOOP SEMISCALE SYSTEM - Aerojet Nuclear Company - 1975.
6. N.CERULLO, L.CINOTTI, G.DEL NERO, F.ORIOLO
La catena di programmi di calcolo RELAP - Theta 1-B:
Analisi del transitorio termico di un PWR in seguito a LOCA.
RP 245(-76) - Istituto di Impianti Nucleari - Università di Pisa.
7. N.CERULLO, G.DEL NERO, G.GIRESINI, F.ORIOLO, F.VITALITI
Results of Calculation of NEA - STANDARD PROBLEM 4 USING RELAP 4--002 COMPUTER PROGRAM
Presented at the second NEA-CSNI workshop, held in Paris on 6-7-8-9 Dec. 1976, on LOCA STANDARD PROBLEMS

Classification
$\boxed{1.1.1}$ (1.1.2.)

Title 1	Country : JRC
Untersuchung des thermodynamischen Ungleichgewichts	Sponsor : BMFT and CEC
	Organization : JRC ISPRA Establishment
Title 2	Project leader :
Investigation of the thermodynamic non-equilibrium	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 vapour 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 :

Netherlands Energy Research Foundation (ECN)

CLASSIFICATION : 1.1.1./1.1.2.

TITLE :

Mechanisch gedrag van het reactorbinnenwerk tijdens grote ongelukssituaties

COUNTRY: THE NETHERLANDS

SPONSOR : ECN

ORGANIZATION : ECN

TITLE (ENGLISH LANGUAGE):

Mechanical behaviour of reactor internals during major accident situations.

PROJECTLEADER : L.H. Vons

SCIENTISTS :
H. van Rij
L.G.J. Janssen

INITIATED : 1977

LAST UPDATING : May 1977

STATUS : in progress

COMPLETED : 1980

General aim

To increase the knowledge of the mechanical behaviour of the reactor internals during normal operating conditions and in particular during major accident occurrence.

Particular objectives

- The evaluation of a "safe" shut-down of the reactor during postulated design accidents such as Loss Of Coolant Accidents.
- The deformation in the reactor internals immediately following a LOCA.

Experimental facilities and program: -

Project status

Several calculations were performed using special elements of the NASTRAN-computer-code to demonstrate that the program is capable to handle contact problems.
The results of these calculations show that the NASTRAN-computercode can handle inpact-phenomena between the construction parts involved (i.e. fuel elements, grid-plates etc.) satisfactorily.

Next steps

During the fall of '78 calculations will be performed on an arrangement of one fuel element and connecting structural parts to demonstrate that a more complex system can be handled by the NASTRAN-program as well.
Depending on the results of these calculation, a simulation of a complete reactor-internal-system will be analyzed.
However this calculation will only be executed provided that reliable values of the time dependent pressure history during major accident (i.e. LOCA) situations are available.

Relations to other projects

Related studies at ECN: Reactordynamics and thermo-hydraulic study.

Reference documents: not yet available.

Degree of availability: N/A.

Budget: Computer cost 1978 + 1979 ~ US $ 20.000.

Personnel: 1978 + 1979 : 0,6 manyear.

81.

CLASSIFICATION 1.1.2	
TITLE 1 BLOWDOWN HEAT Transfer Test Program	COUNTRY Belgium (U.S.A.) SPONSOR ORGANIZATION Westinghouse EPRI
TITLE 2	PROJECT LEADER SCIENTISTS
INITIATED (date) — COMPLETED: End 1976 STATUS: Progressing — LAST UPDATING: NA	

1.0 General Aim

The general objective of the Blowdown Heat Transfer Test Program is to obtain experimental data to determine the key heat transfer parameter during the early stages of a PWR Loss of Coolant Accident, up to and following operations from nucleate boiling (DNB). This experimental data will be utilized in the development of transient DNB correlations for use in ECCS performance analyses.

2.0 Particular Objectives

a) Controlled Parameter Tests - Phase I

The objective of these tests is to obtain data from which transient DNB heat transfer correlations can be developed. This objective is to be accomplished through a series of single parameter tests which impose controlled thermal/hydraulic transients on the test bundle. The proposed range of initial and vamped conditions are expected to provide the data base necessary to conclute the occurence of DNB over a range of conditions applicable to plant LOCA transients.

b) System Response Test - Phase II

The objective of these tests is to obtain data in this facility which demonstrates that DNB does not occur during the early core flow reversal period which is calculated upon a large double ended cold leg break in a PWR. The DNB heat transfer correlation developed in the PHASE I testing will be subsequently verified in the PHASE II tests.

3. Experimental Facility

The Blowdown Heat Transfer Test Facility is shown in Figure 1.

The test facility consists of :

a) A primary loop in which water is circulated to preheat the test vessel and other components to operating temperatures.

b) An auxiliary system in which water is blowdown from the flash chamber through the test bundle under conditions which simulate a PWR LOCA.

c) A 12 foot long test bundle, consisting of 25 heater rods in a 5 x 5 array. The bundle axial power shape is skewed to the bottom with a non-uniform radial power profile. The heater rod instrumentation includes 12 clad thermocouples and 8 element thermocouples.

The range of initial and vamped conditions are :

a)	Initial Heat Flux :	10 Kw/ft - 10 Kw/ft
b)	Initial Mass Flux :	$2. \times 10^6$ - $3. \times 10^6$ lbm/hr ft^2
c)	Initial Bundle Inlet Temperature :	560°F - 600°F
d)	Initial System Pressure :	1750 - 2250 PSIA
e)	Depressurization Rate :	0 - 350 PSI/SEC
f)	Flow Decay Rate :	0 - 2.5×10^6 lbm/hr ft^2/sec.

4. Project Status

Six tests have been conducted and a preliminary report issued to EPRI. An evaluation report will not be issued until December, 1976.

5. Near Term Plans

Approximately 10 additional tests will be conducted in the period June 1, 1976 to August 31, 1976 to further investigate initial conditions parameter vamps, and flow direction.

6. Relation To Other Programs

This program is indirectly related to other development programs (e.g., FLECHT, two-phase pump tests, etc.) aimed at improving LOCA analysis models.

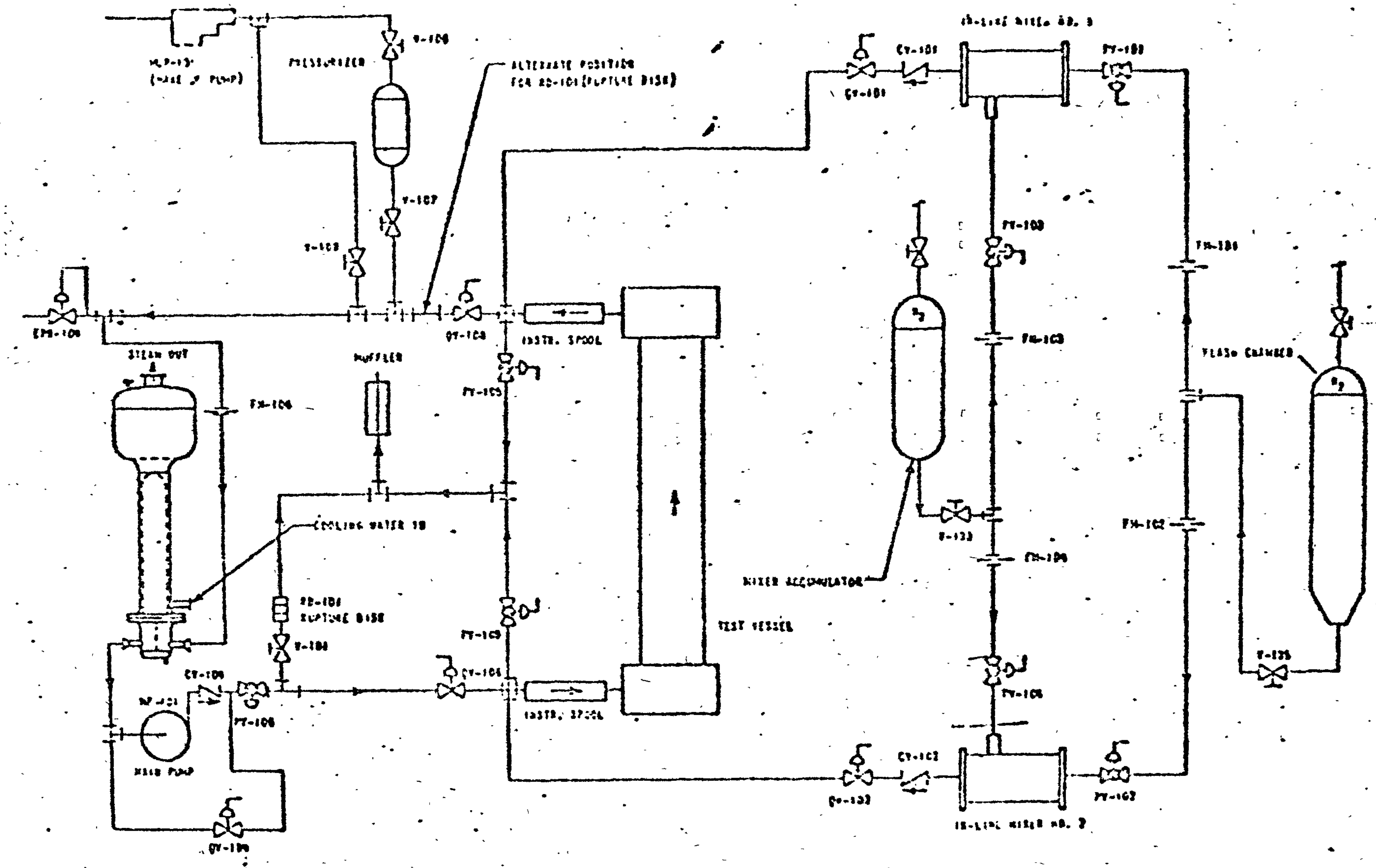

FIGURE 1 : BLOWDOWN HEAT TRANSFER TEST LOOP SCHEMATIC

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.77 - 31.12.77	1.1.2	RS-109

Vorhaben/Project Title	
Einfluß der DWR-Umwälzschleifen auf den Blowdown (LOBI-Projekt) Loop Blowdown Investigations (LOBI)-Project: Influence of PWR primary loops on blowdown	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Kommission der E.G. G.F.S.-Ispra Commission of the E.C. J.R.C. Ispra Establishment

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. Dez. 1973	30. Nov. 1981	Dipl.-Ing. W. Riebold
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Mounting phase completed	Dez. 1977	20,7 Mio. DM

1. General Aim

Design and construction of a large scale two-loop blowdown facility.

Performance of loss-of-coolant experiments (LOCEs) by simulating tube ruptures of different sizes at several locations within a PWR primary cooling circuit system.

2. Particular Objectives

Experimental investigation of the role of the different components of a PWR primary cooling circuit during a blowdown by the measurement of the main thermohydraulic quantities, especially those which influence the core cooling, i.e. the flow and heat transfer conditions and the pressure differences.

The experimental results will be applied to check and improve the blowdown codes and associated theories used for the safety analysis of LWRs.

3. Research Programme

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

Programme A, to be performed for the BMFT-Bonn in the framework of the R&D contract RS-109/143-73-PIHOD, concluded between the BMFT-Bonn and the C.E.C., will be concerned with the investigation of the influence of the following parameters on the blowdown:

- rupture size and location
- pumps operation performances

- initial power level
- heating-power time-function during blowdown
- strenght of heat sink (steam generator secondary side conditions)
- downcomer resistance and volume
- ECC water injection locations.

The preliminary test matrix A, comprising 60 tests previously defined by a German Expert Group has to be revised for being subdivided into two parts A 1 and A 2 of 30 tests each, and for accounting for several modifications applied so far. For the test matrix A 2, suggestions from the Community Member Countries will be taken into account, and the experimental results obtained from A 2 will be freely available to the Community Member Countries.

Programme B, to be performed for the Commission of the E.C. after conclusion of Programme A 2, will be concerned with the

- performance of some reference tests (repetition of tests of programme A) which at the same time constitute reproducibility tests
- performance of component studies, to be done with this test rig after having modified certain components; the purpose of these tests is to investigate the influence of the geometrical shape or the elevation of theses components on the blowdown.

Seven such modifications of the programme A test rig have already been agreed upon by the "ad-hoc Working Group for the LOBI Programme B" composed of expertes from the Community Member Countries:

- variation of the depth of the loop seal (U-tube between the steam generator and the pump) in the intact loop,
- variation of the steam generator elevation in the intact loop,
- variation of the lower plenum (higher l/d ratio),
- two separate accumulators, one for each loop, instead of one accumulator for both loops,
- simulation of a primary tube rupture within the steam generator (of the broken loop),
- simulation of a small rupture within the lower plenum

- ECC water injection into the upper plenum.

The funds from the Commission's budget, necessary for these modifications, had been allocated to the LOBI-project budget in the beginning of 1975 and enabled orders to be placed for these modifications together with the orders for all mechanical loop components.

An appropriate test matrix B is actually being elaborated by the before mentioned "ad-hoc W.G. LOBI B" on the basis of two proposals submitted so far from two member countries.

4. Experimental facilities, computer codes

A 4-loop primary cooling 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 representing the broken "reactor" loop. Both experimental loops are active loops containing a pump and a steam generator each. Fig. 1

Tube ruptures of various rupture sizes (from double ended down to small leak) are to be simulated at three different locations within the broken loop (hot leg, cold leg, loop seal).

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,
- 21 kg/s and 7 kg/s fluid mass flow in the intact and broken loop respectively,
- about o,7 m^3 volume content of the primary loop test system.

The loop system and the components have been designed for 16o bar and 325^o 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 the components
- the elevations of the components
- the lengths of the heat transfer surfaces (core rod bundle, steam generators)

equal to the corresponding reactor values.

Two accumulators (6o bar and 3o^{o} C operating pressure and temperature respectively) of different volume content (28o and 95 dm^3) for the two loops are providing ECC water for both, separate and combined cold leg and hot leg injection into both loops.

A secondary loop system provides heat removal from the steam generators in the primary loops; it operates at 52 bar system pressure and in a temperature range between 21o^{o} and 27o^{o} C (steam generator secondary side inlet and outlet temperature respectively).

The measurement of fluid absolute pressure and pressure differences, absolute temperature and temperature differences, mass flow and density will be done at the boundary of all loop components, where special tube inserts, called spool pieces, being instrumented with appropriate measuring devices, are mounted into the loop tubing. The same fluid quantities will be measured also within the reactor model region (downcomer, core, lower and upper plenum).

Furthermore the outer surface temperature of the heater rods will be measured at 188 positions equally distributed over the heated bundle region.

Test facility design calculations have been done by the LOBI project staff with the codes RELAP4/MOD2, BERSAFE and STRUDL-II, and by the LRA-Garching (now GRS-Munich) with the codes BRUCH-D, DAPSY and ZOCO-VI.

The LOBI group "Programme & Analysis" has performed the reference calculation with RELAP4/MOD2 and RELAP4/MOD5; it will perform an essential part of the survey calculations, too. Both types of code calculations are aimed at generating a basis for defining the test matrix B and contributing to the test matrix A 2.

Pre-prediction calculations for the test matrix A 1 and A 2 will be performed by the LOBI group "Programme & Analysis" and by the GRS-Munich and -Cologne.

Special R&D contracts have been concluded by the Commission of the E.C. covering the "LOBI pump characteristics investigation in two-phase flow conditions". These characteristics are necessary to determine the time function for the LOBI pump speed control during blowdown required for obtaining a pump behaviour similar to that of the reactor pump during a reactor blowdown. They are furthermore necessary for the evaluation of the LOBI test results with respect to the influence of the pump behaviour on blowdown.

The contract work has started in September 1977.

Convergent-divergent nozzles will be inserted into the outlet branches of the "broken" LOBI loop for simulating different rupture sizes for the blowdown tests. These nozzles are intended to be used at the same time for determining the break mass flow from the measurement of pressures, temperature and fluid density. The therefore required calibration of these nozzles under two-phase flow conditions will also be done in the framework of an R&D contract to be concluded by the Commission of the E.C.. The corresponding preparation for the test programme and the contract are under way.

5. Progress to Date

During the report period, the mounting of the LOBI test facility has been completed and the operation tests of different elements and components of the loop regulation and control system have been started. More in detail, the following works have been performed:

- Conclusion of several R&D contracts by the Commission of the E.C. for the investigation of the LOBI pump characteristics under two-phase flow conditions, and preparation of the experimental test matrix and results evaluation methods
- CSNI-Standard-Problem-3 calculations
- Theoretical investigations in two-phase flow through convergent-divergent LOBI break nozzles and preparation of an experimental programme for the nozzle calibration tests and of the corresponding R&D contract
- Preparations for STRUDL-II-code calculations to determine the dynamic loads on the LOBI loop system and scaffolding originating from hydraulic forces during blowdown, calculated with

DAPSY-code by the GRS-Munich/FRG

- BRUCH-D- and ZOCO-VI-code calculations by the GRS-Munich/FRG to determine the pressure history in the LOBI containment tank during blowdown
- Performance of reference calculations with RELAP4/MOD2- and /MOD5-code and preparation of survey calculations for generating basic data for the determination of the programme B test matrix
- Modifications of subroutines of, and extension of utility programmes for the RELAP4/MOD5 code.
- Completion of mounting of mechanical parts of the loop system after strongly delayed delivery of pumps and reactor model pressure vessel and power connecting plate
- Modifications in the drive system and factory acceptance tests of the third LOBI pump foreseen for the pump tests
- Calibration of the heater rod bundle spacers, assembling of the commissioning heater rod bundle and of the reactor model, and mounting of the last one into the loop system
- Loop regulation & control system, 5,5 MW rectifier system and general power supply system:
  . completion of mounting and of cable connecting work
  . performance of operation tests
- Preparations for starting the LOBI facility commissioning tests
- Acceptance tests and calibration of measuring devices delivered
- Continuation of measuring devices prototype testing and of model development for signal analysis
- Commissioning of a small laboratory scale water-air two-phase flow test loop for turbine flow meter testing and calibration
- Performance of comprehensive tests with a completely instrumented LOBI spool piece in the joint test loop of the KfK (Kernforschungszentrum Karlsruhe, FRG)
- Design of an extended parameter range water-air two-phase flow test loop
- Information visit to the INEL of the EG&G-Idaho Inc., Idaho Falls, USA, with a view on Semiscale and LOFT measuring instrumentation system and development
- Continuation of development and testing of mini-computer pro-

grams for the LOBI-D.A.S. for data acquisition and handling, data evaluation and display, and for process control (loop operation, heating power control, pump speed control).

6. Results

The results obtained from the LOBI project activities are reported separately for the five different activity fields of the project.

6.1 Programme and Analysis

After the supplementary contract between the FRGMRT-Bonn and the C.E.C. for a 4 years prolongation of the main LOBI R&D contract had been signed, a further supplementary contract has been concluded between both partners covering the "LOBI pump characteristics investigation in two-phase flow conditions". This investigation has to be performed in the framework of two R&D contracts placed by the C.E.C. with the Westinghouse Canada Limited (WCL), Hamilton/Canada, and the ASTRÖ-Graz/Austria, respectively. Both contracts have been concluded and are running since September 1977.

The objective of this investigation is to set up the LOBI pump characteristics for two-phase flow conditions. These characteristics are necessary to determine the time function for the LOBI pump speed control during blowdown required for obtaining a pump behaviour similar to that of the reactor pump during a reactor blowdown. They are furthermore necessary for the evaluation of the LOBI test results with respect to the influence of the pump behaviour on blowdown.

The experimental investigation will be performed by WCL-Hamilton/Canada with the third of three LOBI pumps of equal size and performances. ASTRÖ-Graz/Austria as the pump manufacturer will participate in this investigation by giving technical assistance to the experiments and scientific support to the results evaluation and the setting up of the pump characteristics. Finally, the LOBI group "Programme & Analysis" is charged with the definition of the experimental programme, the evaluation of the test results and the setting up of the pump characteristics.

During a visit with the laboratories of Combustion Engineering (CE), Windsor/Connecticut/USA, where similar pump tests are running already since several months, first contacts could be taken in view of a future cooperation in this field. Experiences and problems from the EPRI-CE pump tests with a Byron-Jackson pump have been discussed.

An appropriate multi-dimensional interpolation and extrapolation computer program has been prepared and tested for determining matrix points of the characteristics fields from the measured experimental results. From the evaluation of reports on similar pump investigations (Semiscale pump at WCL-Hamilton/Canada, Byron-Jackson pump at CE-Windsor/Connecticut/USA) useful information has been obtained with respect to the test matrix and the measuring instrumentation for the LOBI pump tests at WCL-Hamilton/Canada.

The technical acceptance tests of the third LOBI pump and ancillary system in the manufacturer's factory have been concluded successfully; the pump and ancillary system will be shipped to WCL-Hamilton/Canada at the end of January 1978.

The main objective of a visit to the ROSA Project of Japanese Atomic Energy Research Institute (JAERI) at Tokai, Japan, has been a detailed information exchange on both blowdown projects, ROSA and LOBI. Particular interest has been focused on the discussion and comparison of the objective of the experimental investigation, of the configuration and performance data of the experimental facility with a view on the simulation conditions, of the measuring instrumentation and data acquisition system, as well as of organization problems in performing the test and evaluating the results. The main aspect of the results obtained so far from the ROSA I (BWR geometry, 1970 - 73) and the ROSA II (PWR geometry, 1974 - 77) programme have been shortly outlined and discussed either.

During the 8th and the 9th meeting of the "ad-hoc Working Group on LOBI Programme B" (WG-LOBI-B) the following main items have been dealt with

- discussion of a second proposal for the test matrix of programme B (community programme)
- comparison and discussion of the results from the reference calculations[1] performed by 3 member countries and the LOBI group
- agreement upon the initial and boundary conditions for the survey calculations[2]
- agreement upon which tests of the second programme B proposal will be calculated by which participant
- agreement upon the rupture sizes to be considered for Programme B
- discussion of timing and procedure for setting up the test matrix A 2 and B

Further preparation and development work was concerned with the RELAP4/MOD5 blowdown computer code and was aimed at

- the modification of sub-routines and the correction of input data required for the conversion from british units into international units in the RELAP4/MOD5 input and output parts; a report on this is under preparation
- the extension of the plot program ISPLOTR4 and the program REL4UPD on RELAP4/MOD5; this work has been completed and is described in the external report EUR 5919 e [1]
- the introduction of the quantities pump head and void fraction as "plot variable" and "edit variable"; this introduction into the appropriate COMMON blocks for extending the list of the "minor edit" variables was considerably facilitated by using the MOD5UPD program, a modified version of REL4UPD, both developed by the LOBI group "Programme & Analysis".

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1) agreed upon during the 8th meeting and consisting in a joint reference case calculation with the data of the LOBI facility to assure the comparability of the results of the different participants in the survey calculations

2) are aiming at uncovering the most sensitive test parameters to be taken into account for the definition of programme B.

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A new nodalization scheme for the forthcoming survey calculations has been elaborated to ease the comparison of the results from different participants.

The reference calculation has been made with RELAP4/MOD2 and repeated with RELAP4/MOD5; the comparison of the results with those from RELAP4/MOD2 is under way. See Fig. 2 for a typical example.

The different rupture sizes for the blowdown test will be simulated by inserting convergent-divergent nozzles into the outlet branches of the "broken" LOBI loop. The comparison of results from different calculation models with experimental data has shown considerable uncertainties for the prediction of critical mass flows in the range of small vapour qualities ($x < 0,1$) and subcooled water; this emphasizes the need of calibrating the rupture nozzles if they must be used for determining the critical mass flow from the measurement of pressure, temperature and density.

For this purpose, a physical model has been developed based on homogeneous equilibrium flow, describing the mass flow and the flow phenomena inside such a nozzle for different fluid conditions at the nozzle inlet (subcooled liquid, superheated vapour, saturated vapour with variable vapour quality). An extension of this model allows to take into account also thermodynamic non-equilibrium within the fluid during the transition from subcooled to saturated fluid conditions in the course of the nozzle flow. [2]

During a visit with the laboratory of WCL-Hamilton/Canada where the calibration experiments of the LOBI break nozzles are presumed to be performed after completion of the LOBI pump tests, the discussion of the first preliminary test matrix has shown that due to the operation limits of the test facility this matrix has to be changed. The required calculations have been performed with a computer program developed by the LOBI group "Programme & Analysis" allowing the calculation of homogeneous equilibrium flow of two-phase water-vapour mixture through smooth nozzles. The results have revealed the necessity to subdivide the experimental nozzle calibration programme into two parts:

- steady-state tests with a closed loop for a maximum nozzle inlet pressure of 70 bars
- transient tests with a blowdown vessel in the pressure range of 50 - 120 bars.

Agreement upon this new program has already been obtained with WCL-Hamilton/Canada. It has, however, still to be investigated whether transient tests for subcooled fluid conditions can be included into the test programme. WCL will submit a first tender for these calibration tests in January 1978.

The thermohydraulic non-equilibrium experiments performed in Ispra (RS-77, 1972 - 74) are at present being evaluated with the aim to develop a correlation for the evaporation velocity as function of pressure, degree of super-heating and void fraction of the fluid.

The break mass flow during blowdown becomes dependent on the containment pressure as soon as the pressure ratio over the break nozzle exceeds the critical pressure ratio. This occurs in the late blowdown phase, when the system pressure has sufficiently decreased related to the increased pressure in the containment. Since the containment tank of the LOBI facility for technical reasons could not be designed according to the volume scaling factor nor maintaining the surface to volume ratio equal to the reference plant value, the simulation of the containment pressure history during this late blowdown phase requires a special pressure regulation device. For getting more precise specification data for the necessary performances of such a device, the GRS-Munich/FRG has been charged with ZOCO-VI-code calculations for the determination of the pressure history within the LOBI containment tank. These calculations have been completed and first results have been communicated verbally; a report on the calculation results is under preparation.

The internal surface temperature of the LOBI containment tank is strongly influencing the containment pressure history during blowdown. A rough estimate based on energy balance considerations and results from computer code calculations with a transient heat conduction program yielded internal wall temperature values at the blowdown start of about 6o - 7o° C. [4]

Results from the DAPSY-code calculations of the GRS-Munich/FRG have shown that the dynamic loads on the LOBI facility resulting from hydrodynamic forces within the initial blowdown phase are mainly caused by pressure forces; friction forces are of minor importance. A special computer program is under development at Ispra for the elaboration of these DAPSY-code results in view of their application as input data forSTRUDL-II-code calculations to determine the mechanical stresses within the LOBI test facility and the dimensions of special supports against the scaffolding.

6.2 Mechanical Components & Systems

The mounting work of the LOBI facility has been continued according to the delivery of the various components and could be completed at the end of 1977, Fig. 3. Severe delivery delays, mainly caused by technical difficulties on the manufacturers side, have forced several changes in the mounting time schedule.

Two of the three LOBI pumps have been delivered in February 1977 with a 5 months delay. The third pump will be shipped to WCL-Hamilton, Canada, for being used there to investigate the pump operation characteristics under two-phase flow conditions. The factory acceptance test of this pump could be done after the modifications of the drive control system, required for adaptation to Canadian power supply conditions, had been completed.

The special design of the pump shaft bearings and seals has necessitated the splitting up of the main coolant pumps commissioning into two steps. Therefore the LOBI facility commissioning procedure had to be modified. The cold pressure and seal tests of the mechanical loop components will be performed with the pumps impeller withdrawn and the pumps housing closed by a blind flange. The pumps commissioning will then be done together with the warm pressure tests of the loop system including coolant circulation.

The factory acceptance tests of the pressure vessel, and the power connecting plate for the heater rod bundle of the reactor model, have finally been performed in March 1977, after a series of major difficulties had been overcome.

Because the spacers (Fig. 4) for the heater rod bundles had been delivered without being calibrated, the assembling of the commissioning heater rod bundle (Fig. 5) could be performed only in October 1977. Due to the faulty delivery of the spacers we have been obliged to design and fabricate a special calibration device at Ispra and to perform the calibration in our laboratory.

The reactor model has been assembled in November 1977, but the mounting into the loop system has to be postponed until having dried out the heater rod bundle thermocouples. This supplementary operation has become necessary due to a faulty sealing of the thermocouple end by the manufacturer which caused a considerable decrease of the thermocouple isolation resistance.

The preparation for commissioning the mechanical part of the loop system could be concluded in December 1977 allowing the start of the commissioning tests in January 1978.

Contacts have been taken with the A.N.C.C., the italian technical acceptance authority, in view of obtaining their licence for the LOBI facility operation.

6.3 Electrical Components & Systems, Loop Regulation and Control

The installation and mounting work for the 5,5 MW rectifier system and for the components and elements of the loop regulation & control system has been completed as well as the required cable connecting work within both systems.

The central switchboard for the loop regulation and control system, and the high current conducting copper bars for the power connection between the rectifier system and the heater rod bundle has been installed and connected.

The final design of the central switchboard for regulation and control of the LOBI pumps drive and the rectifier power output has been made after the control units of the pumps had successfully passed the factory acceptance tests of the pumps in December 1976. After several months delivery delay this switchboard could be installed and connected in December 1977.

The operation tests of the single elements and components of loop regulation and control system are still under way; the operation tests of the rectifier system for zero power have been started and preparations for the partial power tests have begun.

6.4 Measurement Instrumentation, Signal Conditioning & Analysis

In the field of measurement instrumentation, and signal conditioning and analysis, the work performed was concerned with acceptance tests and calibration of measuring devices delivered, with prototype testing and with model development for signal analysis.

Resistance thermometers will be used for fluid temperature difference measurements and as steady state reference for the thermocouple signals. After the manufacturer had corrected the lack of accuracy (o,5 % FS instead of o,1 % FS) in the measuring transducers of the resistance thermometers, they have been installed at the place provided for them for performing long term drift tests. The measured drift amounting to less than o,1 % or o,4^{o} C for a period of 8 days and a temperature fluctuation of about 5^{o} C can be considered a quite satifactory result.

A small laboratory scale test loop has become available for testing and calibrating of turbine flow meter probes at room temperature conditions in air (1oo m/s) and water flow (7 m/s). Two prototype turbine flow meter probes of Flow Technology Inc., Phönix-Arizona, have been calibrated.

Full flow turbine flow meters, provided for velocity measurements in the surge line and in the ECC injection lines of the LOBI facility have been tested successfully, the results confirming the required specification performances.

After repair by the manufacturer (Battelle-Frankfurt), static calibration has been performed for the two bi-directional and one uni-directional dragbody prototypes for flow momentum measurements.

Calibration restrictors have been developed and fabricated, which will be used to determine the pressure drop to mass flow relationship of both steam generators during the preliminary tests; this relationship will be used lateron during the experiments for determining the steady state loop mass flow from the steam generator pressure drop before blowdown is started.

For the interpretation of the γ-densitometer signals a theoretical model has been developed allowing the calculation of the error band-width to be expected when determining the density for different two-phase flow conditions like annular flow, stratified flow with plane and curved phase interfaces, homogeneous and non-homogeneous phase distribution over the flow cross section. The average density can be determined with sufficient accuracy from the weighed sum of the two single-beam signals. The error obtained cannot be essentially reduced by the use of more sophisticated evaluation models, because the influence of the finite width and of the energy profile of the γ-beam can hardly be taken into account and furthermore, the use of a 2-beam densitometer device does not allow the determination of the average density, the void fraction and the curvature of phase interface independently from each other.

During a second and very comprehensive test series (12o tests) in the joint test loop of the GfK-Karlsruhe a completely instrumented LOBI spool piece has been tested in the whole range of steady-state two-phase flow conditions. The measurement devices installed in the spool piece have been: one of the four two-beam γ-densitometers at present available, 3 fluid and 1 wall temperature probe, 2 absolute and 1 differential pressure transducer, 2 dragbody devices and 2 turbine flow meter devices, Fig. 6.

All pressure and temperature measuring devices have been exposed for the first time to the maximum operation conditions and have operated satisfactorily.

The dragbody devices have been loaded up to 1o % of their rated operation range only and operated well.

The differential pressure transducer has been heated-up and damaged after rupture of the overload burst disk during the 97th test.

The ball bearings of the turbine flow meter probes failed after 5o to 1oo hours operation time; after repalcement of the bearings, the turbine could be used again. The total lifetime of the bearings has proven to be far beyond the value expected for the given operation conditions.

The major part of the test results have been evaluated. The most important quantities measured have been the flow velocity (turbine flow meter), the flow momentum (dragbody) and the "apparent" density (γ-densitometer). Because no reference measurements for these three quantities have been available, the mass flow had to be determined from the measured results assuming homogeneous two-phase flow and had to be compared with the known mass flow (from single phase measurements upstream the mixing chamber of the test loop).

The dragbody measurements do not yield useful information because the momemtum force of the flow has been too small (< 1o %) compared to the operation range of the dragbodies (signals within error band-width).

The mass flow determination from the turbine flow meter signals reveals an error depending on void fraction and probe location; this error has to be taken into account as "calibration curve".

An analysis of the fluctuating part of the signals with respect to intensity (rms-value), amplitude (skewness etc.) and frequency distribution has shown that the flow regime is influencing these quantities in a characteristic way.

The evaluation of the signals from the turbine flow meter and dragbody has evidenced the need of always two such probes in the same flow cross section. Therefore, the LOBI spool pieces have appropriately been modified for allowing this double instrumentation of both devices.

Technical difficulties encountered with the strain gages of the dragbodies (temperature shock depending zero drift) have delayed the ordering of further devices until a satifactory solution of this problem has been achieved.

During calibration measurements with the γ-densitometer a fundamental malfunction in the electronics part (stabilization procedure) has been stated. Possible measures to eliminate this error have been discussed with the manufacturer.

For determining the start time of a blowdown test, a trigger signal can be used which is released through a potentiometer mounted on the shaft of the quick-opening flaps. A signal of 4 V is obtained from a rotation angle of the flap of 3^o out of the closed position.

The availability on an air/water calibration loop for the LOBI measuring instrumentation has proven to be indispensable. Therefore, preparations have begun to modify and extend an existing water loop. Additional components needed (mixing chambers, separator, air compressor) have been ordered. The loop is designed for 12o m^3/h water flow and 12.ooo standard liters/min. air flow.

A two weeks visit with the Semiscale blowdown project of the USNRC at the Idaho National Engineering Laboratory (INEL) at Idaho Falls has procured interesting information on the total manpower allocated to measuring instrumentation and signal analysis tasks of this project, and especially on the distribution of this manpower according to number and competence on the development, test operation and results evaluation tasks.

6.5 Data Acquisition and Evaluation, Process Control

Since the data acquisition system has been installed and the acceptance tests have been successfully concluded, the activities in this field are focused on the development of mini-computer programs for data acquisition, data handling and evaluation, and for process control.

For controlling the electrical heating power input to the heater rod bundle of the reactor model, a special computer/switchboard interface has been developed for enabling the entry of 123 experiment status signals into the computer. This interface is controlled by a computer program which allows the monitoring and controlling of the different phases of the experiment (filling, pressurizing, heating up and steady state operation of the test facility).

The following mini-computer programs have been developed and tested allowing

- the selection of data from arbitrary signal channels with a selectable interval for the mean values and a selectable data compression; an extension allows the automatic error diagnosis and the setting up of an evaluation listing
- the re-organization of the measured data in view of a fast graphical display (minimalization of disk accesses, time reduction from about 260 to about 5 minutes)
- the conversion of the units of data measured into engineering units and their display.

For the graphical data evaluation, a dialog system is under development for an automatic data channel selection and for an automatic setting up of a data book.

Calibration results for the different data channels can be represented graphically or in form of tables by means of a special program, Fig. 7.

All mini-computer programs are compatible with the real time operation system of the computer and allow the parallel running of several programs without interference.

For the data acquisition program during the blowdown experiment, a control program has been developed for automatic testing of the correct operation of the PCM (pulse code modulation) system by listing the error rate of 10.000 data frames befor each data recording.

7. Next Steps

- Completion of operation tests of elements and components of the LOBI facility
- Commissioning of the LOBI facility and preliminary experiments
- Continuation of preparations of the measuring instrumentation system and signal analysis methods
- Testing the whole mini-computer program package of the LOBI-D.A.S. for the data acquisition and evaluation and the process control by simulating a LOBI test.
- Revision of the preliminary test matrix of programme A and subdivision into A 1 and A 2
- Preparation of input data set for, and performing of survey calculations for programme B
- Performing of first pre-prediction calculations for programme A 1
- Continuation of preparations for the LOBI pump tests at WCL-Hamilton/Canada:
  - mounting of the test facility at WCL
  - shipping of the LOBI pump and ancillary system by ASTRÖ
  - preparation of test programme and evaluation methods
- Continuation of preparations for the LOBI break nozzles calibration tests

- Code calculations for determining the dynamic loads and mechanical stresses of the LOBI facility and scaffolding
- Feasibility study of the application of indirectly heated heater rods.

8. Relations with other Projects

RS 0016 B : Vessel blowdown
RS 0036 B : Refilling experiments
RS 50 A : Containment blowdown
RS 64 : Heat flux investigations in multi-rod bundles
RS 81 : Mixing between adjacent flow channels
RS 93 A : Water-vapor flow from tube leaks
RS 111 : Pump behaviour in two-phase flow conditions
RS 0123A/B: HDR-blowdown experiments
RS 163 : Thermohydraulic core behaviour in the early blowdown phase
RS 177 : Fuel rod behaviour under blowdown conditions
RS 179 : Phase separation
RS 182 : Participation in the LOFT-Project of USNRC
RS 184 : Reflooding hydrodynamics
RS 195 : Water-vapor-air flow through containment vent apertures

Two-Phase flow Measurement Techniques
RS 135 : Signal correlation techniques
RS 145 : Joint two-phase flow test loop
RS 146 : Radiotracer techniques
RS 147 : Dragbody
RS 188 : NMR techniques
RS 225 : Density measurement by ultrasonic probes

9. Reference Documents

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- [1] W. Brewka, W. Kolar: MOD5PLOT and MOD2PLOT - Two Versatile Plot Programs for RELAP4/MOD5 resp. RELAP4/MOD 2, EUR 5919.e., 1977

- [2] H. Städtke: Gasdynamic Aspects of Adiabatic Flows with Phase Transitions
Paper presented at EUROMECH Kolloquium 88 on March 3o - April 1, 1977, at Karlsruhe, FRG
- [3] W. Kolar, M. Lolk Larsen, L. Piplies: Calculations for the Standard Problem 3 using RELAP3, RELAP4, RELAP-UK
Paper presented at the Second CSNI workshop on LOCA Standard Problems, Paris, Dec. 6 - 9, 1976
- [4] L. Piplies, W. Kolar: Temperaturaufbau in der Contain-ment-Wand des Ispra Blowdown-Kreislaufs
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- [5] L. Piplies, W. Kolar: Thermohydraulic Specification of the Ispra Blowdown Loop,
Technical Note 162/134/77/BD, March 1977
- [6] W. Riebold, D. Barschdorff et. al.: Summary of German Instrumentation Research
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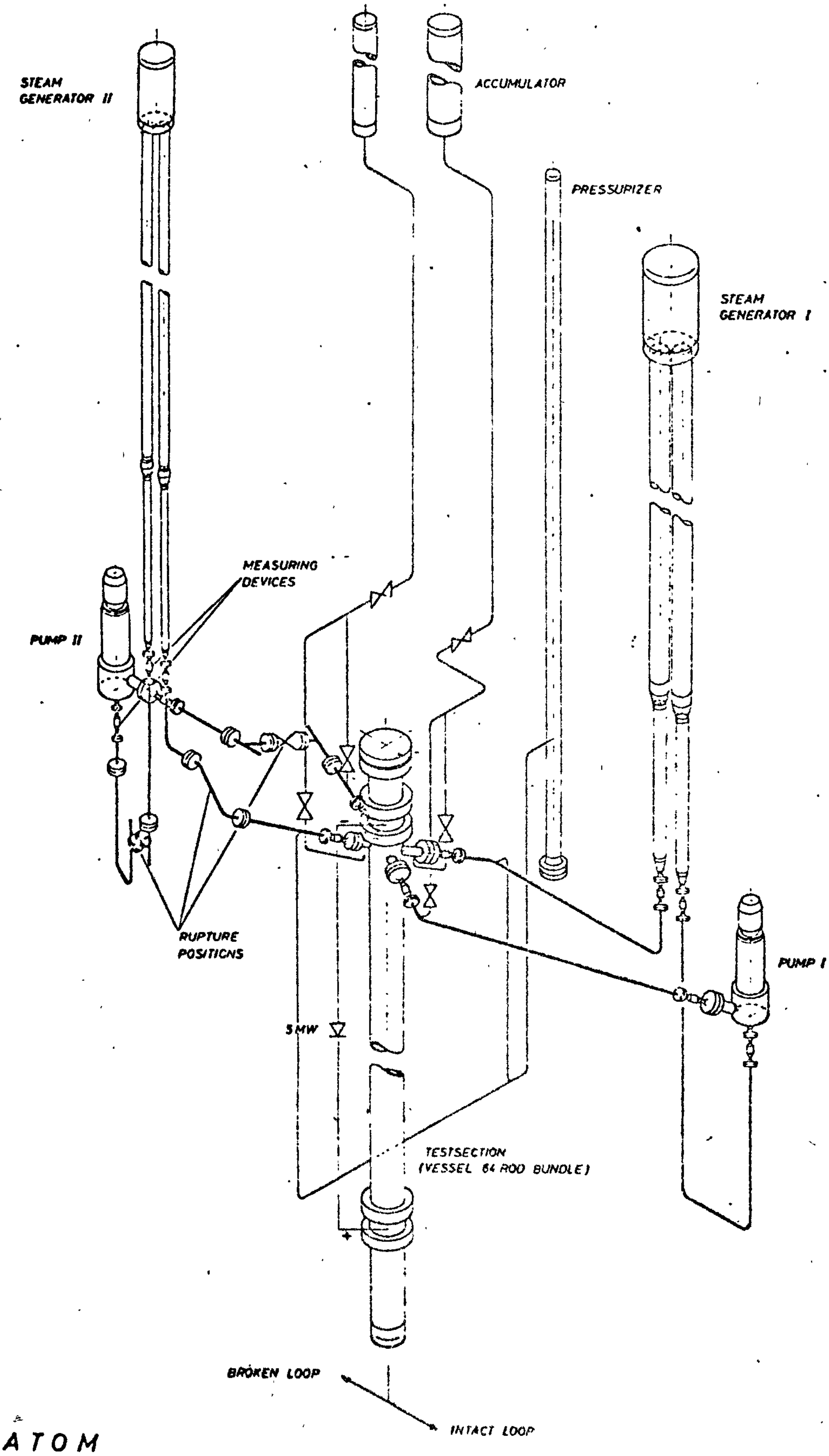
- [7] W. Riebold: Status Report on the LOBI Project (RS-1o9) at Ispra: June 1977,
Technical Note No. I.o6.o1.77.32

1o. Degree of Availability

- Quarterly Reports: from GRS-Köln, Glockengasse 2, 5 Köln 1
- Conference Papers and External (EUR) Reports: from Authors
- Technical Notes: Restricted distribution



EURATOM
C.C.R. ISPRA

FIG. 1: BLOWDOWN LOOP SYSTEM

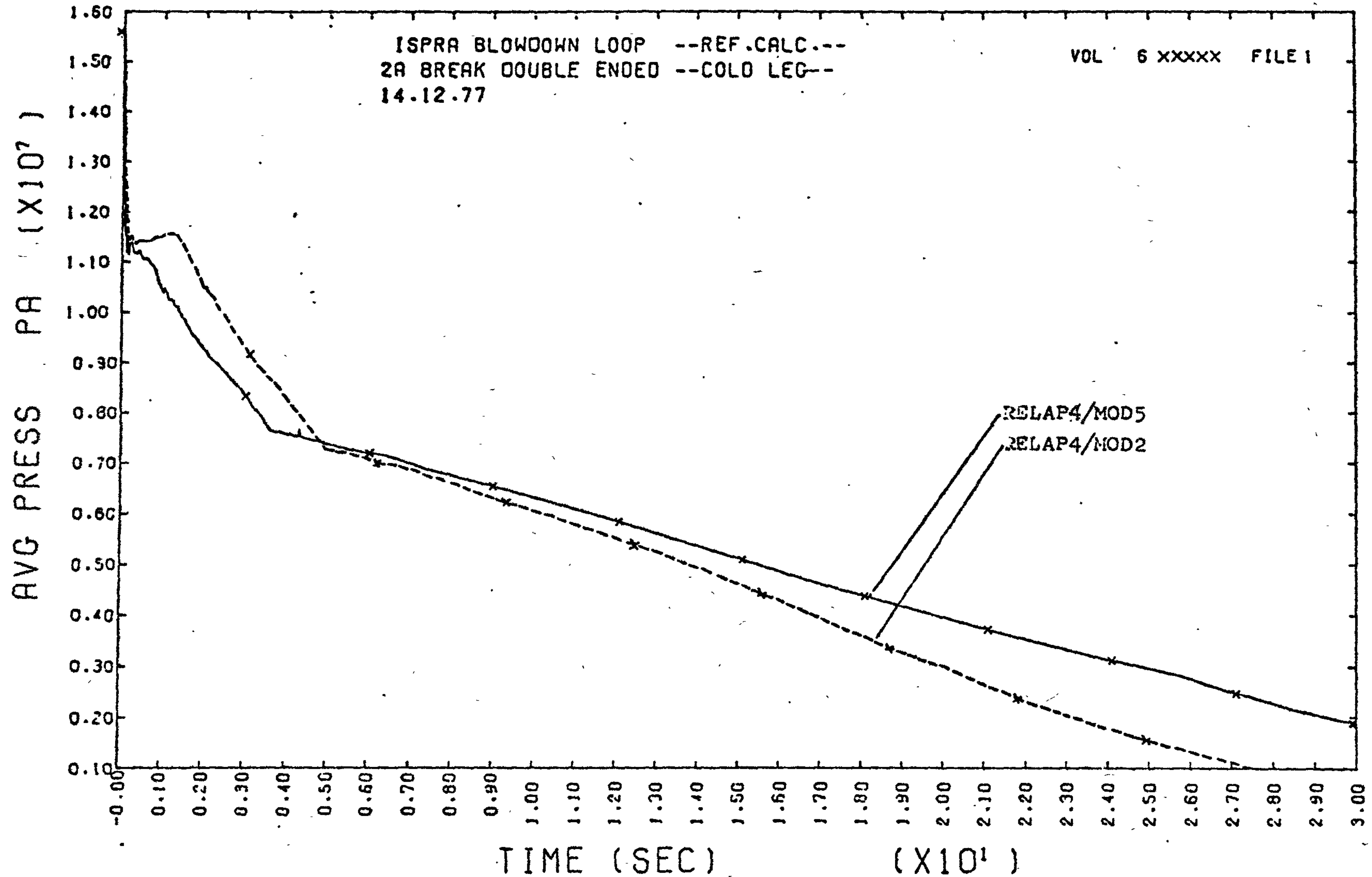


Fig. 2: Pressure middle of heated section

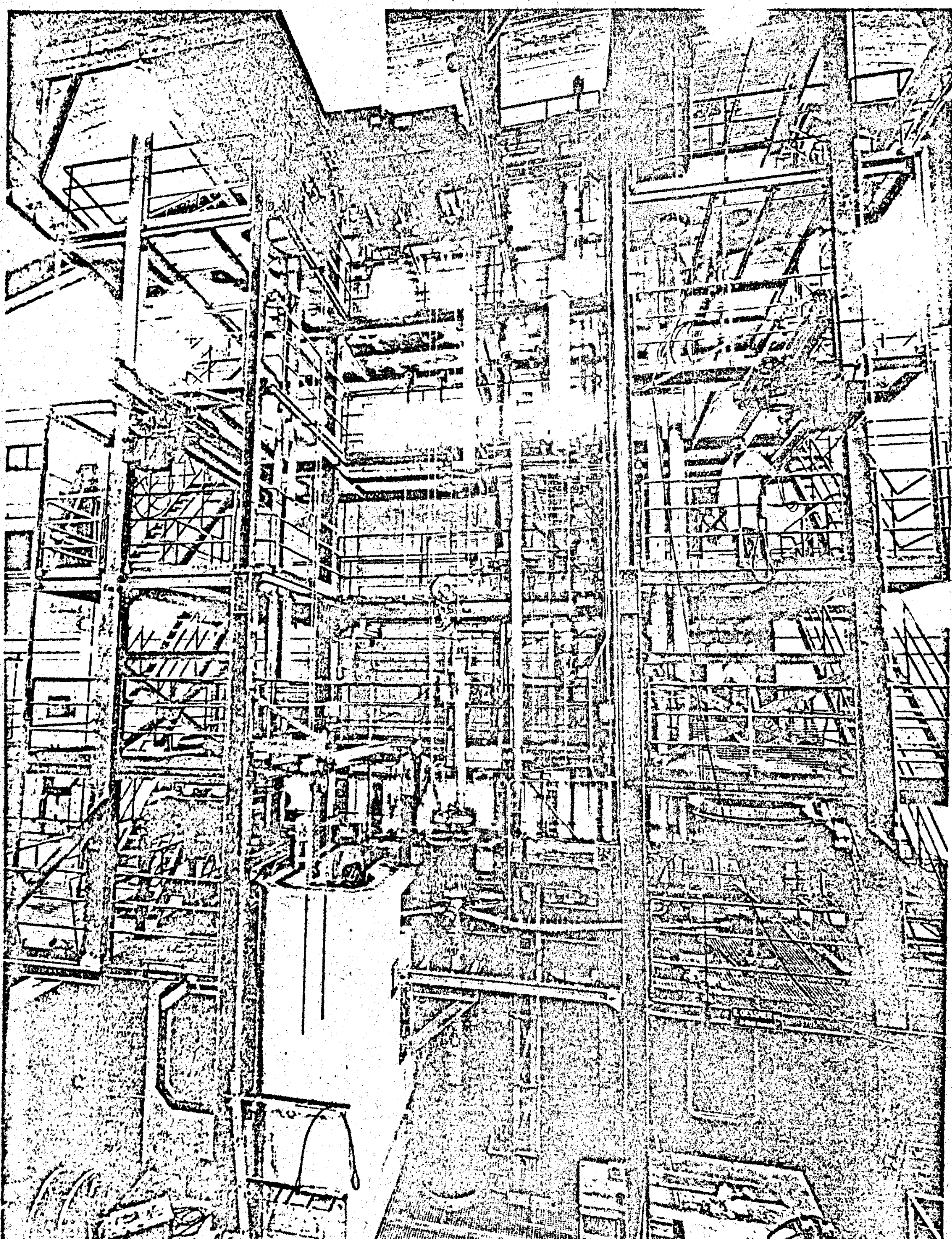


Fig. 3: View on LOBI test facility

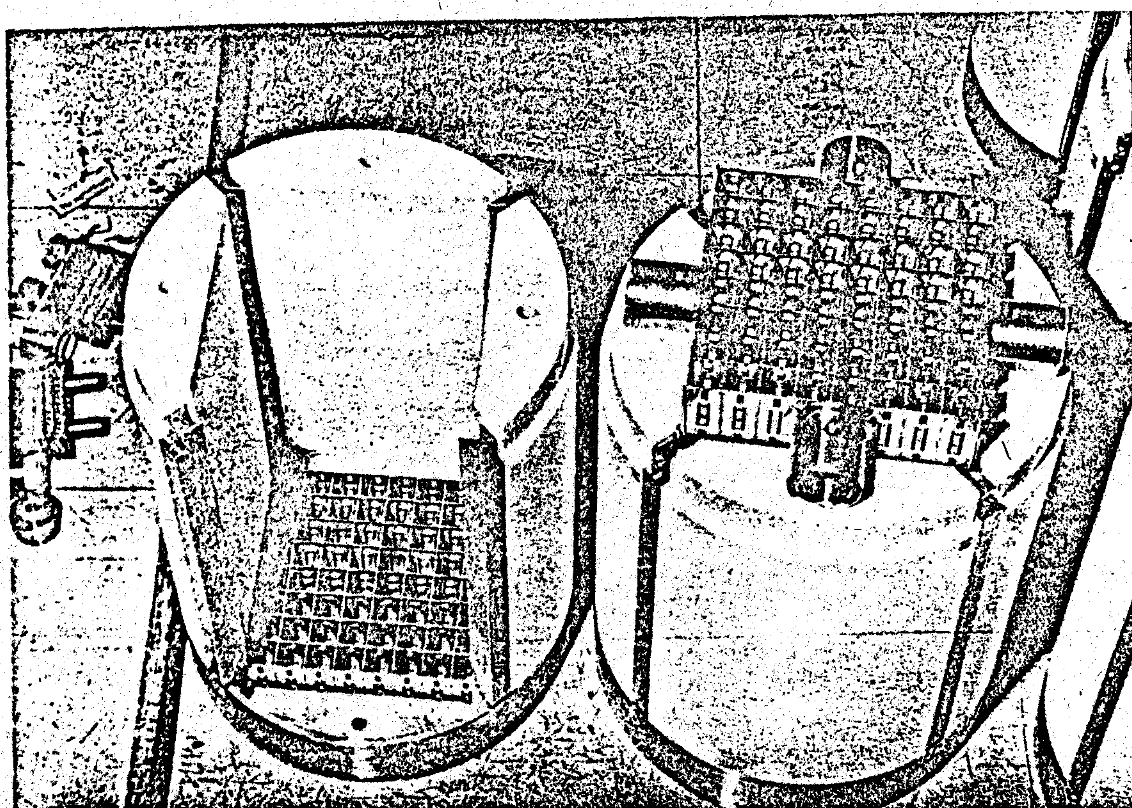


Fig. 4:

a. Spacers with filler and isolator segments for the heater rod bundle

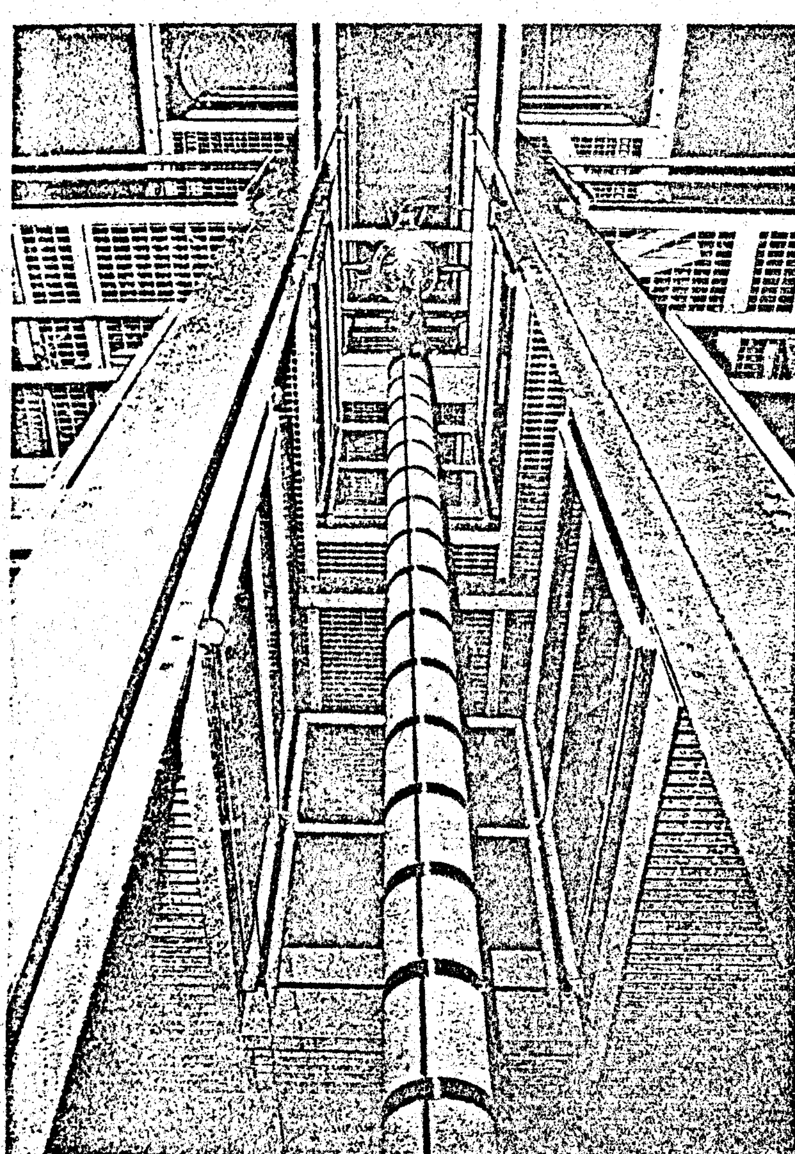


Fig. 4:

b. Heater rod bundle with filler and isolator segments package mounted

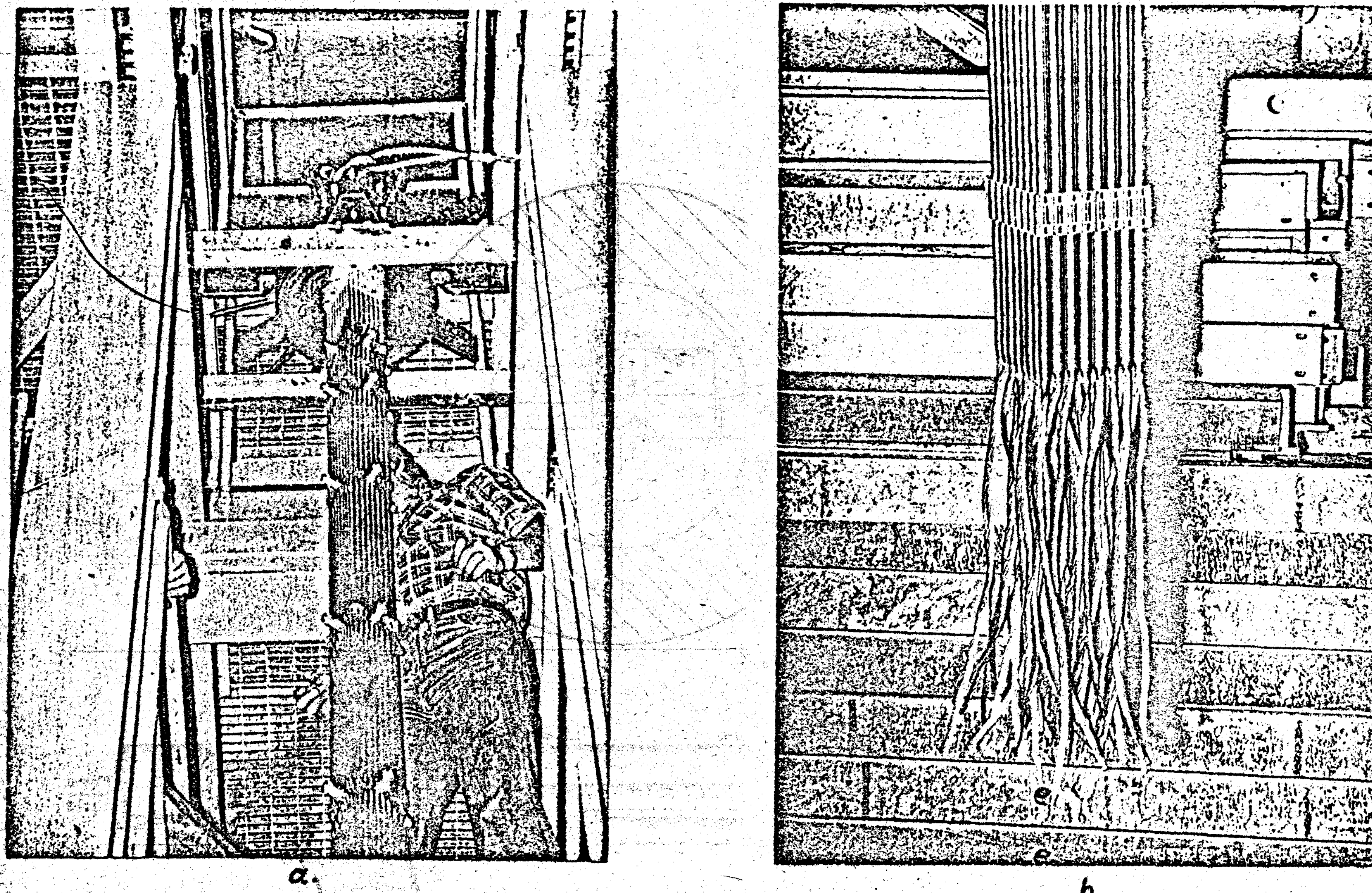


Fig. 5: Heater rod bundle with spacers a. upper part b. lower part

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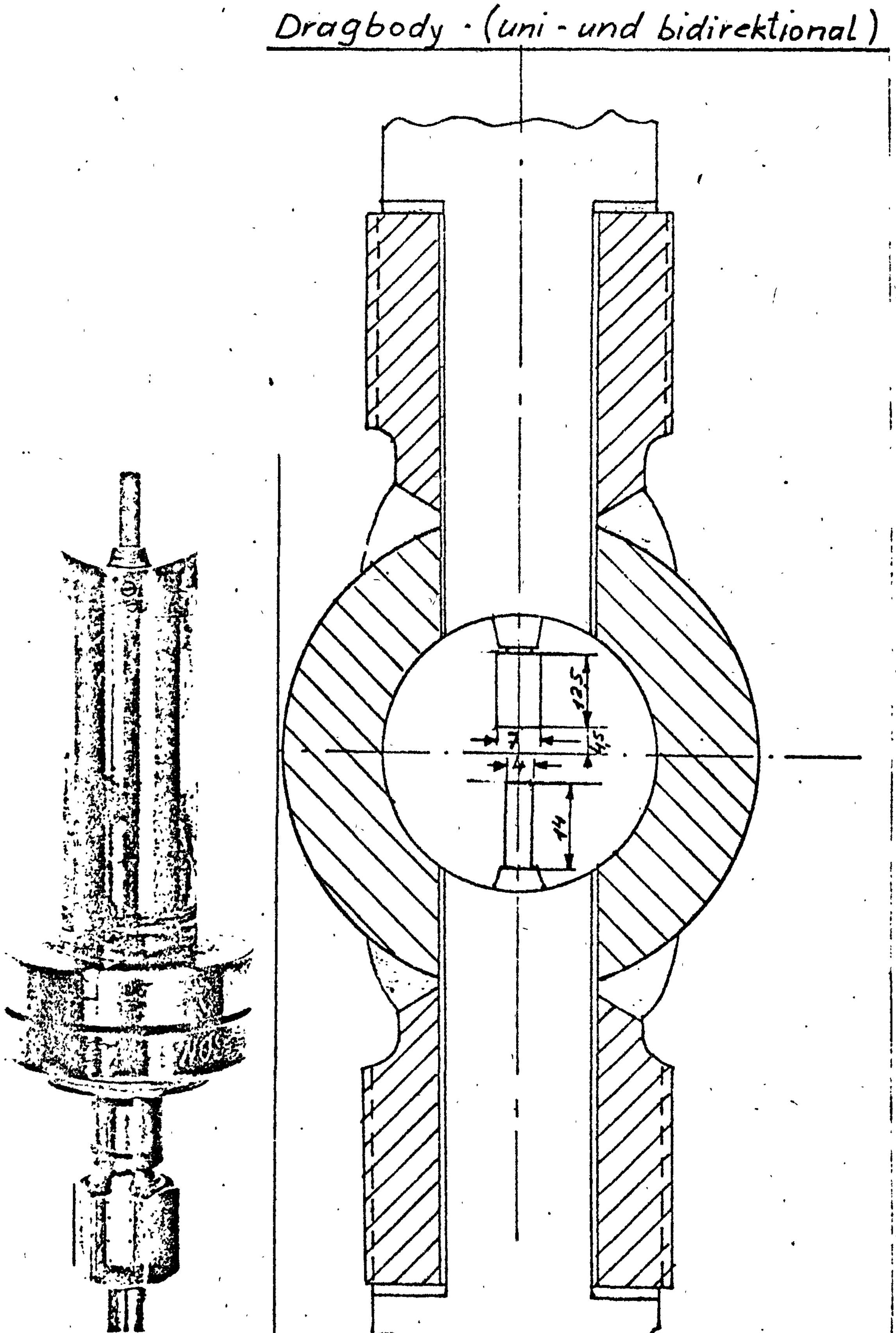
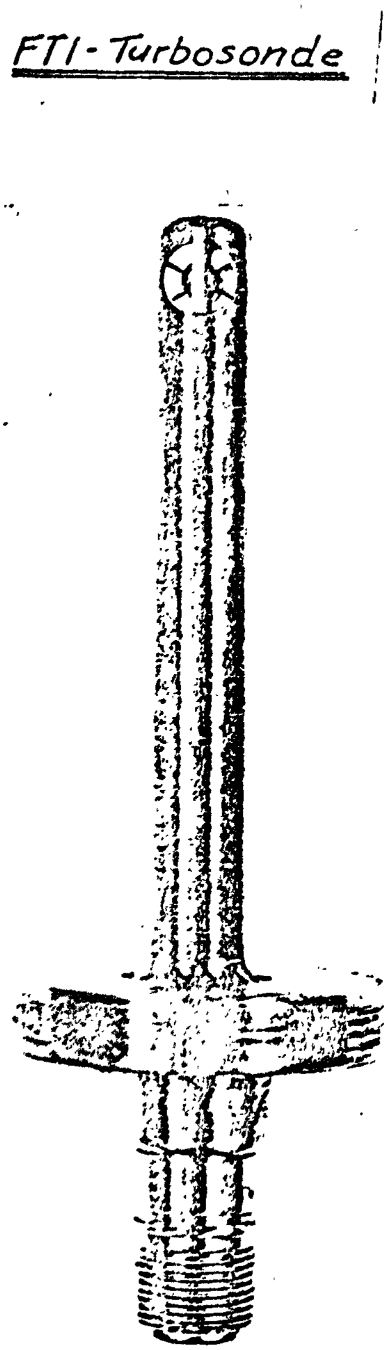


Fig. 6 a: LOBI dragbody



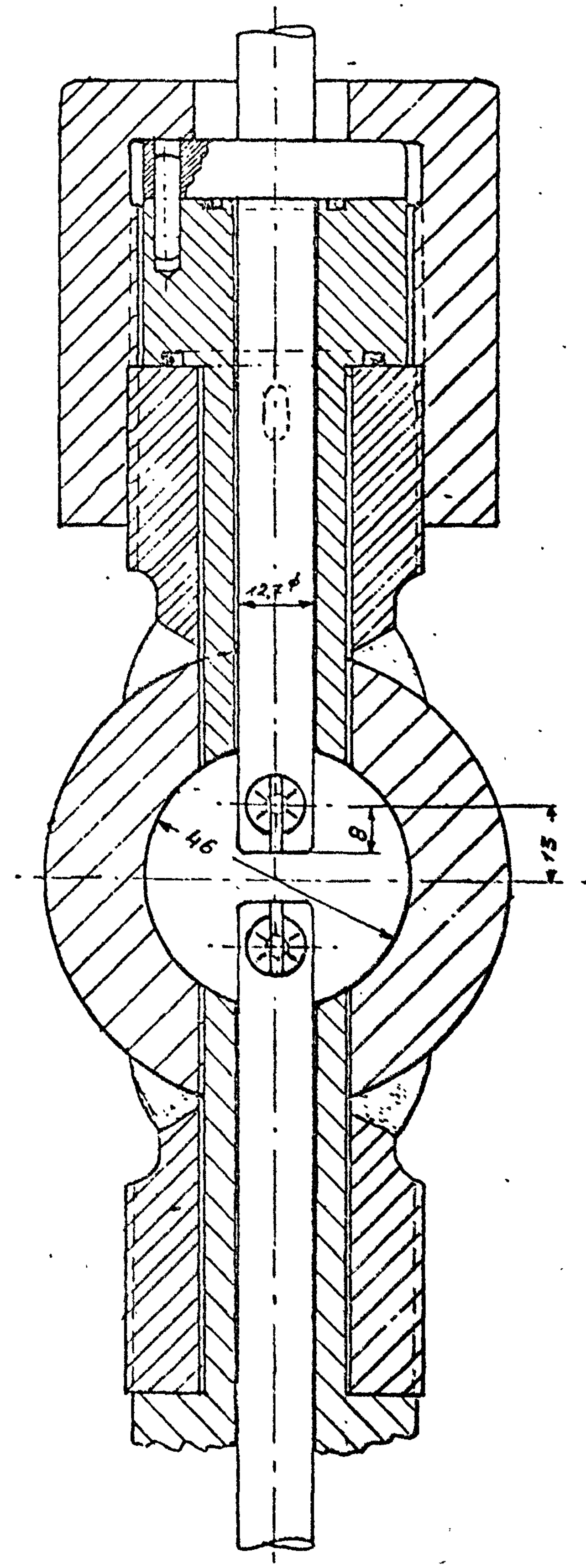


Fig. 6 b: LOBI turbine flow meter

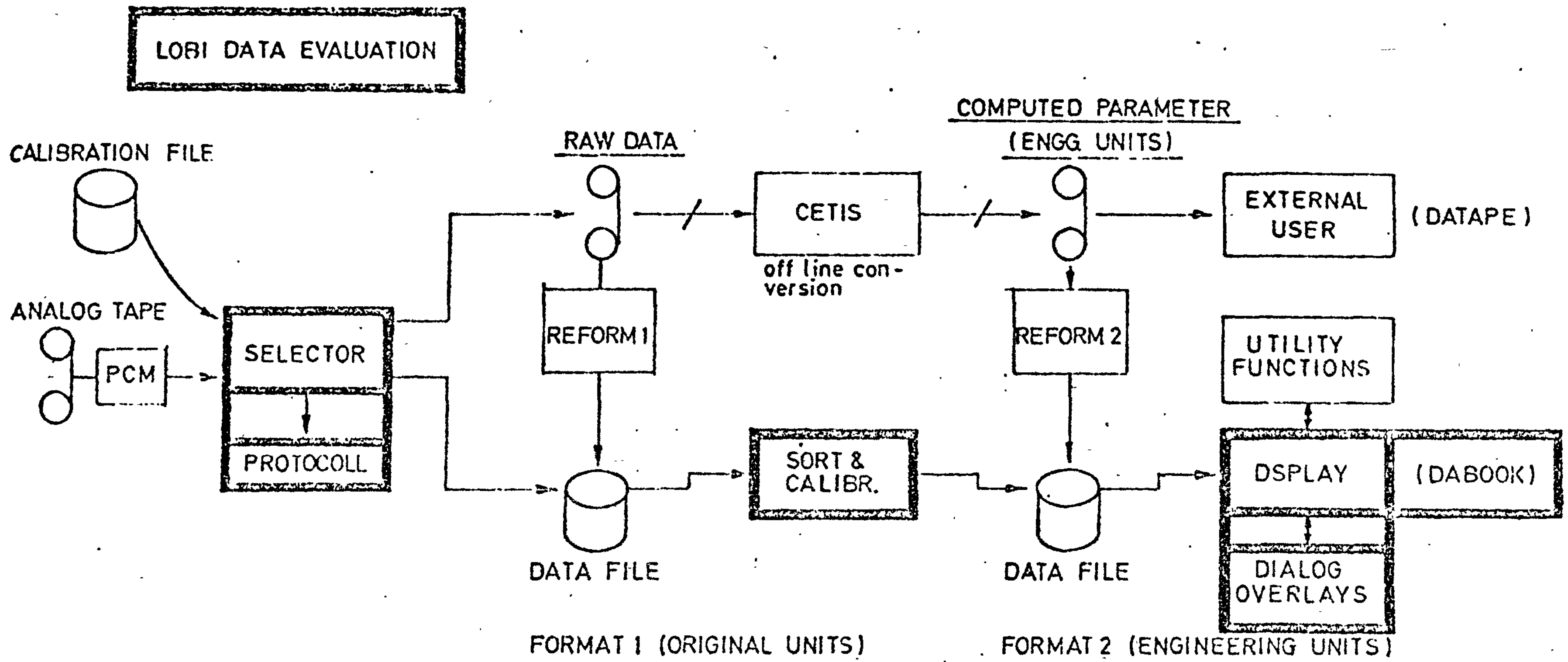


FIG. 7: LOBI DATA EVALUATION

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	1.1.2	RS 111

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Vorhaben/Project Title	
Untersuchungen über das Verhalten von Hauptkühlmittelpumpen bei Kühlmittelverluststörfällen (Phase A) Investigation of the Behaviour of Main Coolant Pumps under MCA Conditions (Phase A)	Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 513, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 9. 74	31. 12. 78	W. Kastner
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 77	4'961.524,-- DM

1. General Aim

During blowdown the flowrate through the core and therefore the temperatures of the fuel rods decisive depends on the behaviour of the main coolant pumps. This behaviour of the pumps under two-phase flow conditions has to be studied, in order to obtain a better experimental and theoretical knowledge and to develop a physical model that can be used in safety analysis.

2. Particular Objectives

Using the experimental results of the pump tests under simulated MCA conditions, a physical model of the pump behaviour will be developed in order to replace the until now applied assumptions in blowdown calculations.

3. Research Program and 4. Experimental Facilities

Two model pumps will be built to scales of 1 : 4 and 1 : 5 of the main coolant pumps of GKN. The single phase characteristic of these pumps will be measured by the manufacturer.

A test loop at C-E will be modified in order to measure two-phase pump characteristics for parameter variations of interest: pressure, flowrate and void. The test matrix for these two model pumps contains about 390 steady-state points in the two-phase region. Also 10 transient tests will be carried out, to investigate whether steady-state results are applicable to transient LOCA calculations.

5. Progress to Date

5.1 C-E-model pump

About 200 steady-state and transient tests of phase I were performed in the two-phase flow region. These tests contain experiments with forward and reverse flow through the pump. Some additional tests with single-phase flow conditions were performed too. The steady-state results were compared with data from the ANC-Semiscale-pump tests. The results of 7 performed blowdown experiments were analysed too.

Phase II was defined by EPRI, the members of the EPRI Pump Two Phase Performance Review Group and C-E, consisting of 54 steady-state and 2 transient experiments. The tests were run successfully at the end of the year.

5.2 KWU-model pump

Based on the results of the EPRI-program the test matrix was changed. For the pump with 200 mm impeller diameter, 10 single-phase flow tests and 70 two-phase flow tests were defined under steady-state conditions. For the second pump with 150 mm impeller diameter also 10 single-phase flow tests and about 200 two-phase flow tests are planned. Additionally 9 transient tests were defined. At the beginning 5 of these tests will be run. About 35 steady-state experiments with special boundary conditions will be necessary, to investigate the possibility of describing the transient pump behaviour by steady-state test results.

6. Results

6.1 C-E model pump

The measured values of head and torque in single-phase flow considerably differed from the data which were supplied by the pump vendor B-J.

For two-phase flow measurements head and torque results showed a decrease with increasing void. For more than 70 % void the head and torque degradation decreased. At 100 % void mearly the same values were reached as for 0 %. This behaviour of head and torque degradation with variing void is comparable to that of the ANC-tests. As in the ANC-Semiscale tests, the range of the investigated hydraulic torques as a function of the void was not very large, a comparison between the EPRI- and ANC-tests was only possible for the head behaviour. The EPRI-tests compared with the data calculated with a LOCA-model based on the ANC-data showed a less pronounced decrease in the head and a more pronounced decrease in the torque. This might be due mainly the different specific speed of the B-J and ANC-pumps.

The evaluation of 7 blowdown-experiments showed, that it will be possible to describe the blowdown-phase using steady-state results. Additional investigation has to be done.

6.2 KWU-model pump

ASTRÖ has completed the detailed measurements of the pump cast structure, manufactoring of impellers, casings, bearings and shafts is going on. AEG delivered all components of the pump drive. The instrumentation and loop equipment is defined, devices and parts for the loop and pump support have been ordered by C-E. The test matrix for investigating the two KWU-model pumps has been completed.
Some auxiliary systems of the pumps (sealing water- and lubrication systems) have been completed.

7. Next Steps

The EPRI test results will be evaluated. The final report will be prepared. For the KWU test program C-E will continue on structuring, planning and ordering of loop-components.

ASTRÖ will complete the pump casings and the electric equipment of auxiliaries. As some auxiliary systems (as sealing water- and lubrication systems) will be needed for experiments with the ISPRA-pump at Westinghouse/Canada, the installation of the KWU-pump will be started in the 4th quarter of 1978.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1977 - 31.12.1977	1.1.2	RS 176

119

Vorhaben/Project Title	Land/Country
Stationäre DNB-Messungen in Frigen mit komplexer Abstandshaltergeometrie Steady state DNB Measurements in Freon with complex Spacer Geometry	FRG
	Fördernde Institution/Sponsor: BMFT
	Auftragnehmer/Contractor: GKSS Geesthacht Institut für Anlagentechnik

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.9.1975	31.12.1978	Katsaounis
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31.12.1977	483.000,-- DM

1. General aim

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

2. Particular objectives

The program is diveded 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. Research program

In order to check the condition of the 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. Experimental facilities

See RS 176 annual report A 76

5. Progress to Date

The fabrication of the test section is finished and the test section ready to be installed.

All components for the modification of the electrical power supply are delivered.

6. Results

As the test programm has not started yet, there are no results.

See RS 176 annual report A 76.

7. Next steps

As the research programm RS 64 had to be interrupted in October 77 (caused by an emergency in power supply system) the freon tests are planed to be continued with the programm RS 176 in the following steps

modification of power supply system

installation of test section

test programm RS 176.

8. Relation with other Projects

See RS 176 annual report A 76.

9. References

[1] GRS-Research reports: GRS-F-40 etc.

10. Degree of Availablility of the Reports

All reports are available with the allowance of GRS, department Forschungsbetreuung.

414

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1977	1.1.2	RS 64
Vorhaben/Project Title Untersuchung der stationären und instationären kritischen Heizflächenbelastung an Vielstabbündeln von Druck- und Siedewasserreaktoren mit Frigen als Modellflüssigkeit Investigation of the Steady State and Transient Critical Heat Flux of Multi-Rod Bundles for PWR and BWR with Freon as Model Fluid		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor GKSS Geesthacht Institut für Anlagentechnik
Arbeitsbeginn/Initiated 1.11.72	Arbeitsende/Completed 30.6.78	Leiter des Vorhabens/Project Leader
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31.12.77	Bewilligte Mittel/Funds 1.723.500,- DM

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:

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

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

Part III - Burnout tests (time delay and heat transfer under 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. Research program

For the experiments of part I and II are provided two test sections with 7 x 7 rods for the core configuration incl. the spacers of PWR and BWR respectively. The experiments of part III will be carried out using test sections with 5 x 5 rods. The axial and radial heat flux distribution of each test section is non uniform.

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 included:

Mixing measurements, 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. Experimental Facilities, Computer Codes

See RS 64 annual report A 76, A 75

5. Progress to Date

As to part I the heater rods of the PWR-test section had been insulated and reinstalled. Part I of the test program had been started again. After finishing the mixing experiments the test program was interrupted by a failure in the power supply system (part of the test section was destroyed). Repairing of the power supplysystem is finished. The tests are planed to be continued with the research program RS 176 in 1978, as repairing of the damaged test bundle needs more time. The BWR-test section is fabricated except insulation of heater rods.

As to part III the revision of the test facilities was especially carried out with regard to a detailed calculation of costs. Particularly the concept of regulating, control system and instrumentation had been worked through.
After calculation of total costs and setting up a general network plan the application to the BMFT followed. In a series of intern GKSS-reports the obtained state of the project was layed down in relation to all technical details. First test runs with a computing program modified for freon 12 were finished successfully.

6. Results

Part I : The results of the mixing experiments at the PWR-test section will be reported soon. The steady state CHF-data will be collected and reported at the end of program part I.

Part III: Total costs and network plan for this part of project are on hand. Preparation for requesting a general tender in the fields of components, control system and instrumentation can be started. Calculation of the loop-characteristic using the computing program COVACU modified for freon 12 are prepared.

7. Next steps

Part I and II: The tests will be finshed under the aspect of manpower, actuality of results and with special respect to the accident in power supply system. In 1978 the tests will start with the program RS 176 to get time for repairing of the PWR test section.

Part III : In reference to the GRS-resulting report of 20.10.77 it was decided against this part of the project. All activities in the field of blow down tests will be dropped.

8. Relations with other projects

RS 164

RS 176

9. References

/1_7 H. Fulfs et al. (German)
Techn.Fachbericht, Notkühlprogramm-Forschungsvorhaben RS 64, Beschreibung des Frigenversuchsstandes und der Meßstrecken DWR-49-Stabbündel und SWR-49-Stabbündel,
GKSS 77/I/32

/2_7 H. Fulfs, H. Rosomm (German)
Techn. Fachbericht, Notkühlprogramm-Forschungsvorhaben RS 64, Innenverkupferung der Heizstäbe für das DWR-49-Stabbündel und das SWR-49-Stabbündel,
GKSS 77/I/34

/3_7 H. Fulfs (German)
RSTAB - Ein Rechenprogramm zur Ermittlung des elektrischen Widerstandes eines innenverkupferten Heizstabes aus dem vorgegebenen Leistungsprofil und der mittleren Stabtemperatur,
GKSS 77/I/31

/4_7 C.v.Minden (German)
Thermohydraulische Auslegungsparameter der Frigenversuchsanlage zur Durchführung von Blowdown-Versuchen für Vorhaben RS 64, Phase III,
GKSS 77/I/21

[5] W.Schulze-Erfurt (German)
Instrumentierungskonzept für die Frigenversuche bei Drucktransienten (Phase III)
GKSS 77/I/30

[6] H.O.Boie, C.v.Minden (German)
Thermodynamische Auslegung der Erweiterung des Frigenversuchsstandes für transiente Frigenversuche (RS 64, Phase III),
GKSS 77/I/23

[7] GRS-Research reports : GRS-F-40

[8] GKSS 72 05 AT C 13 H.Fulfs, M. Stein (German)
Ergebnisse der Durchmischungsversuche DWR/Phase I/RS 64 (1.Abschnitt), revidierte Fassung.

[9] H.J.Gerbracht (German)
Steuerungs- und Regelkonzept für die erweiterte Frigenversuchsanlage RS 64-Phase III

[10] KWU-Arbeitsbericht R 123 - 261/77 (German)
Berner: Festlegung der Burnoutversuche in Frigen 12 bei Mengenstrom- und Leistungstransienten (Versuchsphase II); DWR-Bündel; überarbeitete Version, Erlangen, 20.10.77

10. Degree of availability

All reports are available with the allowance of GRS department "Forschungsbetreuung".

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01. - 31.12.1977	1.1.2	RS 163

Vorhaben/Project Title	
Theoretische und experimentelle Untersuchungen zum thermohydraulischen Verhalten des Reaktor Cores in der ersten Blowdown-Phase Theoretical and Experimental Investigations on Thermo- and Fluiddynamic Behaviour of the Reactor Core in the First Blowdown Period	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Institut für Verfahrenstechnik der T.U. Hannover Callinstr. 36 3000 Hannover 1

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
March 1975	December 1978	Prof.Dr.-Ing. F. Mayinger
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31.12.1977	DM 1.057.624,-

1. General aim

The general aim of these investigations is to predict the thermo- and hydrodynamic behaviour of the two-phase flow in the reactor core in the first blowdown period during a LOCA.

2. Particular objectives

The main objectives are the experimental investigations of the influence of loop components on the thermohydraulic conditions and the cooling performance in the reactor-core during LOCA-conditions.
Furthermore the influence of mixing-processes between adjacent-subchannels and the significance of the entrainment behaviours with regard to the dryout delay time has to be investigated.

3. Research program

The experiments are performed with the modelfluid R 12 instead of water. In the following tabular the main objects of the research program are listed in detail:

3.1 Entrainment-investigations

Experimental studies of the entrainment mass flow rate in the annular flow regime during LOCA-conditions and the relation to dryout-delay time and post dryout heat transfer.
Evaluation of a theoretical model to describe the transient entrainment mass flow.

3.2 Mixing-investigations

Experimental and theoretical studies of the mixing behaviours between adjacent subchannels in the rod bundle of a nuclear reactor core.

3.3 Loop behaviour

Experimental studies concerning the influence of loop components, for example pump resistance and volume of the steam generator on the thermohydraulic behaviour of the coolant flow in the reactor core.

4. Experimental facilities, Computer codes

To 3.1 The entrainment-measurements were performed at an inside cooled tube with an inner diameter of 14 mm and a heated length of 5 m.

To 3.2 For the mixing measurements a two-channel test-section was designed and manufactured. It's cross-section area is shown in fig. 1

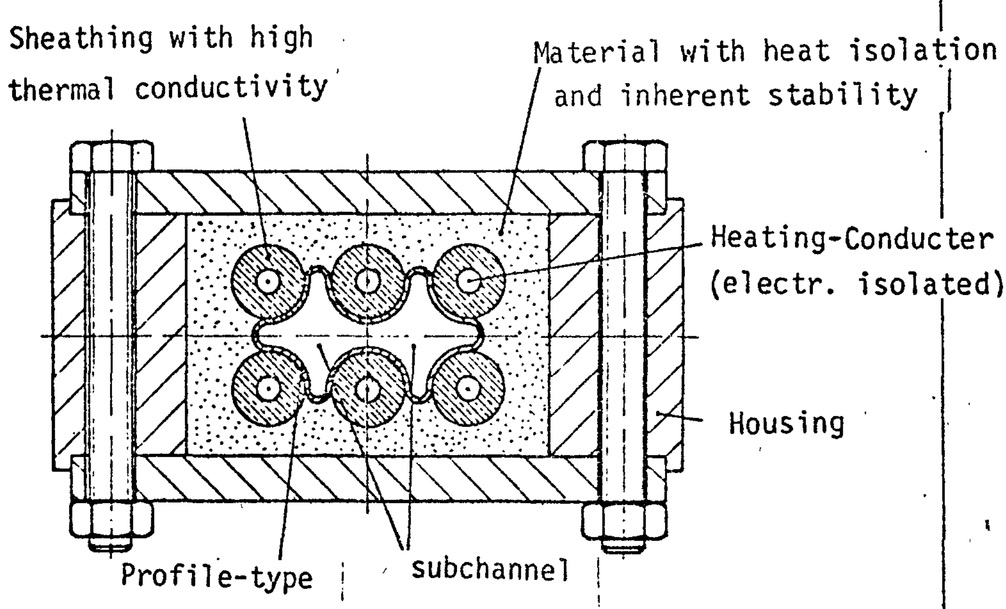


fig. 1: Cross section area of the two-channel mixing test-section

The essential advantages of this test-section design is that the boiling phenomena occuring in the gap-region between two heater rods are simulated exactly as in a nuclear reactor. Furthermore the heat flux densities of the single rods may be varied seperately.

To 3.3 During the period reported a test rig modeling the primary circuit of a PWR was constructed. The design data of this experimental apparatus base on the following simplifications:

- The four loop standard plant is reduced to a two-loop model
- The tests are performed with the model fluid R 12 basing on the same critical pressure ratio.
- Nuclear heating is represented by a direct electrically heated 4-rod-bundle.

- In a first step the ratio between stored volumes in the test rig and in the original loop is stated as

$$\frac{\text{number of heated rods in the model loop}}{\text{number of heated rods in the original loop.}}$$

  Detailed investigations on the base of specified model laws are planed for the next time.
- The flow resistance in the intact loop is simulated by valves.
- The pump behaviour was simulated by a control valve to adapt different pump resistances.

5. Progress up to date

Within this report period the following investigations were conducted.

To 3.1 Entrainment-investigations

a) Entrainment-measurements in transient two-phase flow in an inside cooled tube. The entrainment mass-flow was measured in dependance of the following parameters.
   1) total mass flux at steady state initial conditions
   2) void fraction at initial conditions
   3) time after rupture
   4) break area

b) Another test series was performed to check the application of STORZ-lenses for entrainment measurements in rod bundles.

To 3.2 Mixing tests

a) comparison of different theoretical models by the use of the modified COBRA-III C code

b) cross flow measurements at steady state two components two-phase flows - air-water mixtures - at atmospheric pressure but with artificially reduced bubble size.

c) development of a measuring technique to determine two phase mass flow at steady state and transient conditions.

To 3.3 Loop behaviour

a) The activities to record the experimental data on a on-line data-log system are finished.

b) The influence of heat flux on the dryout delay time in a 4-rod bundle is being measured. In these tests the pump resistance was neglected.

c) Tests variing the pump resistance and informing about the influence of the hydrodynamic behaviour of the coolant flow in the core are in action.

6. Results

To 3.1 The measurements in a transient annular flow under blowdown conditions show that the entrainment mass flow is strongly influenced by two parameters:

a) break area: governing the acceleration of the vapour phase

b) the void fraction at initial steady state conditions: influencing the total amount of liquid in the channel.

The total mass flow at steady state conditions before blowdown starts has no effect on the transient entrainment behaviour. With higher acceleration the liquid fraction entrained increases, leading to a sharp reduce of the film thickness as demonstrated in fig. 2

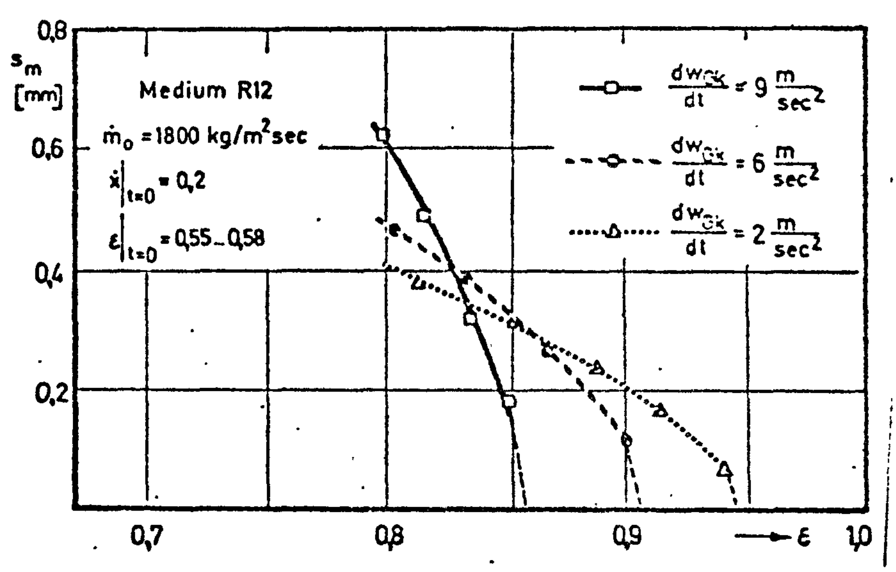


fig. 2: film thickness under transient blowdown conditions

This is the reason why dryout occurs, although there is still quite a lot of liquid in the channel. In fig. 3 the transient entrainment mass flow rate is plotted against the local void fraction measured at the end of the heated test section.

With increasing acceleration of the vapour phase in the gas core, the dryout point is moved towards lower qualities as shown in fig. 3.

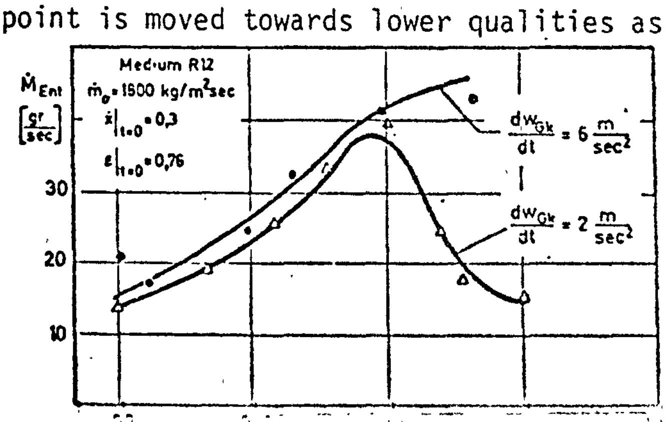


fig. 3: transient entrainment mass flow rate

In addition to the experimental work a calculation model for transient entrainment mass flow was evaluated using the conservation laws for mass, energy and momentum in the vapour core. In addition to the three differential equations a constitutive law describing the interfacial shear stress between the phases is adopted to solve the above-mentioned system of equations. This system of equations can be used to calculate the entrainment mass flow in a transient annular flow, if the time depending gradients of: pressure drop between in- and outlet, total mass flux, system pressure and vapour velocity are known from measurements or from a blowdown code like RELAP. A comparison between calculated and measured entrainment mass-flow rates is shown in fig. 4. The maximum deviation between test-results and theoretical prediction is in the range of about ± 20 %.

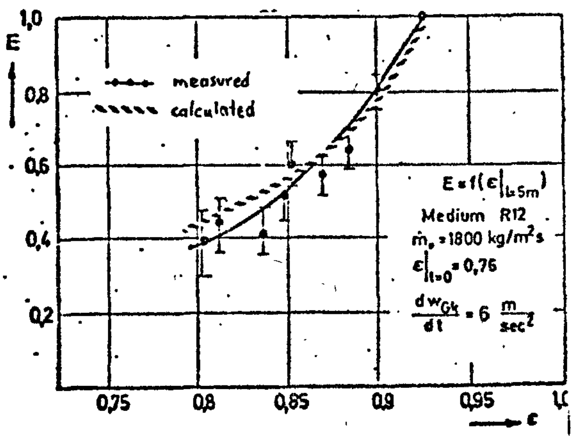


fig. 4:
comparison between calculated and measured entrainment data.

In order to proof the entrainment measurements gained at an inside cooled tube, first experiments with s.c. STORZ-lenses in a rod bundle have been carried out. To check the measuring technique with these lenses, an air-water system at atmospheric pressure has been used in that first step. Fig. 5 shows a photo of a spray flow in the subchannel of a rod bundle which is typical for the end of the blowdown.

To 3.2 Four models given in the literature were used to compare measurements and theoretical prediction:

to model of Rowe
Lahey
van der Ros

and a combined model of Rowe and van der Ros. Figures 6 and 7 show a comparison of results gained with this model with measurements of Lahey for a sym-

metrically heated 9-rod bundle. There are heat differences especially for the corner channel. The mass flow is under-predicted by the theory and the void fraction is over-estimated.
The best agreement between calculations and experimental results could be obtained for symmetrical heated rod bundles. For the assymmetrical case - see fig. 8 and 9 - the Lahey model shows great differences between experimental and calculated results. The Lahey model predicts too small void fraction in the corner channel, although at the upper left corner rod the highest heat flux occurs. A critical inspection of the mixing models shows that all models used are not able to predict the measured behaviour as well for symmetrically as for assymetrically heated rod bundles.

fig. 5: spray flow in the subchannel of a 4-rod bundle

To investigate the phenomena of the mixing mechanisms experimental investigations were carried out with air-water flows at atmospheric pressure. The diversion cross flow was measured at constant total mass flow of each phase. The experimental set up for simultaneous high speed cinematography and pressure recording is shown in fig. 11. Figure 11 shows the mixing mechanisms for the case of one side air injection. By measuring the local values of phase-velocity, void-fraction and static-pressure along the channel the local cross flow could be determined and the following conclusions could be drawn:

1. At the smallest cross-section between the rods plane vortices are formed which have an important influence on the mixing behaviour and especially on the gaseous diversion cross flow.
2. The rotation of the vortices is a function of the velocities difference between the adjacent subchannels and influences the magnitude of the pres-

sure difference.

3. The axial frequency of the vortices is in agreement with the mean frequency of the pressure oscillations.
4. The pressure behaviour shows, that the static pressure in the left channel slips forward with respect to the right one due to the higher mean density in the channel with lower void fraction.

In the experimental investigations the turbulent plane shear flow could be found as a dominant mixing effect especially for the gas phase. The reason of the liquid diversion cross flow are the axial and radial pressure fluctuations. The investigations described above led to physical explanation of the two-phase mixing behaviour.

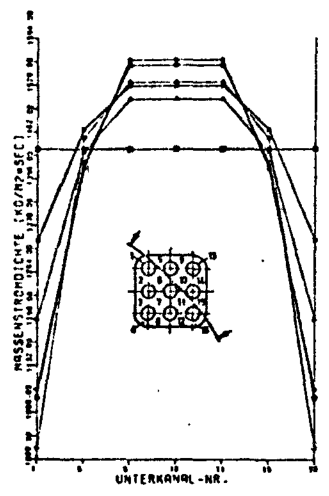


Fig. 6:

✱ MESSUNG LAHEY 2E3
QUERVERMISCHUNGSMODELL NACH:
○ ROWE BETA=0.01
△ LAHEY BETA=0.001 (VOID DRIFT)
◇ ROS CD=0.0004 (M3/S2) (DIFFUSION)
● ROS CD=0.0004 (M3/S2) BETA=0.02 (DIFF.-TURB. MIXING)
□ KEINE QUERVERMISCHUNG

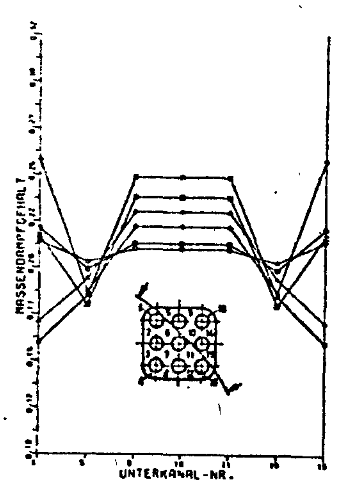


Fig. 7:

✱ MESSUNG LAHEY 2E3
QUERVERMISCHUNGSMODELL NACH:
○ ROWE BETA=0.01
△ LAHEY BETA=0.001 (VOID DRIFT)
◇ ROS CD=0.0004 (M3/S2) (DIFFUSION)
● ROS CD=0.0004 (M3/S2) BETA=0.02 (DIFF.-TURB. MIXING)
□ KEINE QUERVERMISCHUNG

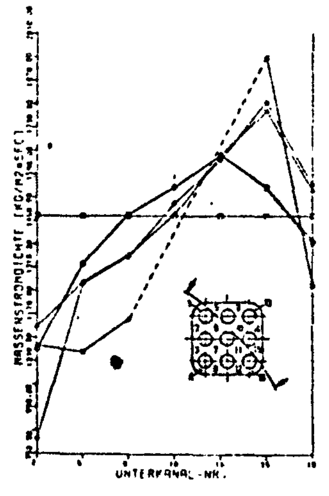


Fig. 8:

✱ MESSUNG LAHEY 3E1
QUERVERMISCHUNGSMODELL NACH:
○ ROWE BETA=0.0375
△ LAHEY BETA=0.0075 (VOID DRIFT)
◇ ROS CD=0.0004 (M3/S2) (DIFFUSION)
● ROS CD=0.0004 (M3/S2); BETA=0.0375 (DIFF.-TURB. MIXING)
□ KEINE QUERVERMISCHUNG

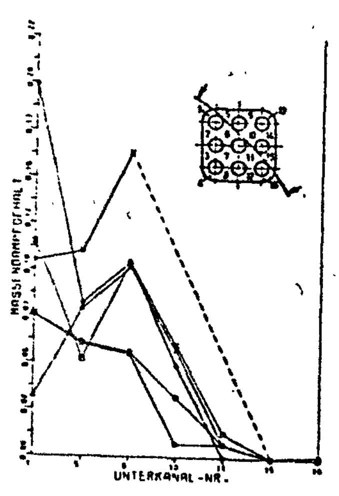


Fig. 9:

✱ MESSUNG LAHEY 3E1
QUERVERMISCHUNGSMODELL NACH:
○ ROWE BETA=0.0375
△ LAHEY BETA=0.0075 (VOID DRIFT)
◇ ROS CD=0.0004 (M3/S2) (DIFFUSION)
● ROS CD=0.0004 (M3/S2); BETA=0.0375 (DIFF.-TURB. MIXING)
□ KEINE QUERVERMISCHUNG

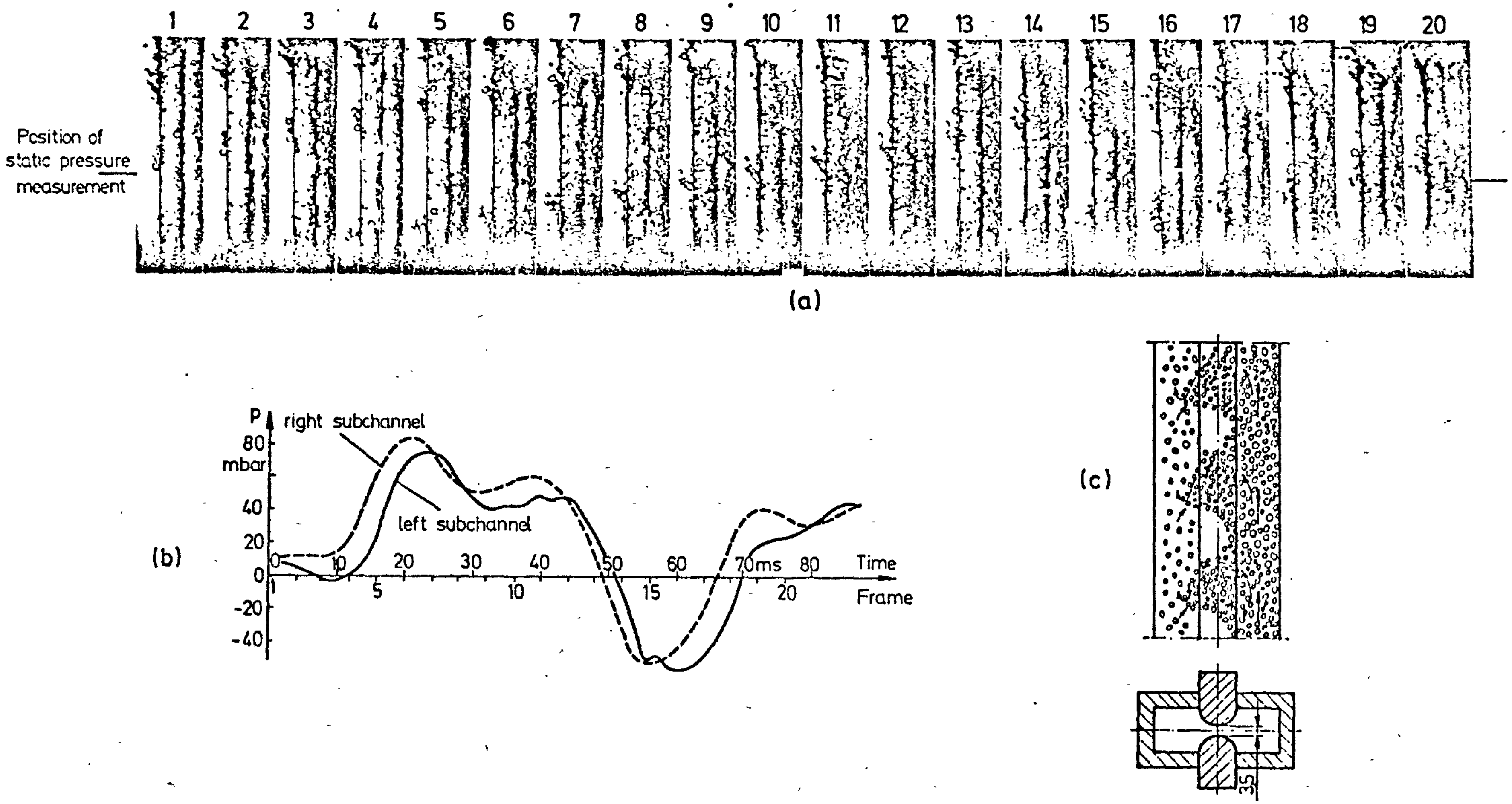


Fig. 10: High frequent mixing movements with simultaneous static pressure measurement

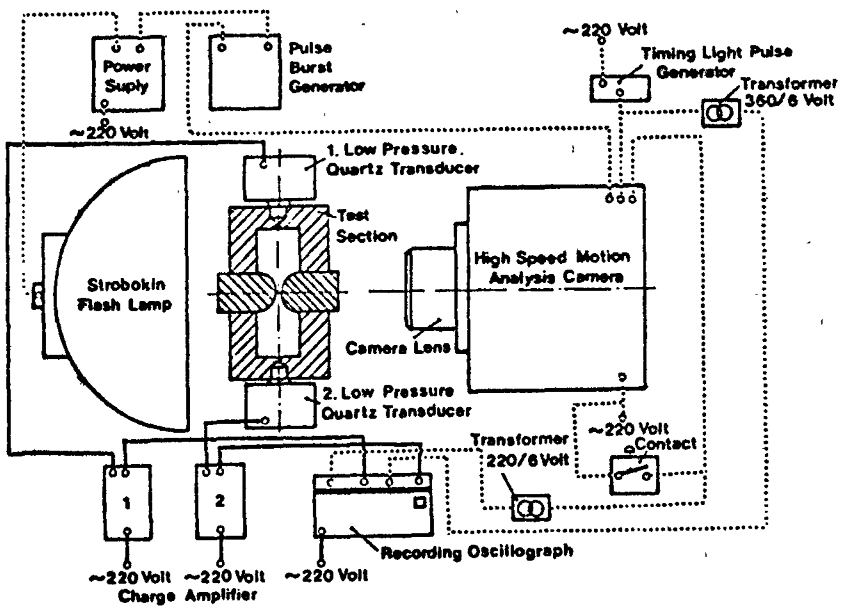


Fig. 11:

experimental equipment

To 3.3 To check the data-log system for the application in transient blowdown tests a test series was performed dealing with the influence of heat-flux on the dryout-delay time in the rod bundle. In this special test series the influence of the pump resistance was neglected. The other essential test data are:

Break area: 310 mm² corresponding to a 0.5 F break

Mass flux : 1800 kg/m² sec

Heat flux : $90/_{9Do:\ STAT} = 0{,}16 \ldots 0.6$

In fig. 12 to 15 typical test results as gained from these blowdown tests are shown. Very interesting and essential for the estimation of the coolant behaviour during LOCA-conditions is the volumetric flow. Fig. 12 shows that flashing in the lower plenum is indicated by an increase of the volumetric flow.

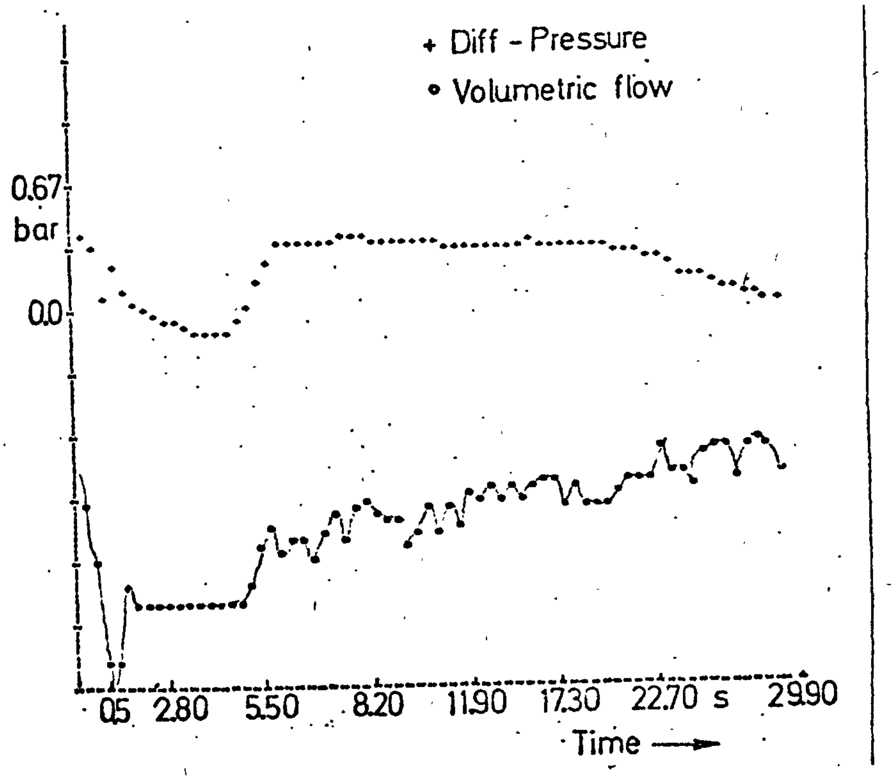


Fig. 12: Rod bundle differential pressure and intact loop cold leg volumetric flow, = 3,5 W/cm², 0,5 F break

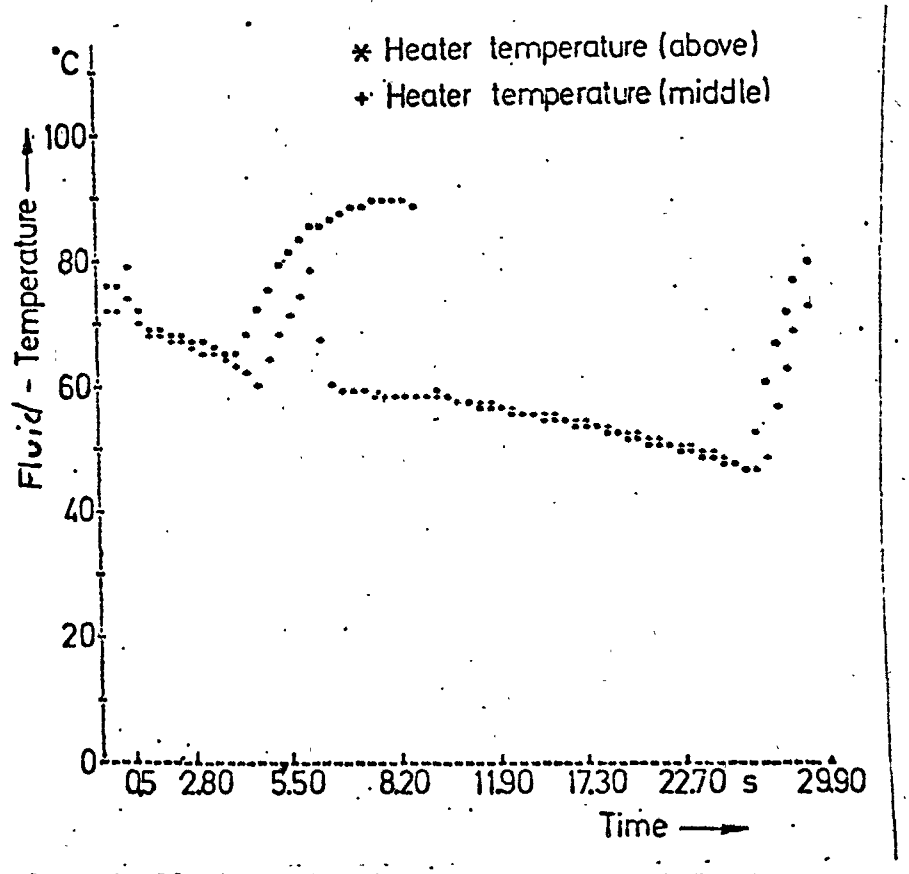


Fig. 13: Electrically heated rods temperature, q=3,5 W/cm², 0,5 F break

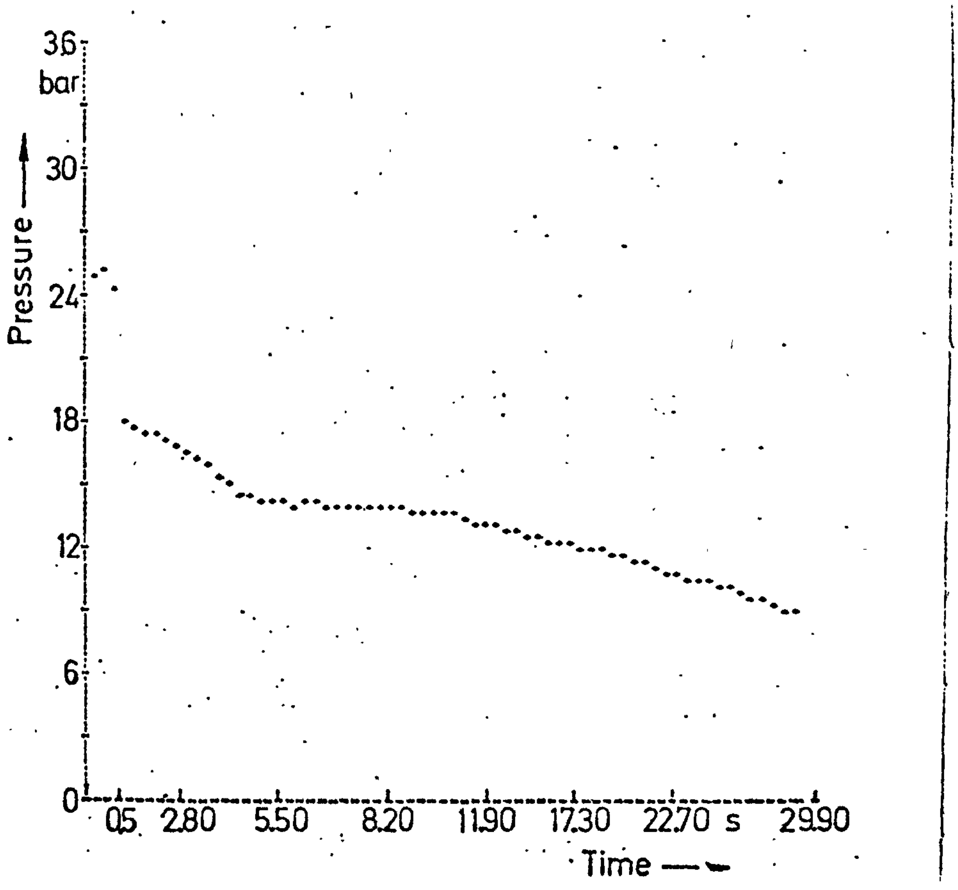


Fig. 14: Upper plenum system pressure, = 3,5 W/cm², 0,5 F break

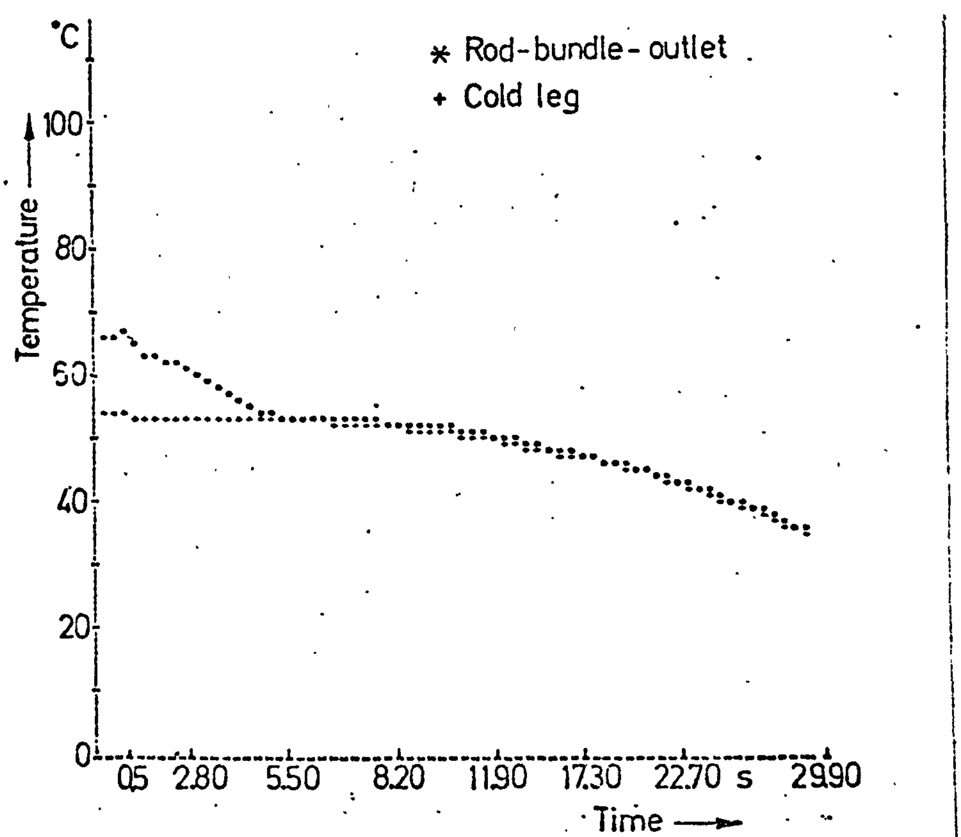


Fig. 15: Lower plenum and upper plenum fluid temperatures, = 3,5 W/cm², 0,5 F break

7. Next steps

To 3.1 For the entrainment measurements in a rod bundle the STORZ-lens will be installed in the R 12-blowdown loop.

Measurements of the liquid fraction entrained will be carried out in dependance of

- total mass flow rate at initial steady state conditions
- break area
- and initial void fraction

to verify the test-results gained at an inside cooled tube.

To 3.2 Mixing measurements at steady state and transient conditions will be performed with boiling Freon 12 in the two channel test section. The investigations of the physical understanding of mixing-processes will be continued to evaluate a mixing calculation model.

An own theoretical mixing model will be evaluated in the next time, and this model will be tested with the aid of the MIT modified COBRA III-C code.

To 3.3 Investigations concerning the influence of the steam generator volume and pump resistance are planned for the next time.

Furthermore the application of a fibre glass optical probe for a direct internal observation of the flow behaviour during transient LOCA-conditions will be tested.

8. Relations with other projects

RS 37, 37-1, 37-2

Investigations of the events within the reactor core under LOCA and emergency cooling conditions at KWU Großwelzheim

RS 48

Theoretical and experimental investigations in model laws for instationary heat transfer conditions in water cooled reactors under emergency cooling conditions.

RS 64

Investigations of steady state and transient critical heat flux of multirod bundles for PWR's and BWR's with Freon.

RS 179

Phaseseparation

RS 81

Boiling-mixing

01.01. - 31.12.1977

9. Reference dominant

/1/ Annular Report BMFT FB RS 163-01 in the series IRS-Forschungsberichte

/2/ Quarterly reports BMFT RS V75/1....4

/3/ W. Belda, F. Mayinger
Calculation model concerning dryout delay time in nuclear reactors
Dr. Thesis at the IfV 1975

/4/ Hewitt, G.F., Bennett
Studies of burnout in boiling heat transfer to water in round tubes with non-uniform heating
AERE-R 5076, See also Trans. Inst. Chem. Eng. 45(8) Oct. 1967

/5/ Rowe, D.S.
COBRA-IIIC: A digital computer program for steady state and transient thermal hydraulic analysis of rod bundle nuclear fuel elements BNWL-1695, 1973

10. Degree of availability

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

11. Budget

This report period: 238.000,00 DM

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1977	1.1.2	RS 135
Vorhaben/Project Title Entwicklung von Meßverfahren zur Bestimmung transienter Massenströme (Wasser/Dampf) durch Signalkorrelation Development of methods for measuring transient two-phase flows (steam/water) by signal correlation		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Technische Universität Berlin Institut für Kerntechnik
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.Januar 1975	31.Dezember 1977	Prof.Dr.-Ing.U.Wesser
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
continuing: Projekt RS 135 A	Dec. 31, 1977	465.615,-- DM

1. General Aim

The purpose of this project is the 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 gamma or x-ray beams.
The determination of the fluid velocity is based on measuring the transit time of variations in fluid temperature between two points along the direction of flow. The transient time is determined by using cross correlation techniques, while the temperature fluctuations are detected by thermocouples.

3. Research Program

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

4. Experimental Facilities, Computer Codes

There is a one- und two - phase-flow water loop for low fluid velocities up to 5 m/s which is used to study and calibrate the measurement apparatus.
Beyond that a small blowdown facility (50 bar) was built up to

run the measurement assembly under real blow-down conditions.

5. Progress to Date

The original project (RS 135) is terminated with the end of 1977, but was extended now until the end of 1978 (RS 135 A), to make more tests especially at the Joint-Two-Phase-Flow-Test-Loop at GfK-Karlsruhe.

The development and construction of four amplifier-systems for the thermocouple-signals is finished. Each consists of one preamplifier located nearby the test section and one main amplifier wich can be located outside the test facilities. The preamplifier consists of a low-noise high gain amplifier coupled with an isolation amplifier to break troublesome ground loops. The main amplifier on the other hand consists of an electronic network to improve the response-time of the thermocouple, a low-pass-filter (cut-off-frequency 500 Hz) and an automatic-gain-control (AGC) to fit the output signal to the tape-recorder input. Also the computer software developed for Time-DATA-Series analyzer to evaluate the time depented fluid density, fluid velocity and mass-flow-reate is completed.
Although the method to measure two-phase-mass-flow-rates canyet be used, there are some questions which cannot be answered now with accuracy. Concerning the fluid velocity measurement this is especially the influence of optimal thermocouple distance and configuration and the range of void-fraction and pressure within the cross-correlation-method is applicable.

6. Results

First experiments with both density and velocity apparatus under blow down conditions showed very encouraging results. The integral over the time function of the Mass-Flow-rate differs less than 5 % from the content of the upper vessel before starting the blow-down (figure 2).

The velocity measurement method was tested at the Joint-Test-Loop at GfK-Karlsruhe within a wide range of steam qualities (0-100 %) and pressure (10-100 bar). It was found that the results are sufficient for void fractions ranging from greater 0 % up to 50 %. Outside these bonds the results tended to become better at higher pressures. The insufficient results with pure water flows can be explained with the long distance to the next heat source and may be overcome.

7. Next Steps

The blow-down tests will be continued to evaluate both optimal distance and configuration of the thermocouples. Also some more measurements will be made at GfK-Karlsruhe to get better knowledge about the range of application (void-fraction,pressure flow-regime).
The work on the density measurement with scattered beam to evaluate density distribution will be continued.

8. Relations with Other Projects

Is is intended to calibrate the velocity measurement method under nuclear reactor conditions at a two-phase flow loop at GfK-Karlsruhe (RS 145).

Final aim is the mean mass flow measurement at the blow-down loop of EURATOM-Ispra (RS 109).

9. References

/1/ Lübbesmeyer, D., Ulber, M.:
Messungen von Fluidgeschwindigkeitstransienten auf der Basis von Kreuzkorrelationsanalysen
atw 22 (1977) 5 S.271-273

/2/ Ulber, M., Lübbesmeyer, D.
Erfassung und Verarbeitung von Temperatursignalen bei Blow-down-Versuchen
Vortrag FV-22
KTG-Fachtagung: Experimentiertechnik auf dem Gebiet der Reaktorthermo- und Fluiddynamik
Hannover, 28.2. - 2.3.1977

10. Degree of availability of the Reports

/1/ free
/2/ free (proceedings to the pubblished in 1978)

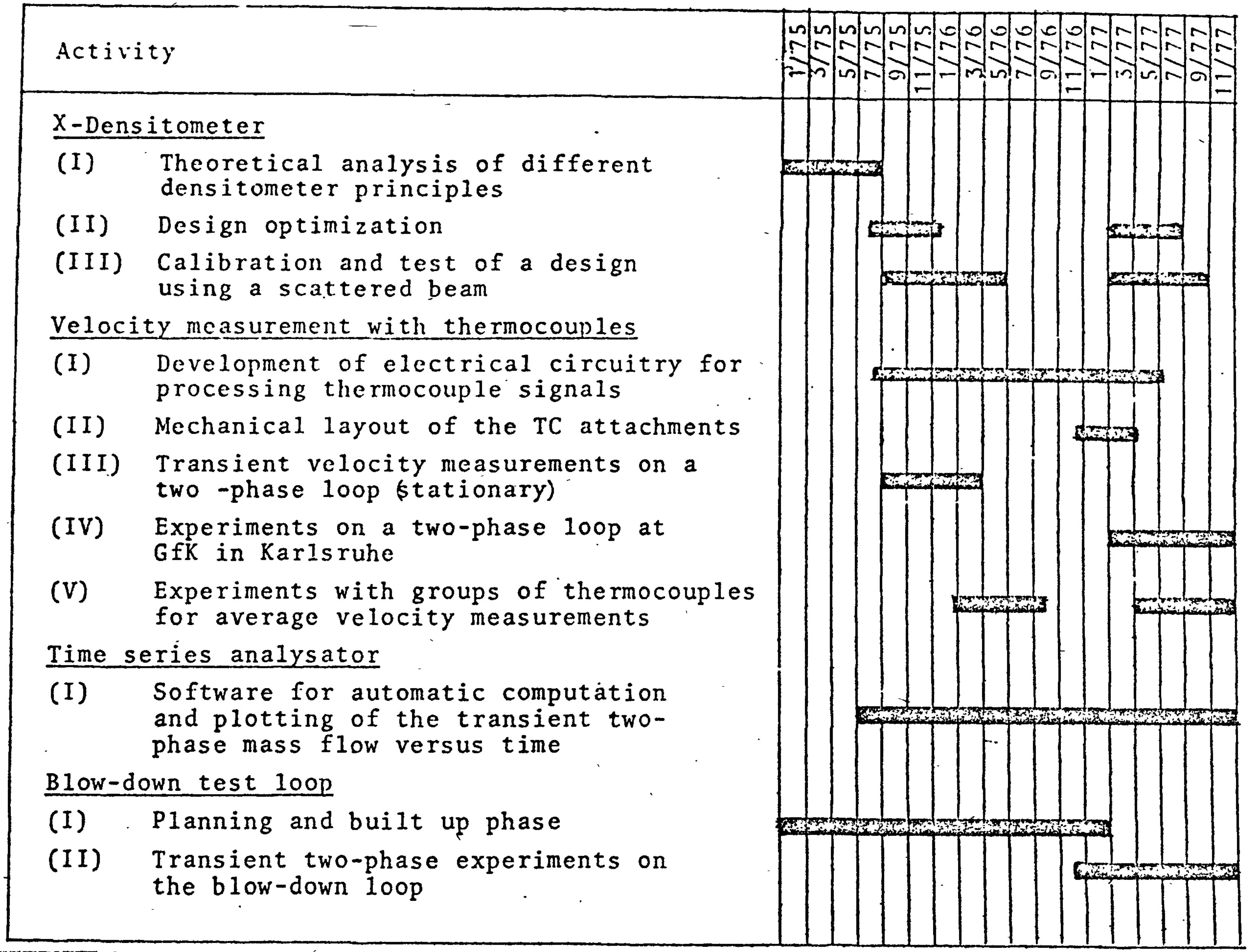
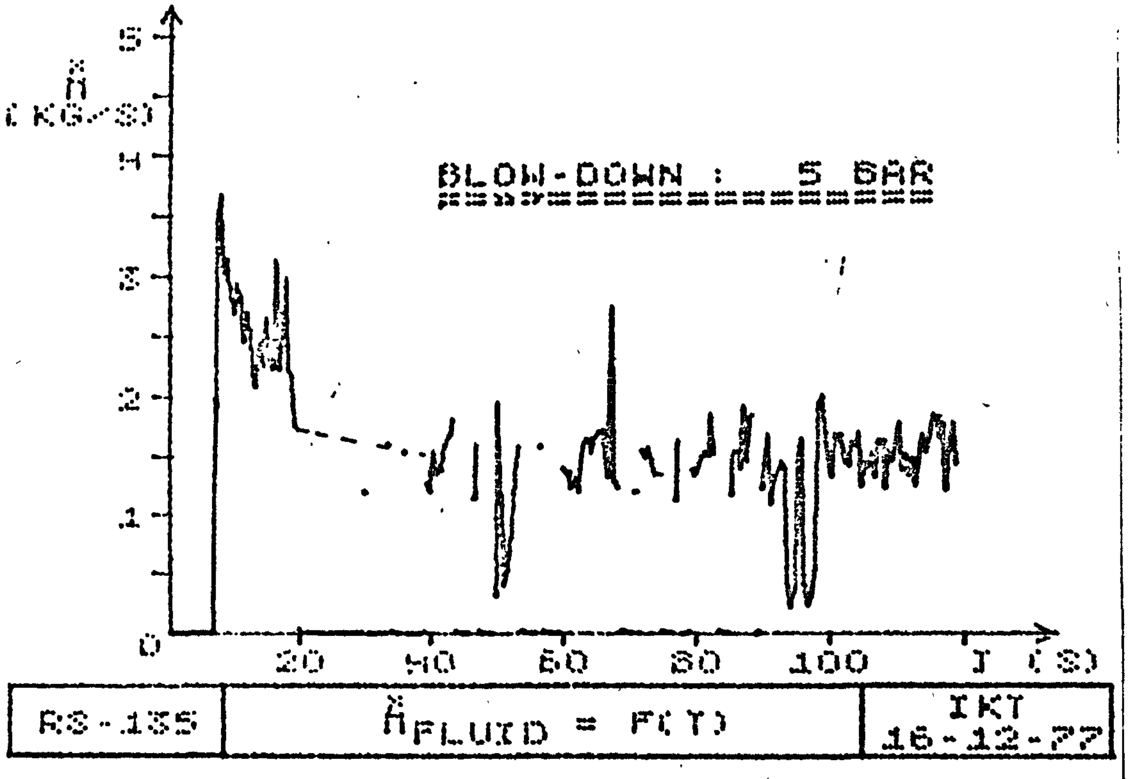


Figure 1: Schedule of the Research Program

1.1. - 31.12.1977



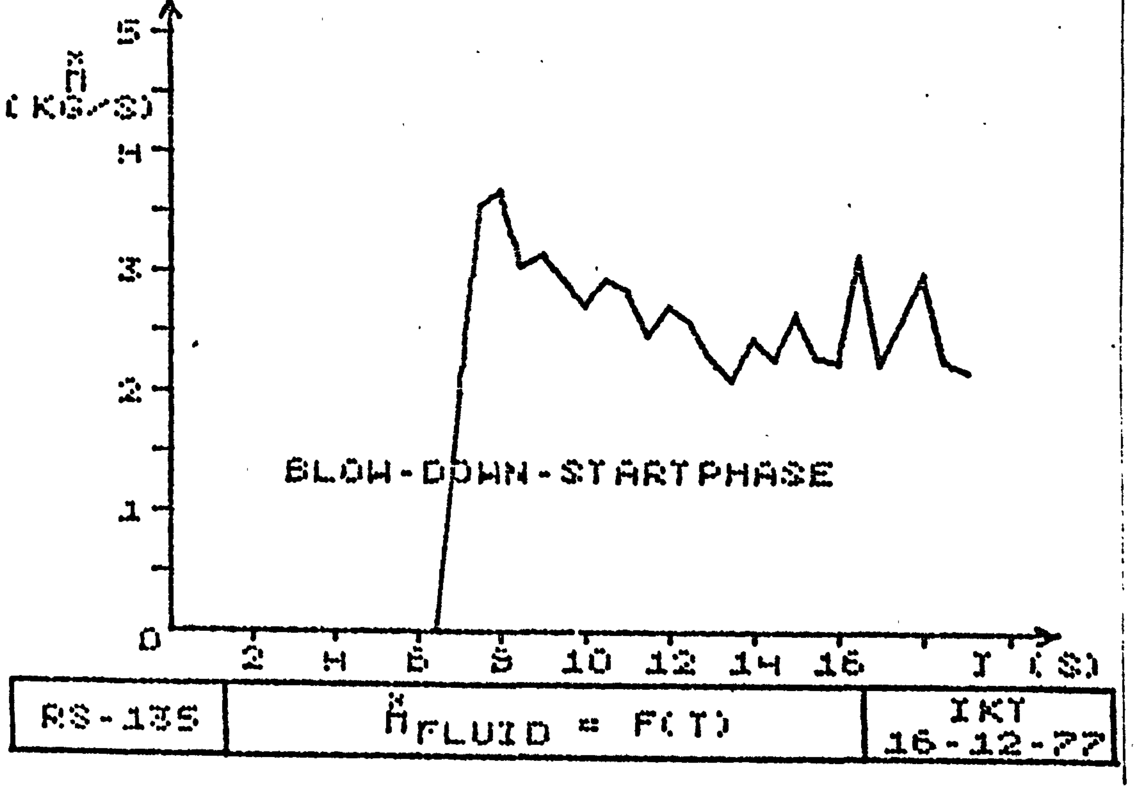


Figure 2: Mass-Flow over the time during the Blow-down

Berichtszeitraum/Period 1.1. - 31.12.1977	Klassifikation/Classification 1.1.2	Kennzeichen/Project Number RS 145/PNS 4137 (4215)
Vorhaben/Project Title Gemeinsamer Versuchsstand zum Testen und Kalibrieren verschiedener Zweiphasen-Massenstrom-Meßverfahren Joint Test Rig for Tests and Calibration of Different Methods of Two-Phase Mass Flow Measurement		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KfK Projekt Nukleare Sicherheit IRB
Arbeitsbeginn/Initiated Sept. 1974	Arbeitsende/Completed 1978	Leiter des Vorhabens/Project Leader J. Reimann
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating Dec. 1977	Bewilligte Mittel/Funds DM 560.000,-

1. General Aim

Test of two-phase mass flow measuring devices for LOCE.

2. Particular Objectives

Different measuring methods that are being developed in other instituts are to be tested and calibrated in steady-state steam-water and air-water flow.

3. Research Program

3.1 Test of methods sponsored by the BMFT (Federal Ministry of Research and Technology).

3.2 Tests of methods developed for the LOFT- and Semiscale-Experiments; sponsored by the USNRC.

4. Experimental Facilities

For these tests, loops for steady-state steam-water flow and air-water flow have been built.

5. Progress to Date

To 3.1: The following methods have been tested:

- Two pairs of free field turbine meters and drag discs together with a 2 beam densitometer (RS 109, Euratom Ispra).
- Temperature noise signal correlation (RS 135, IKT-Technische Universität Berlin).
- Radiotracer technique (RS 146, KFK-LIT)
- Dragbody (RS 147, Battelle Frankfurt)

Fig. 1 shows schematically the position of the various measuring devices.

1.1. - 31.12.1977

To 3.2: The LOFT free field DTT togehter with a 3 beam densitometer was tested both in a 5" and a 3" test section. Tests with the semiscale full flow DTT together with a 2 beam densitometer were carried out in a 3" test section. The greatest part of the experiments was performed with a drag screen, the rest with a drag disc. For determining the accuracy of the 2 beam densitometer a low energy scanning densitometer was used. In the semiscale tests the test section inlet was varied.

For detecting flow regime and measuring local void fraction different KFK-IRB-impedance probes were used: for the 5" pipe a transversable probe, for the 3" pipe 2 fixed probes. In a part of the experiments the KFK-LIT radiotracer technique together with a 6 beam densitometer was tested too. Fig. 2 shows the various test configurations.

6. Results

Tab. 1 and 2 show the test matrix for the different test series; the evaluation of the numerous test data is under way.

7. Next Steps

- Comparison of several densitometers (scanning-, KFK-LIT-, KFK-IRB-LIT-densitometer) with the 50 mm test section.
- Further experiments with the LOFT instrumentation.
- Further experiments with the RS projects and the true mass flow meter (KFK-IRE).
- Evaluation of test data.

8. Relation with Other Projects

RS 109, 135, 136, 146, 147, PNS 4236, LOFT (INEL), Semiscale (INEL), PBF (INEL).

9. References

Reports in the series GRS-Forschungsberichte, Report KFK 2500, 1977.

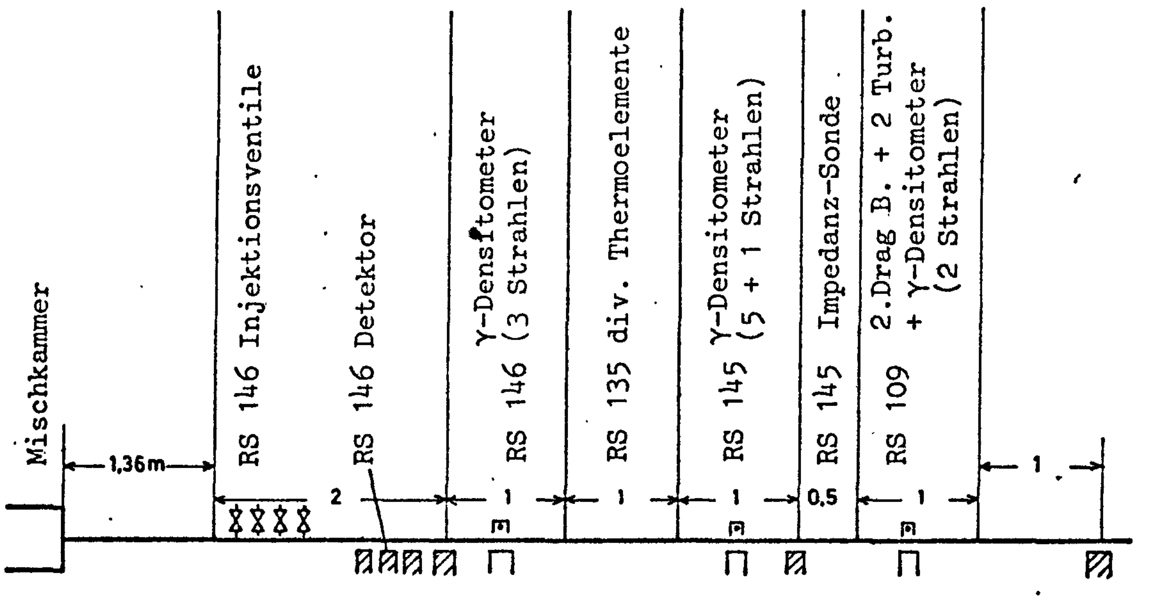


Fig. 1 Schematic Diagram of Test Configuration (RS-Projects, September 1977)

c_d (m/s)	p = 10 bar, c_w (m/s): 2,0	1,0	0,5	0	p = 25 bar, c_w (m/s): 2,0	1,0	0,5	0,25	0	p = 50 bar, c_w (m/s): 2	1,0	0,5	0,25	0
80						a,b	a,b	a,b						
60						a,b	a,b	a,b			a,b			
40	a,b,c	a,b,c	a,b,c			a,b	a,b	a,b	a,b		a^2,b,c	a^2,b,c	a,b	a,c
20	a^2,b,c^2	a^4,b,c^4	a,b,c	a,b,c	a,b,c	a,b,c	a,b,c		a^2,b,c	a^2,b,c	a^2,b,c^2	a,b,c	a,b	a^2,b,c
10	a^2,b,c^2	a^3,b,c^3			a,b,c	a,b,c	a,b,c			a^2,b,c^2	a,b,c	a,b,c	a,b	
5	a,b,c	a,b,c			a,b,c	a,b,c	a,b,c			a,b,c	a,b,c	a,b,c	a,b	
2,5					a,b,c	a,b,c				a,b,c^2	a,b,c	a,b		
0	a^3,b,c^2				a^2,b,c	a,b				a^3,b,c	a,b			

c_d (m/s)	p = 75 bar: 2,0	1,0	0,5	0	p = 100 bar: 2,0	1,0	0,5	0,25	0
20		a,c	a,c	a,c		a	a		a
10	a,b,c	a,b,c	a,b,c	a,b,c	a,b,c	a,b,c			a,b,c
5	a,b,c	a^2,b,c^2	a,b,c		a,b,c	a,b,c			
2,5	a,b,c	a,b,c	a,b,c		a,b,c	a,b,c			
0	a,b,c				a,b,c				a

beteiligte Meßverfahren:
a:RS 109; b:RS 135; c:RS 146

Indices: bei Reproduzierungstests: Anzahl der Versuche

Parameter: superficial velocities c_w, c_d, Druck p

Tab. 1 Matrix of Steam-Water-Tests (September 1977)

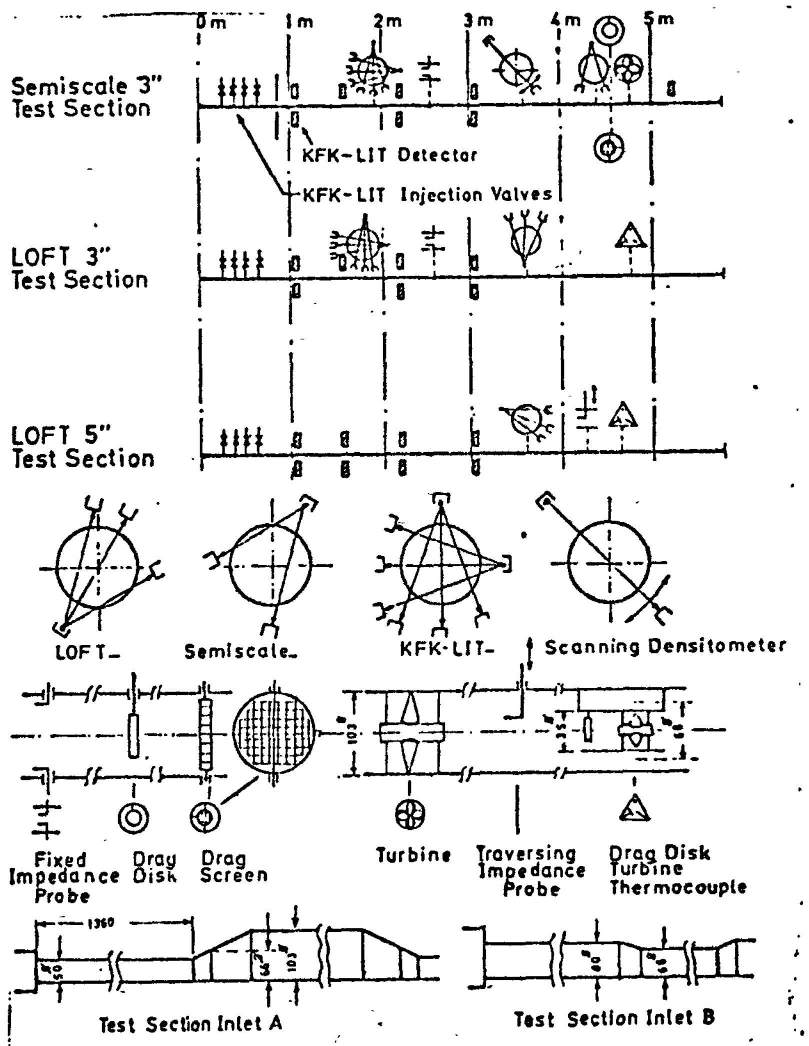
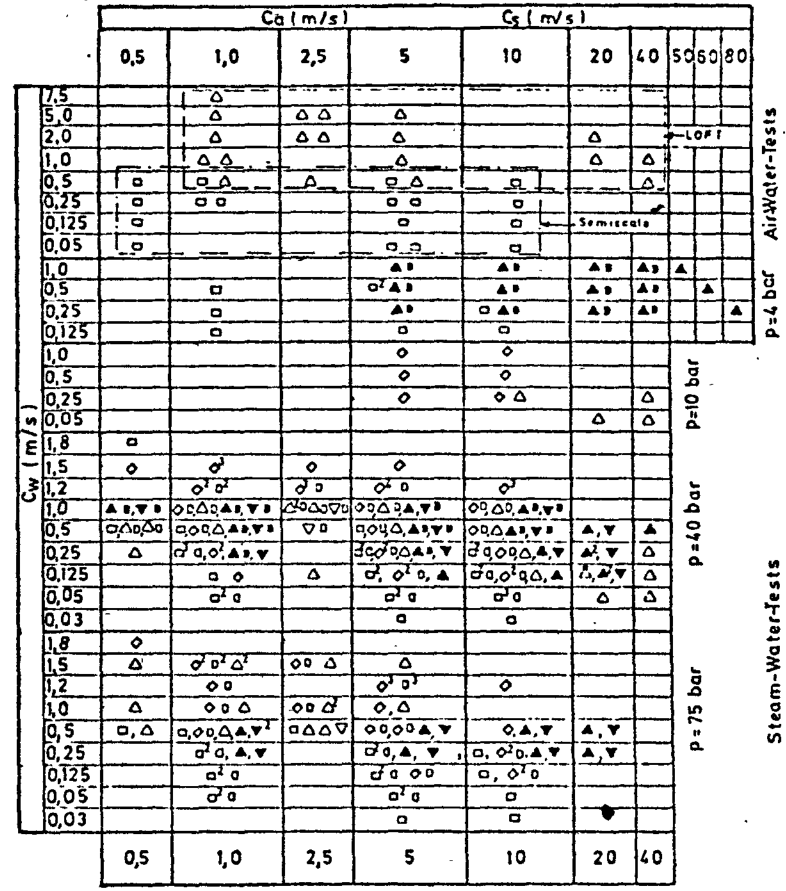


Fig. 2 Schematic Diagramm of Various Test Configurations (Test of LOFT-, Semiscale- and KFK-LIT-Instrumentation) October - December 1977



Symbol	Instrumentation	Test Section	Test Section Inlet
□	LOFT DTT	5"	A
◇	LOFT DTT	3"	A
△	Semiscale: Turb. + γ-Dens. + Drag Screen	3"	A
▲	Semiscale: Turb. + γ-Dens. + Drag Screen	3"	B
▼	Semiscale: Turb. + γ-Dens. + Drag Disc	3"	B
▫	KFK-LIT: Radiotracer + γDens.	5"	A
▫	KFK-LIT: Radiotracer + γDens.	3"	A
▪	KFK-LIT: Radiotracer + γDens.	3"	B

Tab. 2 Matrix of Air-Water- and Steam-Water-Tests (October-December 1977)

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.7. - 31.12.1977	1.1.2	RS 146/PNS 4136 (4214)

Vorhaben/Project Title	
Entwicklung eines Radionuklidmeßverfahrens zur Massenstrommessung in instationären Mehrphasenströmungen Development of a Radionuclide Method of Mass Flow Measurement in Non-Steady State Multiphase Flows	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Kernforschungszentrum Karlsruhe PNS/LIT

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.10.1974	31.12.80	R. Löffel
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1977	DM 827.000,--

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 and liquid phase using a radiotracer technique,
- 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 high time resolution (better than 100 msec) and shall be also applicable 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 Großwelzheim Superheat Reactor HDR).

2. Particular Objectives

2.1 Method of Radiotracer Velocity Measurement

Based on the transit time 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 dimension. Having passed an initial section, the radioactive tracer injected into the flow is recorded as an activity distribution plot at two measuring points placed in a staggered arrangement along the tube. The velocity is determined from the distance between the two measuring points and the transit time of the radiotracer. Periodic injection allows also a quasi-continuous measurement 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 densitites.

3. Research Program

The experimental program consits of:

- test of the radionuclide method under blowdownlike conditions on the "Joint Test Rig for Test and Calibration of Different Methods of Two-Phase Mass Flow Measurement" (RS 145/PNS 4215)
- preparation of the HDR-experiements (RS 123).

4. Experimental Facilities

Installation of a radiotracer two-phase velocity measurement device at the "Joint Test Rig" (RS 145) was terminated in March 1977. In October a Gamma densitometer was additionally installed.

5. Progress to Date

At the "Joint Test Rig (PNS 4215) for testing and calibrating different two-phase mass flow measuring techniques" roughly 250 tests were performed within the range of $p = 5 - 100$ bar with $\alpha = 0 - 1$ and $\dot{m} = 0 - 4$ kg/s.

6. Results

Velocities of 0.1 - 50 m/s for both phases and slip ratios of 1.0 - 7.0 were measured. A typical result has been represented as an example in the two diagrams.

Until completion and installation in the test rig of the γ-absorption density measuring system the two-phase mass flow was calculated from the steam and water velocities measured by the radiotracer method as well as from the pressure, temperature, and superficial velocities. The values so determined showed good agreement with the balance mass flows evaluated by the IRB Institute. It was demonstrated in these two-phase velocity measurements that the gaseous and liquid tracers mix well with the steam and liquid phase, respectively, and stay in this phase as a gas and liquid, respectively. Moreover, further knowledge was derived of the length of starting and measuring sections. Good results have been likewise obtained with the multiple beam γ-absorption density measuring system, which started operation in the fourth quarter, so that it is possible now to determine the mass flow directly from the velocities of the two phases and from the density of the two-phase mixture.

7. Next Steps

In 1978 the method will undergo final testing under conditions resembling blowdown at the "Joint Test Rig for Test and Calibration of Different Two-Phase Mass Flow Measuring Techniques" (PNS 4215/ RS 145). Further applications of the radionuclide method are planned for the HDR blowdown experiments from 1978 until 1980.

8. Relation with Other Projects

RS 33 — Joint Reactor Safety Experiments in the Power Station of Marviken, Sweden
GKSS, Geesthacht, 1971 - 1977

RS 109 — Experimental Investigation of the Influence of PWR-Loops on Blowdown

RS 123 — Safety Investigations performed at the decommissioned HDR plant

1.7. - 31.12.1977

RS 145 (PNS 4215) Joint Test Rig for Tests and Calibration of Different Methods of Two-Phase Mass Flow Measurement
GfK-IRB, Karlsruhe, 1974 - 1978

9. Reference Documents

Report KFK 1859 (1973) (German)
Report KFK 2050 (1974) (German)
Report KFK 2130 (1975) (German)
Report KFK 2195 (1975) (German)
Report KFK 2375 (1976) (German)
Report KFK 2435 (1976) (German)
Report KFK 2500 (1977) (German)
VDI-Berichte Nr. 254 (1976) (German)
Reports in the series GRS-FORSCHUNGSBERICHTE
KFK-Nachrichten, 9 (1977) No. 2, 38-42
Report EUR 5961e (1977)

10. Degree of Availability of the Reports

Unrestricted distribution.

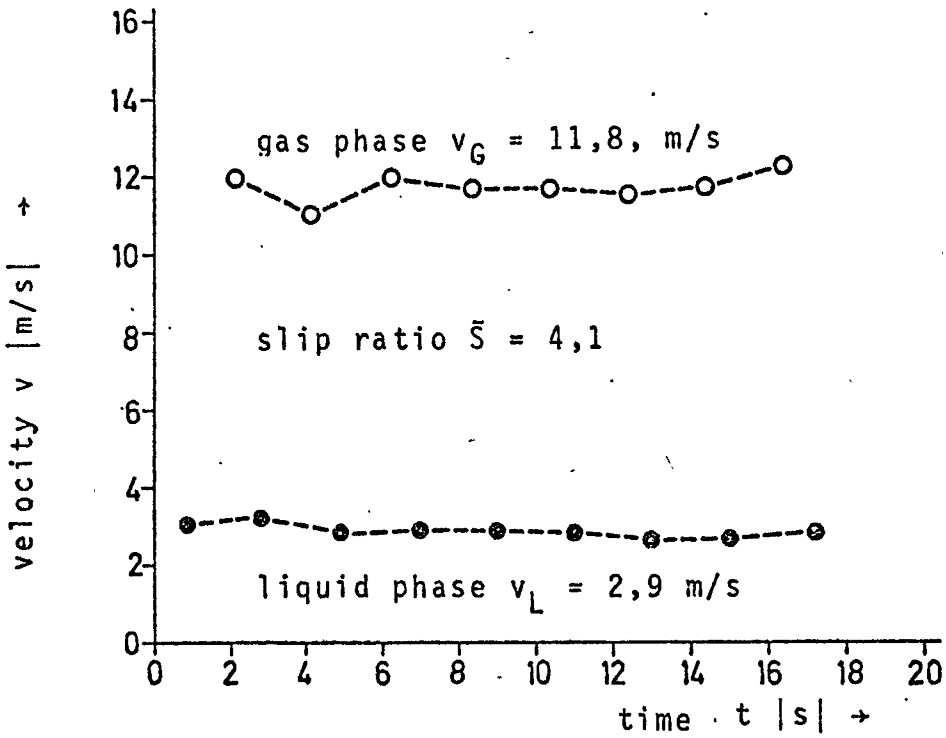


fig. 1: Velocity of gas and liquid phase (experiment 11)

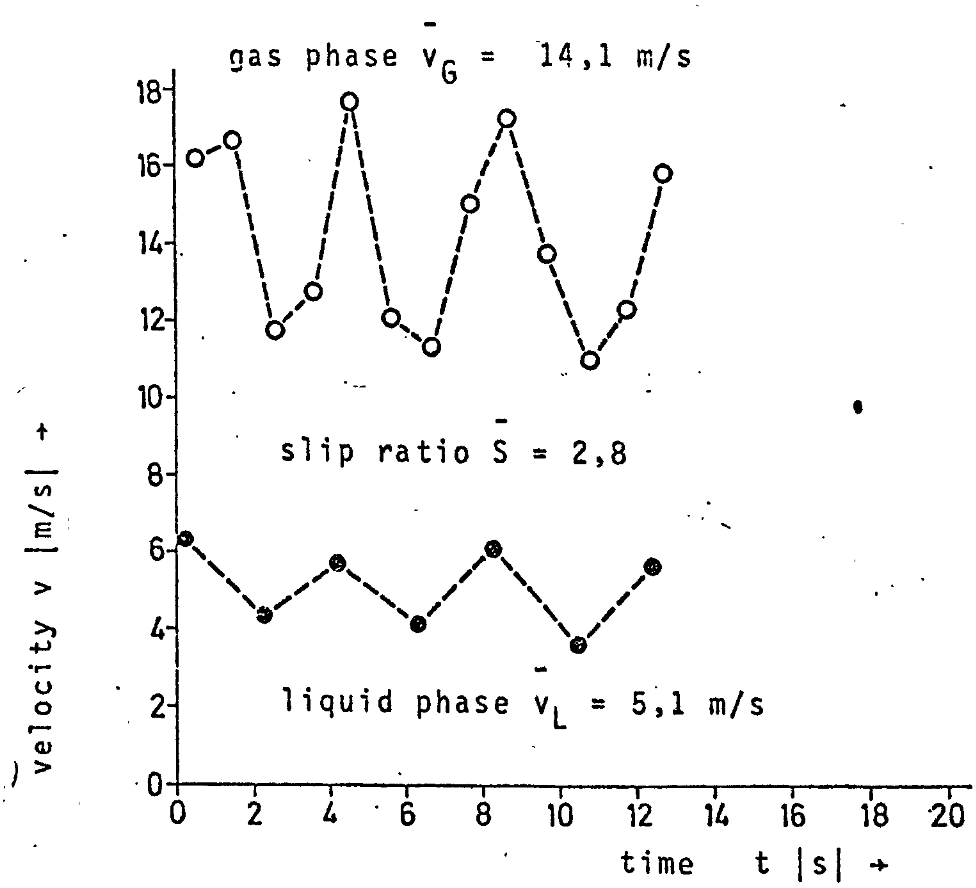


fig. 2: Velocity of gas and liquid phase (experiment 12)

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.77 - 31.12.77	1.1.2	RS 147

Vorhaben/Project Title	
Weiterentwicklung eines Drag-body für die Massenstrommessung bei Blowdown-Untersuchungen im Forschungsvorhaben RS 109 Improvement of a Drag Body for the Mass Flow Measurement in Blowdown Experiments of the Research Project RS 109	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: BATTELLE-INSTITUT E.V. Frankfurt am Main Abt. Energietechnik

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
Sept. 15, 1974	Dec. 31, 1977	G. Hampel
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	Dec. 31, 1977	DM 609.420,--

1. General Aim

The RS 147 project is aimed in particular at optimizing 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

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3. Research Program

3.1 Theoretical investigations of the dynamic behaviour of the measuring system.

3.2 Experimental investigations as a supplement to 3.1.

3.3 Manufacturing of four drag bodies of different measuring ranges for application in RS 109 and one drag body for application in RS 145.

3.4 Theoretical investigation of the drag coefficient of the drag body in unsteady incompressible flow.

3.5. Theoretical investigation of the magnitude of the drag coefficient of the drag body in compressible steady flow.

3.6 Analysis of thermodynamic effects on the measured signal.

3.7 Experimental investigation of the behaviour of the drag body in steady state two-phase flow.

4. Experimental Facilities, Computer Codes

Performance of the investigations covered by Section 3.7 in the two-phase test loop (RS 145) of the "Gesellschaft für Kernforschung, IRB, Karlsruhe". The work covered by Section

3.1 (e. g. calculation of the frequency response curve) will be carried out with the aid of the "DRAGBODY" program, which is based on the Finite-Element Method.

5. Progress to Date

Ad 3.7 An instrumented loop section 50 mm in diameter and 50 mm in length was designed and constructed, which is used in the two-phase test facility RS 145 of the GfK, Karlsruhe, for testing the drag body. The section was equipped with 1 drag body, 3 pressure transducers and 2 temperature transducers, subjected to a pressure test, and inserted into the loop. In cooperation with the other experimenters (projects RS 109, RS 135, RS 145 and RS 146), a common experimental program was developed. Since this program was based on different experimental conditions than had been originally envisaged by Battelle-Institut, it was necessary to define a new design range for the drag body and to make appropriate structural modifications. After several start-up trials, the actual measurements were performed.

Ad 3.1
.. 3.3 80 % of the first part of the report on the project was completed.

6. Results

Ad 3.7 Assuming a homogeneous phase distribution and using a calculated density value (the density values measured at GfK have not yet been evaluated), if was found that one of the measured results agreed within 5 % with the caluculated value. A preliminary evaluation of the second experiment was not possible without having more detailed information about the phase distribution(according to flow rate and steam proportion the phase distribution had to be heterogeneous).

In the course of the experiments it was found that this small drag body with small measuring range is not suited for long-term measurements. The measuring system was destroyed after a total operating time of about 15 hours, because the vibrations of the measuring system excited by the two-phase flow substantially reduce the fatigue strength of the drag body support. If the drag body was designed to have a strength enabling it to withstand the unexpectedly high cyclic stresses occurring in the experimental facility stand RS 145, it would no longer be possible to perform reasonable strain measurements with the type of strain gauge available. As the strain gauges have about the same size as the drag body, they have a decisive effect on the support stiffness and thus reduce the sensitivity of the drag body substantially.

7. Next Steps

After completion of the first part of the report, the elaboration of the second part will be started.

A continuation of the two-phase experiments in loop RS 145 is desirable.

8. Relation with Other Projects

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9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1o. 1977 - 31.12. 1977	1.1.2	RS 188

Vorhaben/Project Title	
Entwicklung einer Massenstromdichte-Meßmethode für transiente Zweiphasenströmungszustände mittels der magnetischen Kernspinresonanz Development of a mass flow density measuring method for testing transient two-phase-flow states by use of Nuclear Magnetic Resonance	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: RWTH Aachen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.4. 1976	28.2. 1978	Prof. Dr. R. Kosfeld
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
completed	31.12. 1977	366.4oo,-- DM

1. General aim

Development of a measuring method based on NMR, for determining the quasi-stationary and transient mass-flow density up to about $1o^3 g/cm^2 s$ in two-phase flows (water/vapor) with a time resolution up to the millisecond range and pressures up to 16o bar, temperatures up to 325^o C and flow velocities up to $1o^4$ cm/s. This goal should be attained in three consecutive steps; experiments in the first step should deal with temperatures up to $15o^o$ C, pressures up to 5 bar and flow velocities up to 1o m/s.

2. Particular objectives

The efficiency of an NMR method for determining the mass flow in two-phase flows up to the experimental conditions encountered in refilling and blow-down experiments should be ascertained.

3. Research program

3.1 Construction of a test apparatus for flow measurements covering fluid velocities up to 1o m/s
3.2 Conduction of measuring test series with the experimental setup.
3.3 Development of a transformation algorithm for evaluating the NMR signals
3.4 Aligning and testing the apparatus for T_1, T_2, D_s-measurements.
3.5 Measurements of relaxation times T_1, T_2 of water and vapor at high pressure and temperature
3.6 Estimation of signal strength and signal-to-noise ratio

1.1o. 1977 - 31.12. 1977

4. Experimental facilities, computer codes:

An apparatus for flow measurements within the range of fluid velocities up to 10 m/s had been constructed and put into service. A capillary for injecting nitrogen had been added. A pressure arming for the NMR probe coil designed to withstand pressures of 3oo bar and temperatures of 400°C had been built, and the complete apparatus for T_1, T_2, D_s-measurements had been assembled. Several FORTRAN programs for matrice inversion had been developped and tested. A library program had been modified to meet the requirements of the problems associated with the retransformation algorithm.

5. Progress to date

NMR measurements had been performed for water-nitrogen two-phase flows. Calibrating the meter window indicated that fluid velocities up to 8 m/s corresponding to a total mass flow of about 36 kg/min for single-phase flows had been obtained. After some constructive changes of the high pressure NMR probe head measurements of relaxation times T_1 and T_2 of water had been carried out. This led to an estimation of the sensitivity of a flow measuring apparatus as a function of relaxation time T_1, diameter of the tube, fluid velocity and density. Further work was done for developing a numerical method for solving the integral equation of the Fredholm type governing the retransformation algorithm.

6. Results

NMR signals of two-phase (nitrogen-water-mixture) as well as of single-phase water flows with fluid velocities up to 8 m/s had been recorded. Due to the model character of this test apparatus for the planned blow-down measuring apparatus equipped with a polarizing length of 1 m, the obtained signals are equivalent with a fluid velocity of about 4o m/s. Trying to solve the Fredholm integral equation for transforming the signals from the time domain back into the velocity domain by a numerical method showed that the equation system was ill-conditioned, so it had been modified for applying an improved numerical algorithm. Plots of signal strengths and signal-to-noise ratios as functions of fluid velocity and density tube diameter and relaxation time T_1 had been made, which allow simple determination of all interesting values. Measurements of relaxation times T_1 and T_2 of water up to 360° C had been obtained by the improved high pressure NMR probe head; temperature gradients within the head did not allow measurements in the range from 360° C up to the critical point where the vapor density is high enough to get reasonable signals.

7. Next steps

- Modification of the RF coil in the NMR head to obtain a better S/N ratio.
- Improvement of temperature and pressure control to eliminate the temperature gradients thus allowing measurements of relaxation times of vapor.
- Further numerical works on the retransformation algorithm.

8. Relation with other projects

No changes.

9. References

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10. Availability

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1977	1.1.2	RS 225
Vorhaben/Project Title Dichtemessung in Zweiphasenströmung (Wasser/Dampf) mittels Ultraschallsonden. Density-measurement in a two-phase-flow (water/steam) by means of ultrasonic detectors		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Technische Universität Berlin Institut für Kerntechnik
Arbeitsbeginn/Initiated Oct. 1, 1976	Arbeitsende/Completed Sept. 30,1978	Leiter des Vorhabens/Project Leader Prof.Dr.-Ing.U.Wesser
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating Dec. 31, 1977	Bewilligte Mittel/Funds 244.600,-- DM

1. General Aim

Development of a measuring method to detect density and density variations in fluid flows. Development of design-features of a fluid-level monitoring system by use of ultrasonic transducer probes.

2. Particular Objectives

The density and/or fluid level variations within stationary and transient temperature-, flow velocity- and density- conditions are to be measured. Simplification of the electronical and analysis requirements are investigated.

3. Research Program

The research program is divided into four (4) interconnected activities.

3.1 Design and Testing of alternative probe configurations.
3.2 Design and Optimization of transducers and electronic circuitry.
3.3 Density measurements (stationary/transient)
3.4 Fluid-Level measurements (stationary/transient)

4. Experimental Facilities, Computer Codes

The detector system will be tested and operated in different experimental facilities at the Institut für Kerntechnik

5. Progress to Date

5.1 Probe design

Two alternative probe configurations using extensional (33-) drive mode to detect fluid-level variations have been developed.

The conceptional design is based on fluid-level dependent ultrasonic radiation into the surrounding medium. Either the RMS-amplitude characteristics (continuous drive mode) or characteristic variations of pulse-shapes (broadening of the RF-pulse by refraction/pulse RMS-values) are indicative of the fluid level. The geometrical shape of the probes is either of helical rod type with driving wave-length an integer multiple of the geometrical data (radius of winding), or alternatively notched cylindrical rods with driving wave-lengths of specific "geometrical" resonances.
To detect local density (-variations) a "point"-probe was developed, the main feature of which are defined reflexion-areas.

5.2 Transducers-/electronics-design
To enable easy exchange-possibility as well as mechanical adjustment (tuning) the transducer configuration consists of mechanical pre-stressed extensional vibrators with impedance-matching probe coupling and conical backing with labyrinth-characteristics. Concerning the electronic requirements, at the moment extensive testing electronics is still in use to investigate optimum driving modes and analysis procedures.

5.3 Density-Measurements
The investigation of density-temperature-dependent variations of analysis-parameters in stationary and transient operational conditions is continuing to determine optimum operation mode and calibration requirements.

5.4 Fluid-level-monitoring
cf. 5.3

6. Results

6.1 Probe Design
Either cylindrical rod probes of helical shape as well as notched cylindrical probes of appropriate segment design are usable to detect fluid-level variations. The respective geometrical configuration to meet the requirements concerning active length may be varied within certain bounds. To avoid measurement disturbences by external or flow-induced excitation the preferred drive mode lies well outside the rod-and transducer resonances.

(fundamental modes), within the range of the "geometrical" resonances of the probes. The temperature-dependency of the resonance-parameters may be used in connection with known media to calibrate for density.

6.2 Transducer -/Electronic Design

Results up todate showed in principle the feasibility of continuous as well as RF-pulse driving mode. The operational frequences lie in the range > 100 KHz, thus avoiding the sensitivity of the probe system against external excitations-disturbences.

6.3 Density-Measurement

The density-determination is accomplished by attenuation-measurements caused by radiation/reflection at appropriate designed interfaces. Calibration data and definition of the parameter-dependency is continuing.

6.4 Fluid-Level-Monitoring

Reproducable results in stationary and nonstationary operating conditions could be achieved. The signal analysis might be based on pulse-characteristics as well as RMS-amplitude characteristics. The choise of the operational mode will be oriented on the electronical requirements.

In the case of notched rod-probes, the use of "geometrical" resonance-modes enables a well-marked segment indication, which may be used for calibration and reference.

7. Next Steps

The up to now achieved results at temperatures $\angle$ 40°C are to be verified at elevated temperatures and different density- and test conditions. To develop and adjust model-parameters calibration-data are determined. To avoid problems concerning the Curie-temperature, the extraction of the signals to be analysed via appropriate ultrasonic transmission lines is to be investigated.

The optimization-investigation concerning signal-analysis and electronic circuitry will be continued.

8. Relations to other Projects

It is intended to test the applicability of the measuring system in liquid-metal surrounding (RS 259)

1.1. - 31.12.1977

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.11.77 - 31.12.77	1.1.2	RS 277
Vorhaben/Project Title Vergleich erprobter und in Entwicklung befindlicher Massenstrommeßverfahren für Zweiphasenströmungen Comparison of Proved and Currently Developed or Planned Mass-Flow Measuring Methods for Two-Phase Flows		Land/Country FRG. Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor BATTELLE-INSTITUT E.V. Frankfurt am Main Abt. Energietechnik
Arbeitsbeginn/Initiated Nov. 1, 1977	Arbeitsende/Completed Oct. 31, 1979	Leiter des Vorhabens/Project Leader G. Hampel
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating December 31, 1977	Bewilligte Mittel/Funds DM 655.539,--

1. General Aim

The aim of this work is to provide a comprehensive survey of existing measuring methods and measuring systems for direct and indirect mass-flow measurements in thermohydraulic experiments in light-water reactor safety research. The mass-flow measuring methods for two-phase flow that have already been developed or are being planned are to be compared systematically, using specific criteria such as pressure and temperature ranges, detection of transients, phase distribution, etc.

The comparative description of the developed measuring methods is to be related to specific applications (experimental facility, conditions, results) and will help users of computer codes and experimenters in their evaluation of experimental data.

The compilation of the planned measuring methods is to provide decision-making aids in the further development and application of these methods in future research projects.

2. Particular Objectives

3. Research Program

The status report is to be drawn up in two phases:

- The first phase consists in the compilation of a manual (Catalogue) for tried two-phase mass-flow measuring methods for which first experiences and results have already been gathered.

- The second phase concerns new planned measuring methods that can be used in future thermohydraulic experiments in LWR safety research.

The following main steps are scheduled:

3.1 Collection of data on the tried measuring methods, the conditions for their application, and experimental results.

3.2 Collection of data on the planned measuring methods.

3.3 Establishment of criteria for the cataloging of the methods.

3.4 Review and evaluation of the data and study of the literature on the methods and their foundations.

3.5 Description of the experimental facilities where relevant tried measuring methods are already being used.

3.6 Description of selected experiments, their objectives, experimental conditions, and results.

3.7 Comparison of the measuring methods on the basis of application-oriented criteria (in the case of the developed measuring methods with due regard to 3.5) in a uniform and systematized survey.

3.8 Discussion of the results and final considerations.

4. Experimental Facilities, Computer Codes

5. Progress to Date

The following work was completed in the reporting period:

- Development of a detailed program for the comparison of the methods.
- Development of a detailed concept for the steps of the research program.

After this preliminary work, the research steps outlined in 3.1 to 3.8 can be undertaken.

6. Results

7. Next Steps

Continuation of the work according to the research program, starting with 3.1.

1.11.77 - 31.12.77

8. Relation with Other Projects

RS 16 B, RS 33 A, RS 37 A, RS 37 C, RS 48, RS 50, RS 81, RS93, RS 109, RS 123, RS 135, RS 136, RS 145, RS 146, RS 147, RS 161, RS 163, RS 179, RS 188, RS 195 plus other European and US-American projects.

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period 1.1. - 31.12.1977	Klassifikation/Classification 1.1.2	Kennzeichen/Project Number RS 182
Vorhaben/Project Title Beteiligung der GRS am LOFT-Programm der USNRC Participation of the GRS at the LOFT-Program of the USNRC		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH
Arbeitsbeginn/Initiated July 1976	Arbeitsende/Completed December 1980	Leiter des Vorhabens/Project Leader K.J. Liesch/F. Rohde
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1977	Bewilligte Mittel/Funds 5.523.150,-- DM

1. General Aim

The analysis of the LOFT series will improve the knowledge about certain regimes of the LOCA, which are not understood completely. In close cooperation with American research groups it is hoped to gain more experience in actual experimental procedures.

2. Particular Objectives

Verification of computer codes which are commonly used in Germany by pre- and post-calculations of the LOFT series, leading to modification of the codes where necessary. Comparison of the results of the LOFT series and of the German RS-projects and analysis of this comparison.

3. Research Program

The work of the GRS includes two main points:

- Pre- and post-test calculations of the LOFT experiments
- Participation in the performance and the analysis of the LOFT project.

4. Experimental Facilities, Codes

The codes DAPSY /1/ and WHAMMOD mainly are used to calculate pressure wave propagation phenomena. The codes BRUCH-D-06 /2/, RELAP4-GRS and DRUFAN /3, 4/ are applied to simulate the complete decompression process including the ECC injection process until the beginning of the reflood phase.

5. Progress to Date

The simulation of the L1-2 test with DAPSY within the short term range was completed by a very detailed description of the geometry and the fluiddynamic behaviour of the quick-opening-blowdown-valves (QOBV). A very good agreement between the calculation and the experimental data is seen in figure 1.

The WHAMMOD calculations for the subcooled blowdown phase were already completed in 1976 and are documented in /6/ and /8/. The measured and calculated data also correspond well.

The simulation of the L1-1 with BRUCH-D-06 within the long term range also yields a good comparison. The experience in applying the code to the LOFT primary circuit was enlarged by posttest calculations of the L1-3A test, obtaining good results, too. Small modifications of certain models of BRUCH-D-06 were made, in order to simulate the behaviour of the pressurization and the heat removal from the structures in a more realistic way. The calculation of the stored energy removal is the most important simulation of these complex processes which take place during the entire transient, in order to get a good agreement between calculations and experimental data. The pretest calculations of the L1-4 test with BRUCH-D-06 was completed in time for participation in the "blind" standard problem No. 7 of the USNRC, respectively No. 5 of the OECD. The participation was entirely successfull in both competitions.

The results of the posttest calculations had been improved by a rather detailed simulation of the heat removal even from the piping (see figure 2).

During the computation of the L1-4 posttest analysis with the code DRUFAN several modifications of certain physical models had to be carried out. In order to take into account the special requirements of the complicated test loop geometry and the physical processes occurring during the transient, the model for the heat addition to the fluid was improved as well as the procedure for switching to critical mass flow and vice versa.

It was the first time that the DRUFAN code has been applied to an integral test with a complex system and its important components. As expected from a code which considers thermodynamic nonequilibrium effects, no difficulties appear concerning the problems of ECC injection and in general the phase change of the fluid. The results of this simulation were submitted to the "open" standard problem No. 5 of the OECD (see figure 3).

The RELAP 4-GRS calculations submitted to the same "blind" standard problems showed good results for the blowdown phase, however, the code failed some seconds after the start of ECC-injection. Therefore refill and reflood were calculated in a different way. The code was modified for avoiding the problems induced by phase changes. A posttest calculation for LOFT L1-4 was done for the entire transient (figs. 4 and 5) and showed improved results.

6. Results

The results of the posttest calculations of the L1-2 test, computed with DAPSY /5/, WHAMMOD /6/, and RELAP4-GRS /7/ were presented at the Reaktortagung 1977, Mannheim, FRG.

The results of the pretest calculations of the L1-4 test, computed with BRUCH-D-06 and RELAP4-GRS had been submitted to the USNRC "blind" standard problem No. 7 and No. 5 of the OECD.

The results of the posttest calculations with the modified RELAP4-GRS-version and DRUFAN had been submitted to the "open" standard problem No. 5 of the OECD.

7. Next Steps

A detailed analysis of the L1-series should clarify some uncertainties with the experimental data and the computed results.

9. References

/1/ T. Grillenberger: The Computer Code DAPSY for the Calculation of Pressure Wave Propagation in the Primary Coolant System of Light Watereacotrs. Internal Report. MRR-I-66, April 1976

1.1. - 31.12.1977

/2/ K. Hofmann: BRUCH-D-06 - Ein Rechenprogramm zur Analyse transienter fluid- und thermodynamischer Vorgänge im Primärkreis von Druckwasserreaktoren oder in Versuchskreisläufen. Programmbeschreibung. MRR-P-25, Dezember 1976

/3/ K. Wolfert: The Simulation of Blowdown Processes with Consideration of Thermodynamic Nonequilibrium Phenomena, presented at the OECD/NEA Specialists' Meeting on Transient Two-phase, Toronto, August 1976

/4/ K. Wolfert: A new Method to Evaluate Critical Discharge Rates in Blowdown Codes that are Based on the Lumped-parameter Technique, presented at the Thermal Reactor Safety Meeting, Sun Valley, USA, August 1977

/5/ K.J. Liesch, K. Hofmann, F.J. Ringer: Die Nachbildung des Bruchöffnungsvorgangs im LOFT-Versuchsstand und Möglichkeiten der Simulation im Rechenprogramm DAPSY, presented at Reaktortagung, Mannheim, March/April 1977

/6/ M. Müller: Experimentelle LOFT-Ergebnisse der unterkühlten Blowdownphase und Vergleich mit Post-Test-Rechnungen, presented at Reaktortagung, Mannheim, March/April 1977

/7/ R. Ullrich: Vergleichsrechnungen zu den isothermen LOFT-Blowdownexperimenten, presented at Reaktortagung, Mannheim, March/April 1977.

/8/ R. Ullrich, F. Rohde, M. Müller: Analyseergebnisse zum nichtnuklearen LOFT-Versuch L1-2 (Nachrechnungen) GRS-A-7 (Februar 1977)

10. Degree of Availability of the Reports

Documents are available through Gesellschaft für Reaktorsicherheit (GRS) mbH, D-8046 Garching, Forschungsgelände, FRG.

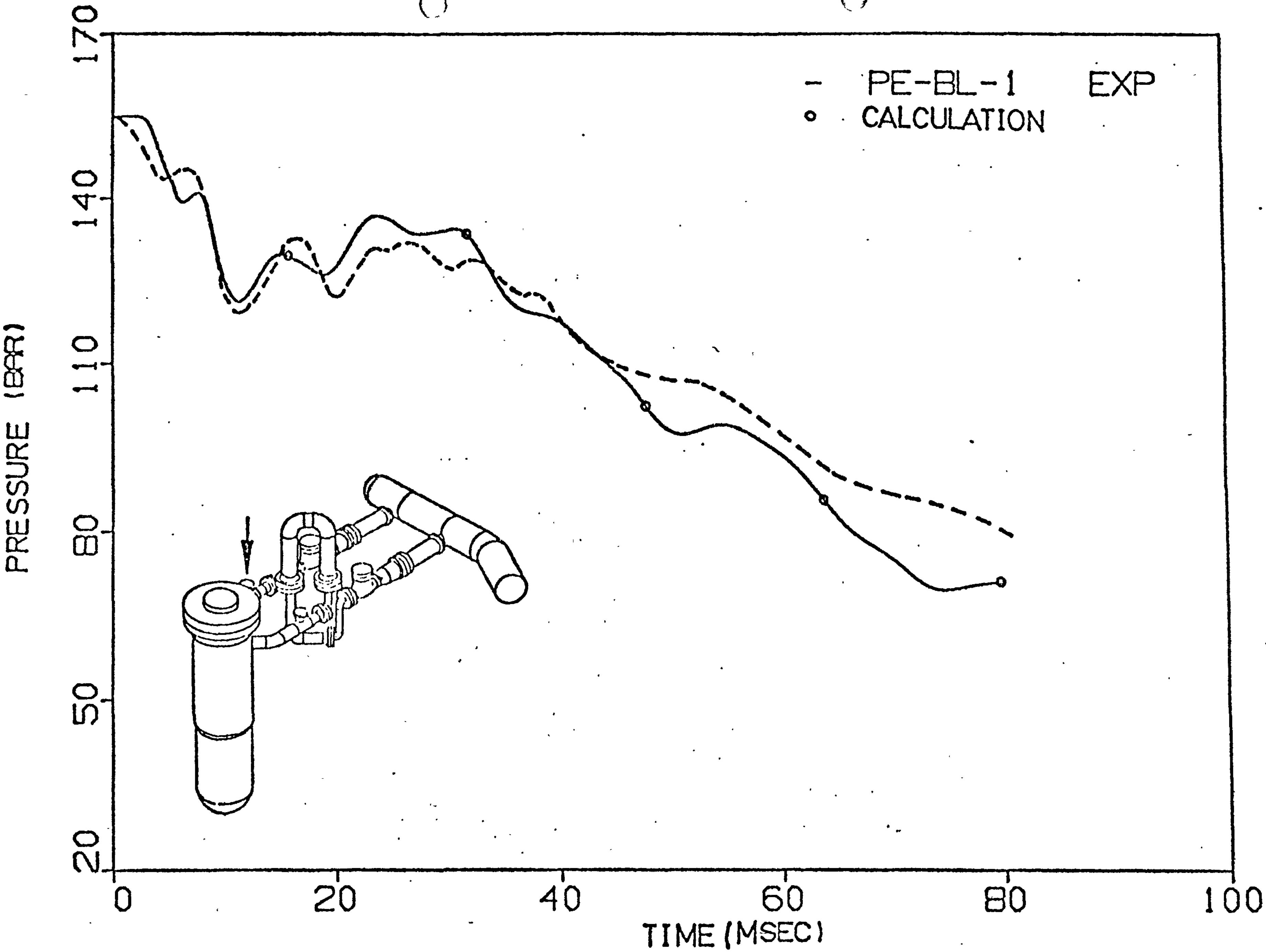

FIG.1 PRESSURE, BLOWDOWN LOOP, COLD LEG.

1.1. - 31.12.1977

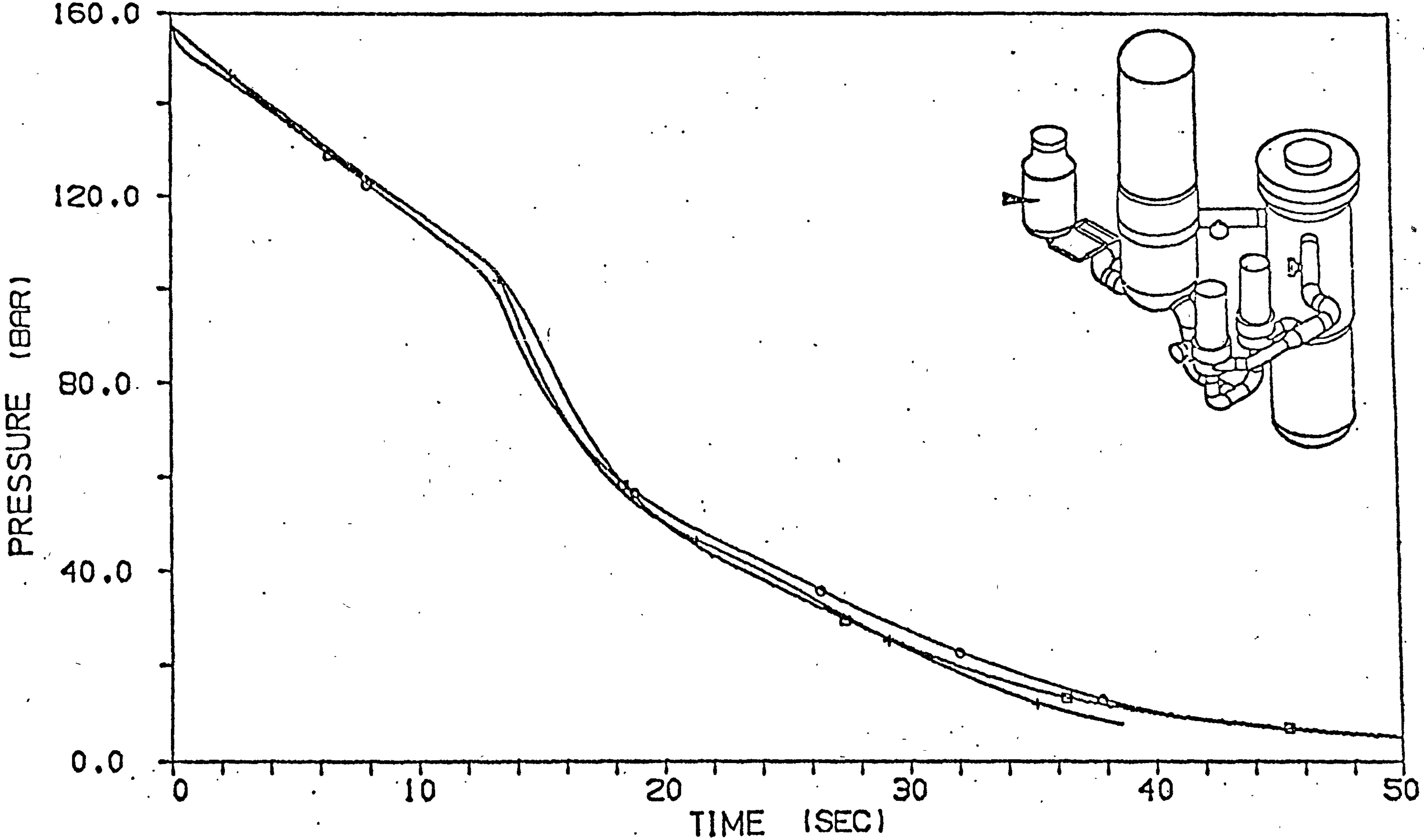

FIG. 2 PRESSURE, PRESSURIZER
▫ PE-PC-4 o PRETEST CALCULATION + POSTTEST CALCULATION

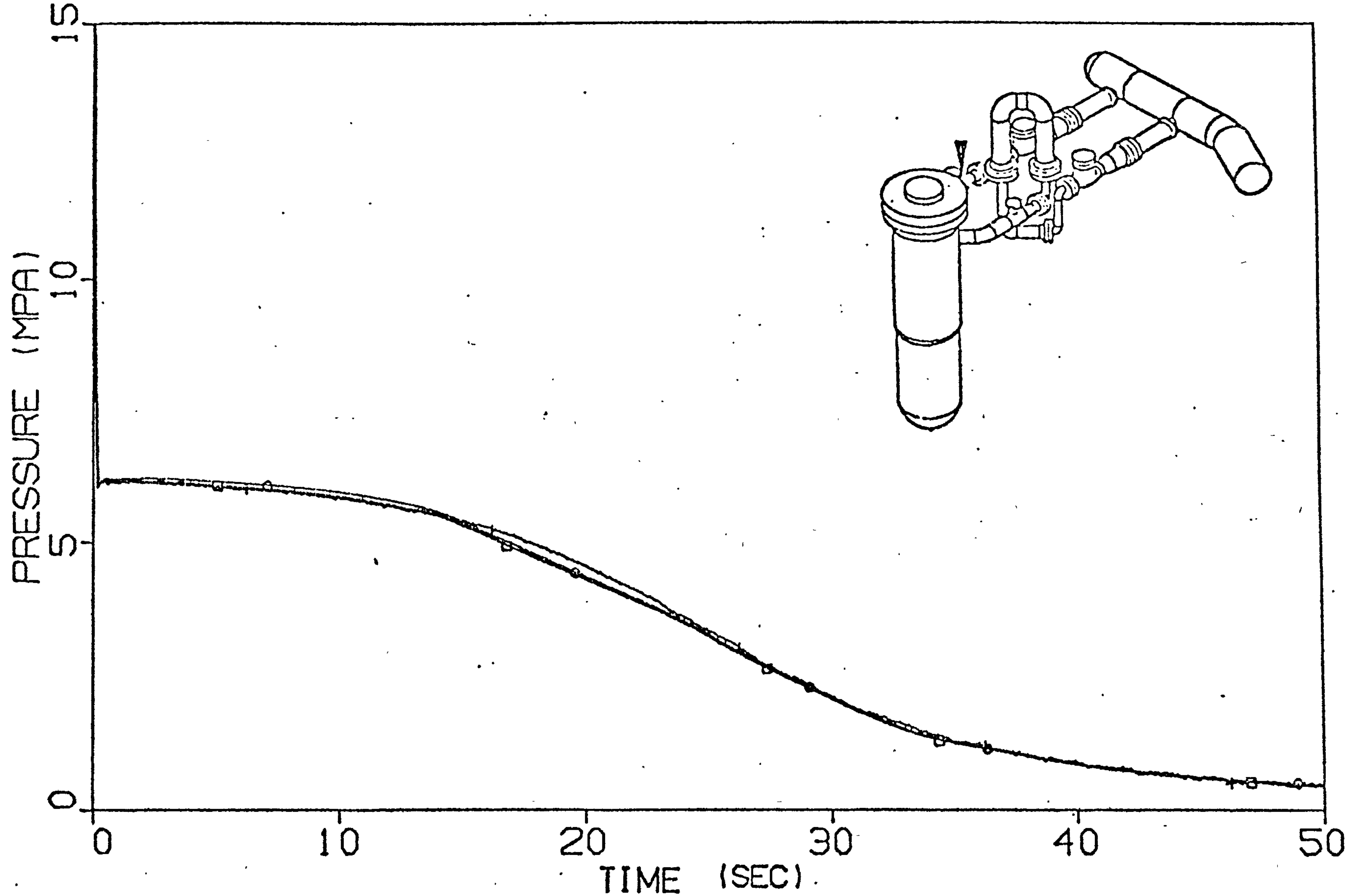

FIG. 3 PRESSURE, BLOWDOWN LOOP, COLD LEG
□ PE-BL-1 + POSTTEST CALCULATION

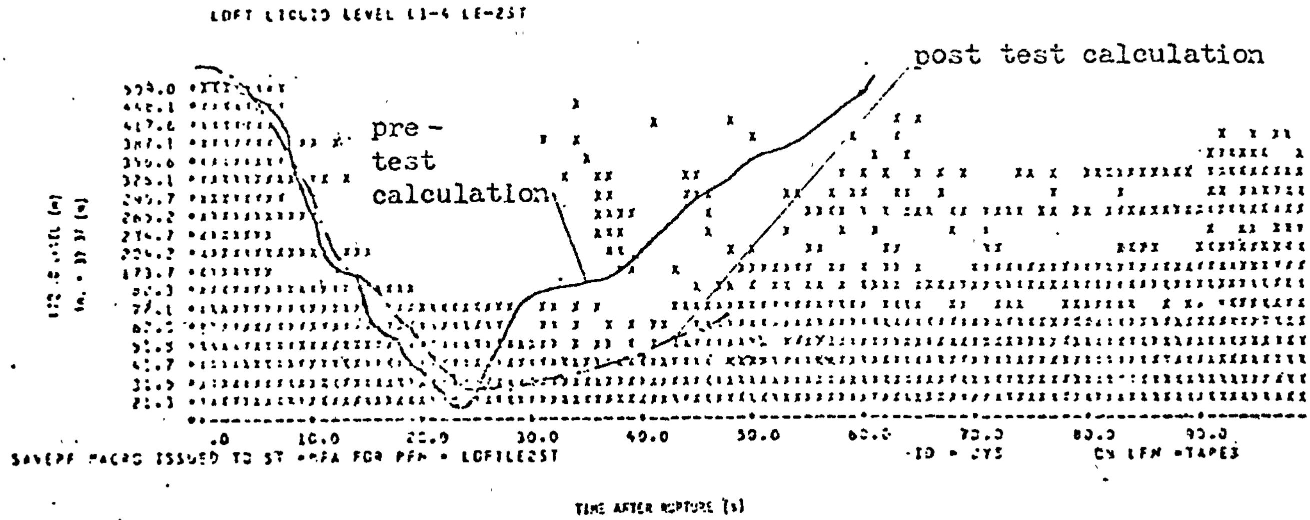

Fig. 4: LOFT L1 - 4,RELAP 4/GRS calculations for the liquid level at downcomer stalk 2

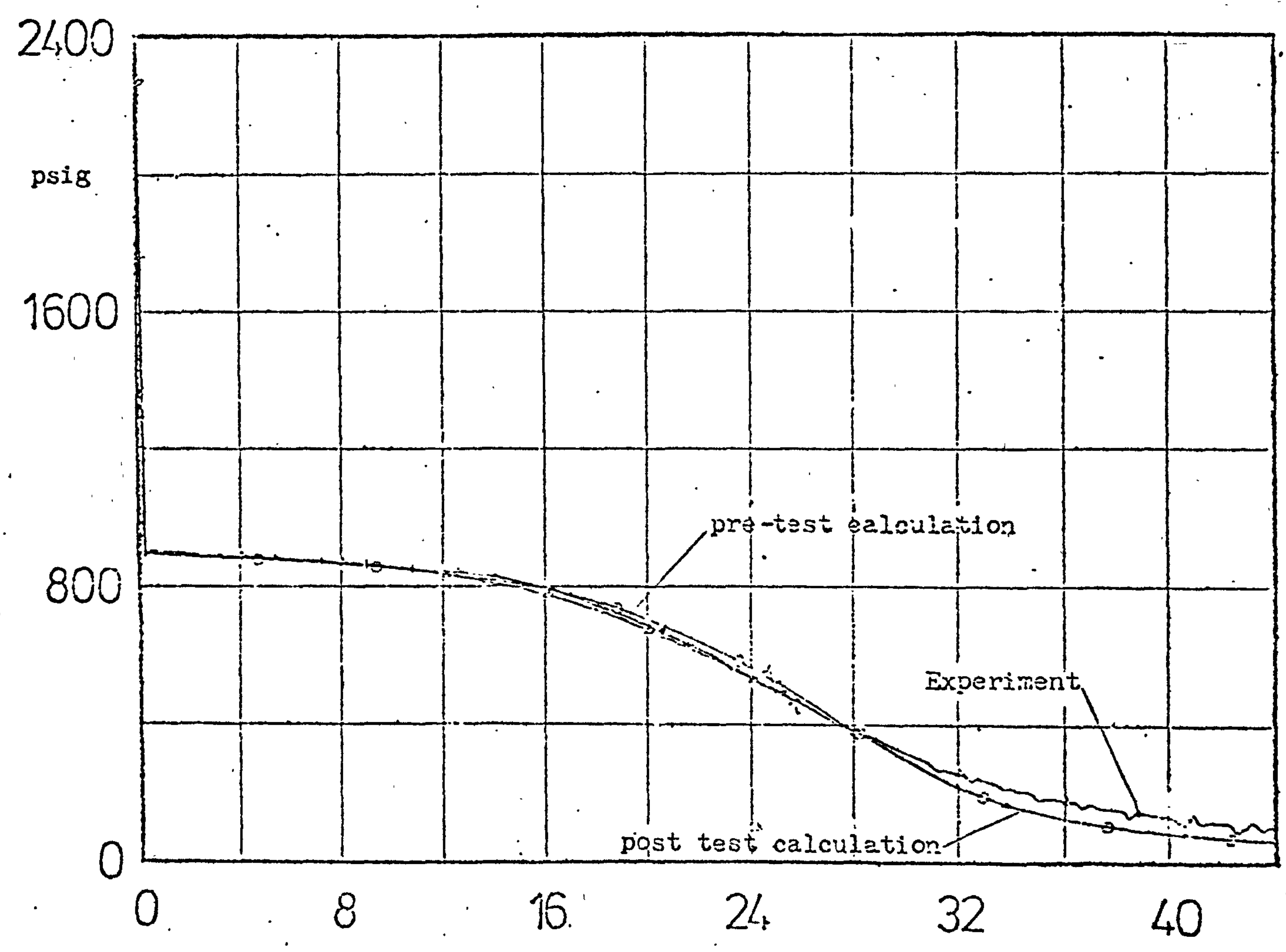

Fig. 5: LOFT L1 - 4,RELAP 4/GRS calculations for lower plenum pressure

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1 . - 31.12.1977	1.1.2	RS 179
Vorhaben/Project Title Erstellung eines theoretischen Phasenseparationsmodells zur Erfassung des Wassermitrisses und Vergleich der theoretischen Vorhersage mit experimentellen Ergebnissen Development of a theoretical phase separation model with respect to liquid entrainment. Comparison of theoretical prediction with experimental data		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor T.U. Hannover Institut für Verfahrenstechnik, Callinstr. 36 3000 Hannover 1
Arbeitsbeginn/Initiated Aug. 1975	Arbeitsende/Completed July 1978	Leiter des Vorhabens/Project Leader Prof. Dr.-Ing. F. Mayinger
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating Dec. 1977	Bewilligte Mittel/Funds DM 351.900.00

1. General aim

The main activities of this research work consist in developing a phase separation model, which is able to determine the interaction of vapour and liquid during a LOCA. In principle phase separation investigations at a free liquid surface have to be performed, considering the following mainly influencing parameters: Void fraction, vapour-liquid velocities and saturation pressure. Experimental analysis has to be done for heated and adiabatic flow with different geometries, using the modelling fluid R12 (CF_2Cl_2).

2. Particular objectives

With the activities a theoretical phase separation model has to develope leading to a better understanding of the fluid behaviour during LOCA conditions. This model should give a temporal and local description of the void fraction in the vessel and of the flow conditions at the outlet break. Special regard is to be given to the vapour generation in the mixture, the vapour separation at the mixture surface and the liquid carry over. This model should be simple enough to be used in a computer code to predict the man discharge, liquid level settlement and the pressure decrease. The theoretical work has to be supported by steady state and blowdown experiments, performed at two different test sections such as a pressure vessel model without core structure and furthermore a vessel including a 4-rod bundle simulating the reactor core.

3. Research program

The research program of this project includes three groups of investigations. In a first step the phase separation at a free liquid surface has to be investigated at steady state conditions. In a liquid filled chamber vapour is injected from the bottom. By experimental and theoretical treatment the vapour separation mechanism, droplet formation and droplet carry over at the liquid pool surface should be analysed.

In the second step the vapour escaping from the liquid and the liquid carry over behaviour has to be investigated for blowdown conditions, with special regard to the influence of various pressure transients. The third step serves to investigate the phase separation and droplet carry over in a simulated reactor core. For these tests, experiments have to be carried out in an electrically heated 4-rod bundle.

4. Experimental facilities, computer codes

All experiments involved in this research program are carried out by the use of the refrigerant R12 (CF_2Cl_2) as a modelling fluid instead of water substance.

For the tests a supplying loop containing the usual components (pump, condenser, evaporator, vapour superheater, pressurizer) was constructed to feed the different test sections.

Fot the steady state vapour injection experiments a cylindrical glass made vessel was constructed, allowing an optical and cinematographic investigation of the phase interaction phenomena. The vapour is injected from the bottom by seven nozzles, which can be opened individually and replaced by those of some different orifice diameters. An alternative form of vapour injection in a more homogeneous manner is given by installing a porous sinter plate at the bottom of the test section.

Beside its application to steady state experiments this vessel can be used for blowdown experiments, too. Beside the optical instrumentation the pressure vessel model is equiped with a number of measuring instruments as indicated in Fig. 1. The system pressure and the fluid temperatures are recorded by aid of a pressure gauge and several thermocouples. The mass discharge rate is indicated by a weight measurement using a force transducer. The time dependant mean void fraction in the vessel is picked up at different heights by aid of the γ-ray attenuation method. In front of the outlet orifice simulating the break cross section the average void fraction is detected by the γ-ray method, while the phase distribution is indicated by a capacitive probe, consisting of three plate condensators.

Beside this first test section a second one, likewise constructed from glass can be connected with the circuit alternately to be used for blowdown tests, too. This test set-up is equiped with a capacitive probe combined with a thermocouple rake to study the flash evaporation rate in comparison with the fluid temperature behaviour during strong pressure transients and to get an information on the thermodynamic non-equilibrium. In addition to these both vessel models an electrically heated 4-rod bundle arranged in a quadratic duct which itself is placed in a larger surrounding tube was constructed. This facility models are a part of the reactore core with a simulated downcomer. In the test rig it will replace the second vessel model.

For the theoretical description of the thermohydraulic fluid behaviour with special emphasis to the phase separation and liquid carry over a simple computer code is used consisting of a model for critical outflow and another one to describe the void fraction and quality distribution in the vessel and at the break location. With the help of that code different outflow models /1,2,3,4/ and some literature given phase separation models /5,6,7/ as well as an own model for the separation and carry over behaviour can be checked. For this purpose own experimental R12 data and some water experiments, performed at the Battelle Institute Frankfurt /8/ were verified.

5. Progress to Date

Within this reporting period the following activities were conducted.

a) Investigations under steady state conditions:
   Theoretical description of droplet carry over at a free liquid surface and separation behaviour of entrained drops in the vapour dome.

b) Blowdown tests:
   Optical investigation of the flashing behaviour. Parameter studies: Influence of the initial (before blowdown) liquid level, break area and system pressure on the pressure transient, mass discharge rate and void fraction distribution in the flashing mixture and at the break. Temperature behaviour of the fluid during pressure transients.

c) Activities in developing a phase separation model. Verification of own experimental data by use of some empirical approximations implied in the computer code. Converting the empirical factors, gained from R12 tests to water conditions by use of some scaling laws, the computer model could be checked when verifying test results from a standard test /8/.

6. Results

Steady state investigations:

Fig. 2 shows the separation mechanism of a vapour bubble at a free liquid surface and the formation of a droplet carried over into the vapour dome. The schematic drawing represents the profile of a rising liquid jet. From a literature research a system of equations could be found, describing the droplet generation and carry over process. These equations are indicated in fig. 2, too. The separation behaviour of entrained droplets in the upstreaming vapour on their way along the vessel height is presented in fig. 3. From a simple balance regarding the forces acting at a droplet in a vapour stream, a differential equation could be evaluated. For a given droplet diameter and vapour velocity the carry over condition could be described with the help of some boundary conditions indicated in fig. 3 the differential equation can be solved, and so an expression for the droplet rise velocity and for the rising height could be gained.

Blowdown investigations:

Fig. 4 shows the flashing of an initially saturated liquid pool in the first blowdown period. The series of these high speed photos demonstrate the foaming of the mixture level. It has to be mentioned, that the mixture surfaces moves upwards in a very uniform manner, comparable to a piston. Fig. 5 shows the transient pressure behaviour influenced by a variation of some system parameters. A variation in the initial system pressure gives no remarkable change in the qualitative pressure decrease, but quantitatively the slope in the late blowdown period decreases with decreasing initial pressure.

The variation of the initial liquid level showed an identical pressure decrease in the very first blowdown period due to the vapour expansion in the dome, but the expansion time decreases with the deminuation of the vapour dome volume.

The flashing behaviour is characterized by the evaporation rate. With increasing liquid level that rate is enhanced, leading to a higher re-increase of the system pressure. A reduction in the break cross section area decelerates the complete blowdown process and gives a slower pressure decrease as it is well known. This is due to an enhanced phase separation of the mixture causing an outflow of relatively high void fraction.

Fig. 6 shows the transient void fraction distribution at different heights of the test vessel and in front of the break, recorded from some γ-ray attenuation measurements. The initial conditions are indicated in the figure, and correspond to the optical investigations of fig. 4. Those signals received at positions below the initial surface increase continously within time. The other ones indicate pure vapour until they are reached by the flashing mixture. From these curves the flashing velocity is determinable. Furthermore these signals can be used to describe the void fraction distribution versus the mixture height at any time, to be compared with some phase separation models in the literature, for example that in the RELAP Code. In fig. 7 for the first 0.5 seconds that local distribution is indicated.

Besides the void fraction investigations, the time depending fluid temperatures were compared with the saturation temperature according to the pressure transient. Fig. 8 gives the deviation of the temperatures from equilibrium conditions. While the vapour in the dome is only slightly superheated, the liquid remain nearly at its initial temperature. The liquid superheating is reduced in a short time interval causing a violent flash evaporation which predominates the outflow rate, so that the system pressure reincreases. This two mechanisms flashing and pressure reincrease effect the liquid temperature to remove to the equilibrium conditions. Additionally in fig. 8 the principal blowdown behaviour is shown schematically. With the help of this idea the blowdown process can be divided into 4 main regions, governed by the fluid behaviour in the vessel.

Region I: Isentropic vapour expansion, flashing delay in the mixture. Outlet void fraction = outlet quality = 1. Single phase accelerated outflow model.

Region II: Flashing of the mixture, calculation of the evaporation rate with respect to the non equilibrium by use of an empirical factor. Rising mixture in the vessel. Single phase critical outflow until the mixture reaches the outlet tube.

Region III: Similar to region II but using the homogeneous model to describe the quality at the break. Homogeneous critical outflow model according the Köberlein /1/. Comparison of the evaporation rate and the outstreaming volumetric flow. If they are equal, change to region IV.

Region IV: Homogeneous mixture model in equilibrium conditions, critical outflow model of Köberlein /1/.

Fig. 9 shows a comparison of own experimental data and their verification by use of the elaborated theoretical model code. Due to an always inexact description of the exit void fraction in the late blowdown period by an under prediction of the phase separation the obtained agreement is not even satisfactory. It is tried to get a better verification by use of a modification in the separation model of the RELAP Codes.

With the help of the dimensionless evaporation number K

$$K = \frac{r}{c_p \cdot \Delta\vartheta}$$

r = latent heat of evaporation
c_p = specific heat
$\Delta\vartheta$ = temperature difference

and with respect to the critical flow rate differences in R12 and H_2O, the empirical constants considering the non-equilibrium could be scaled from our R12 tests to water conditions.

Introducing these values in our computer code, a vessel blowdown test of the Battelle Institute Frankfurt was recalculated. A comparison of experimental data and our calculation is shown in fig. 10.

The deviation in the later blowdown period results from the inexact prediction of the void fraction at the break. Unfortunately no representative experimental data could be obtained to be compared with the theoretical void fraction results.

7. Next steps

A modification of the phase separation model given in the RELAP code will be introduced in our code to give a better description of the temporal and local void fraction and quality distribution.

Some blowdown experiments using the 4-rod bundle test section will be performed to investigate the influence of an additional heat input to the flash evaporation, phase separation and void distribution.

These tests will be calculated by use of the computer model.

8. Relation to other projects

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9. References

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10. Degree of availability of the Reports

/1/ GRS Garching

/2/ - /14/ free

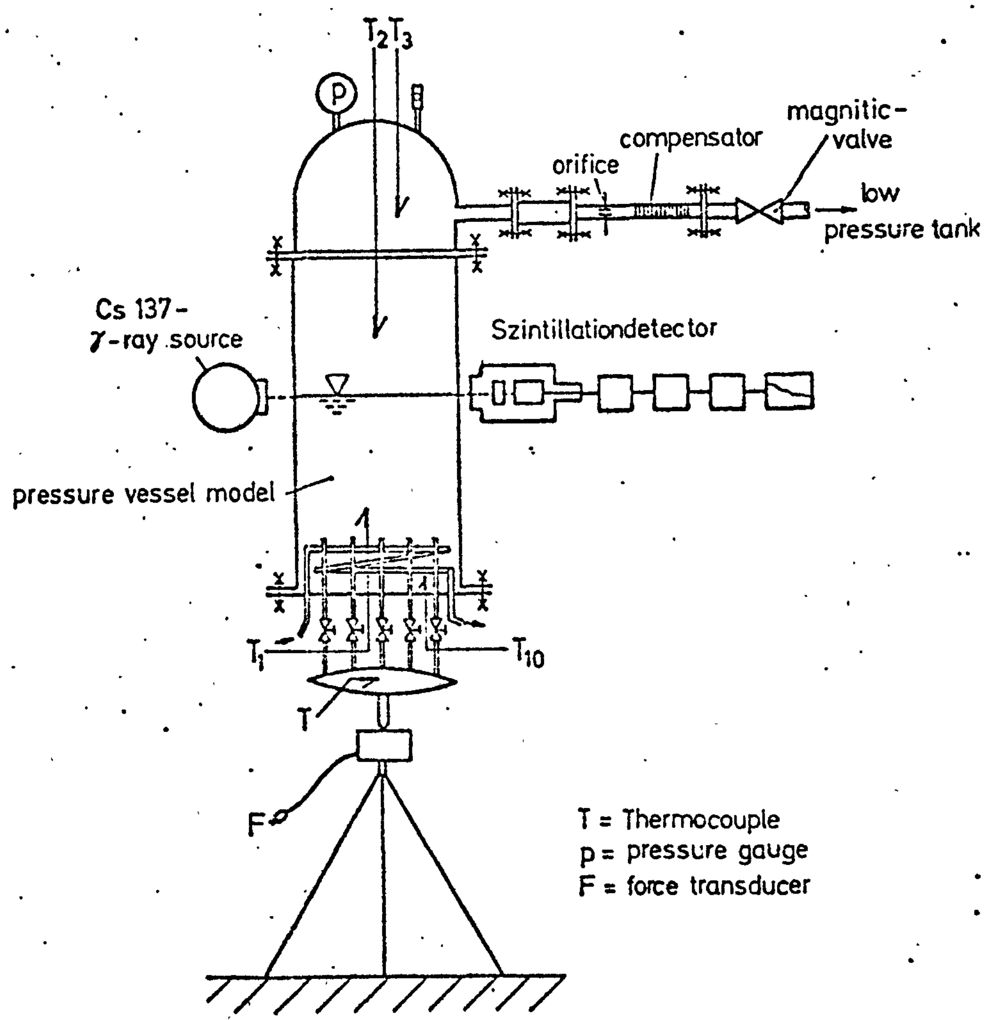
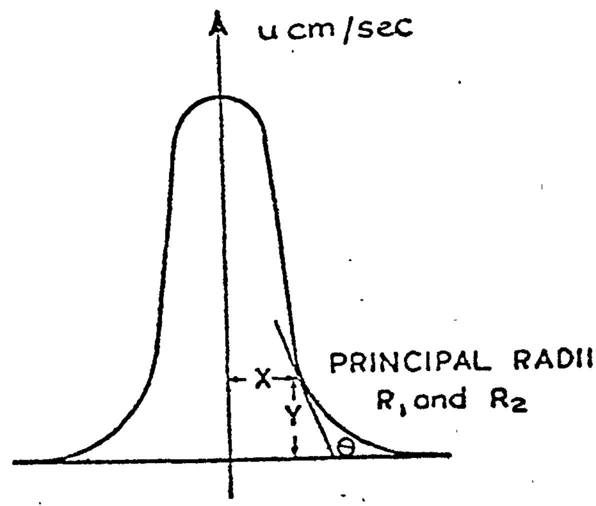


Fig. 1: Instrumentation of the pressure vessel model

1.1 . - 31.12.1977



—Profile of a rising jet

<u>Bubble collapse pressure = f (Radii)</u>

$\Delta p = 2\sigma \left(\frac{1}{R_1} + \frac{1}{R_2}\right)$

<u>Level difference of bubble base to liquid surface</u>

$r = 2\sigma / (g \cdot \rho_w \cdot h + C)$

σ = surface tension   g = gravitational acceleration
R = principle radii   C = emp. constant
h = hydrostatic head ρ = density

<u>Characteristic of the rising jet (see scheme)</u>

<u>Profil of the jet</u> (according to Newitt /9/)

$-y\,dy = \frac{1}{\frac{g \cdot \rho_w}{\sigma} - \frac{1}{K}}$ $K = x \cdot y$ = hyperbolic curve

<u>Height of jet befor break-up</u>

$h = \sqrt{\frac{2}{\frac{g \cdot \rho_w}{\sigma} - \frac{1}{K}}}$

<u>Velocity of the rising jet</u> (acc. to Davis /10/)

$u = \frac{P_m \cdot t}{2M} \; \frac{3}{2} \; \frac{\sigma \cdot t}{\rho_w \cdot r}$ with

$P = 2\sigma\pi r$ (vertical force) $P_m = \frac{P}{2}$

$M = \frac{2}{3}\pi r^3 \rho_w$ (moved liquid mass to fill up the crater)

<u>Jet break-up</u>

<u>Unbroken lenght l = f (initial velocity)</u>

$l = t \cdot u_0$

<u>Break-up criterion (acc. to Rayleigh /11/, Weber /12/)</u>

$l > \pi D_j$

<u>Relation ship between drop diameter D_d and jet diameter D_j</u>

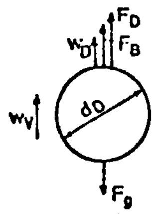
$D_j = 0.53\, D_d$

<u>Break-up time t (acc. to Castleman /13/)</u>

$t = \frac{2.3}{K^*} \log \frac{\alpha}{\alpha_0}$

K = emp. constant
α = amplitude of surface disturbance
α_0 = initial surface disturbance

<u>Fig. 2:</u> Vapour separation and formation of a liquid droplet carried over at a free liquid surface
- system of equations -



F_D = drag force
F_B = buoyancy force
F_g = gravity force
d_D = drop diameter (sphere)
w_D = droplet velocity
w_V = vapour velocity

M_D = mass of droplet
ρ = density
C_D = drag coefficient ($500 < Re < 5000$ assumed to be constant)
t = time
g = gravitational acceleration
h_D = height of rise of a droplet

Subscript: V = vapour
D = droplet

Relationship of the vertical forces acting on a droplet carried in the vapour steam

$$\Sigma F_{vertical} = M_D \frac{dw_D}{dt} = F_D - F_g + F_B \qquad (1)$$

$$M_D \frac{dw_D}{dt} = \left[\frac{1}{2}\rho_V (w_V - w_D)^2 \frac{\pi d_D^2}{4} \cdot C_D\right] - \left[\rho_D \frac{\pi}{6} d_D^3 \cdot g\right] + \left[\rho_V \frac{\pi}{6} d_D^3 \cdot g\right] \qquad (2)$$

Boundary condition: $M_D \frac{dw_D}{dt} = 0$ gives the terminal rise velocity

$$w_{D,terminal} = w_V - \left[\frac{4}{3}\frac{d_D}{C_D} \cdot g \frac{\rho_D - \rho_V}{\rho_V}\right]^{1/2} \qquad (3)$$

Carry over condition:

$$w_V > \left[\frac{4}{3}\frac{d_D}{C_D} \cdot g \frac{\rho_D - \rho_V}{\rho_V}\right]^{1/2} \qquad (4)$$

Integration of equ. 2 with the boundary conditions $t = 0$ $w_D = w_V$
$t = \infty$ $w_D = w_{D,terminal}$

gives:

$$w_D(t) = w_V - \left(\frac{4}{3}\frac{d_D}{C_D}\frac{\rho_D - \rho_V}{\rho_D} \cdot g\right)^{1/2} \tanh\left[\left(\frac{3}{4}\frac{\rho_V}{\rho_D}\left(1 - \frac{\rho_V}{\rho_D}\right)\frac{C_D}{d_D} \cdot g\right)^{1/2} \cdot t\right] \qquad (5)$$

supposing $\frac{dh_D}{dt} = w_D$

Integration of equ. 5 gives:

$$h(t) = w_D \cdot t - \frac{4}{3}\frac{\rho_D}{\rho_V}\frac{d_D}{C_D} \ln\cosh\left[\left(\frac{3}{4}\frac{\rho_V}{\rho_D}\left(1 - \frac{\rho_V}{\rho_D}\right) \cdot \frac{C_D}{d} \cdot g\right)^{1/2} \cdot t\right] \qquad (6)$$

Fig. 3: Droplet separation from a vapour stream /14/.
- System of equations -

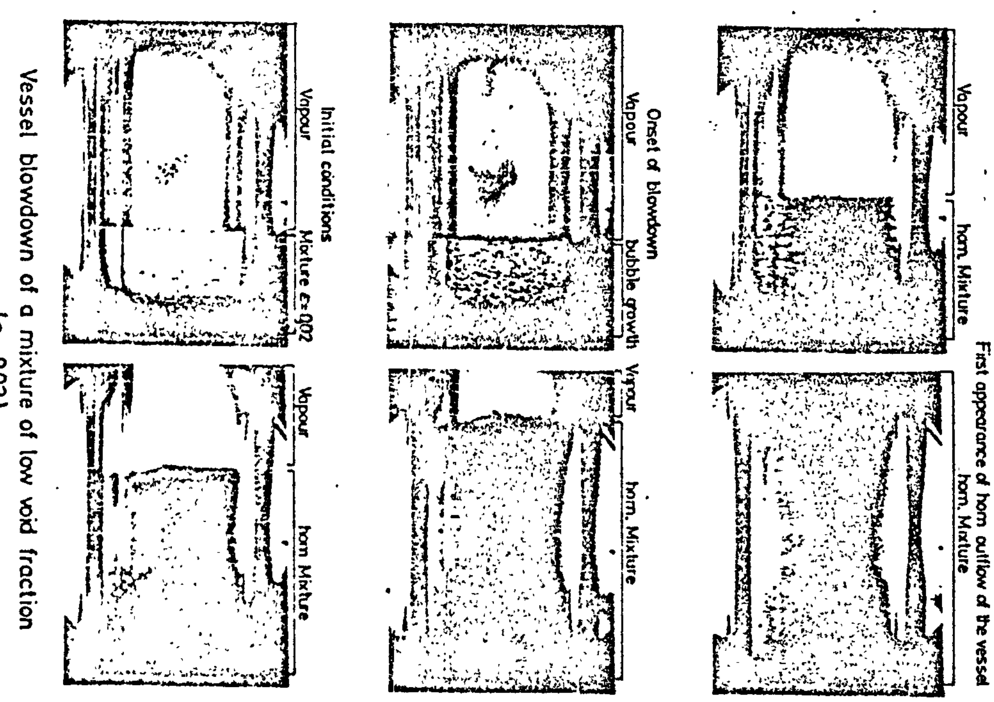
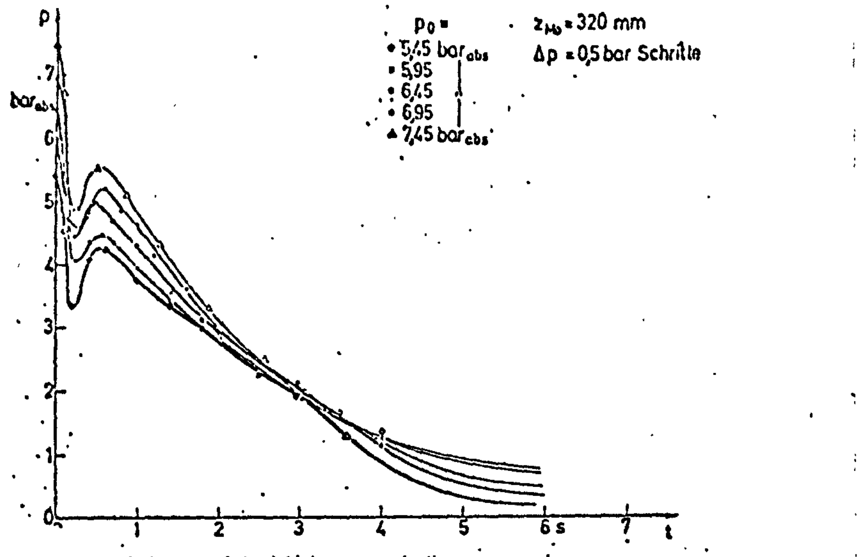
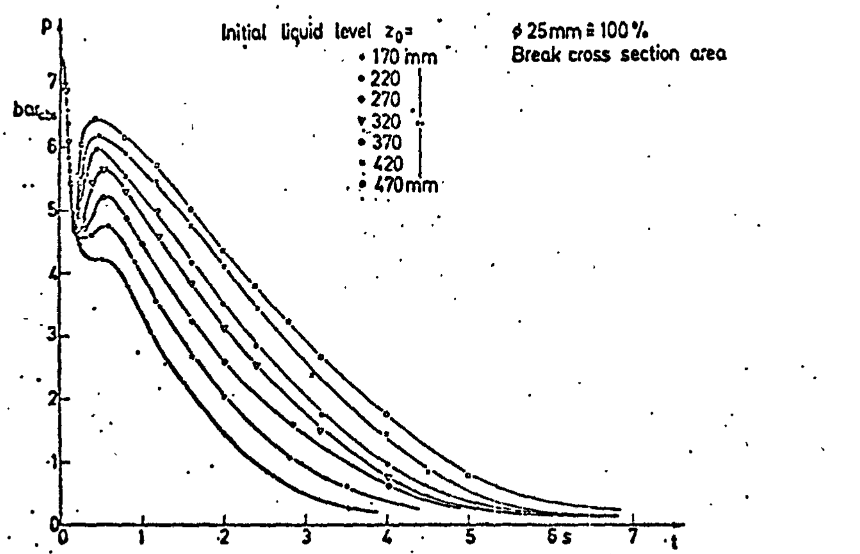


Fig. 4: Flashing behaviour of a liquid pool

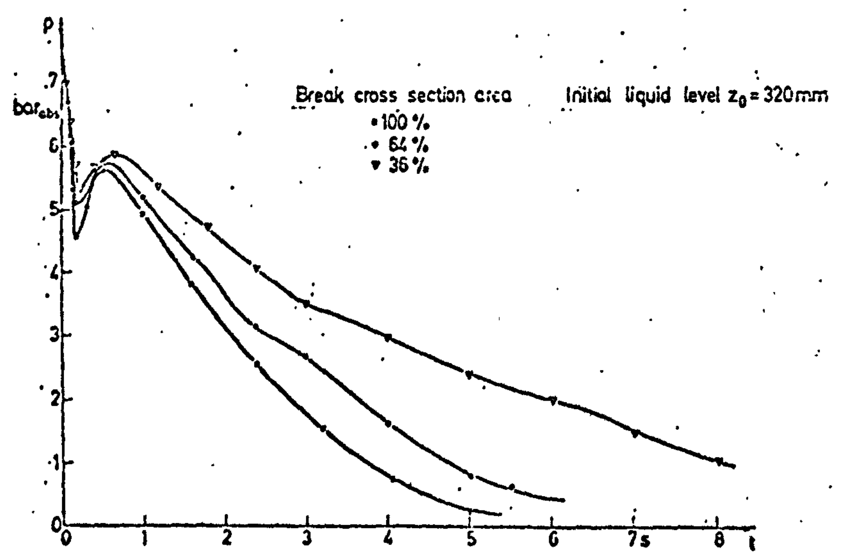
1.1 . - 31.12.1977



Influence of the initial pressure to the pressure decrease



Pressure decrease = f (z_0)



Pressure decrease = f (break area) z_0 = 320 mm

Fig. 5: Transient system pressure behaviour during a vessel blowdown
- parameter variation -

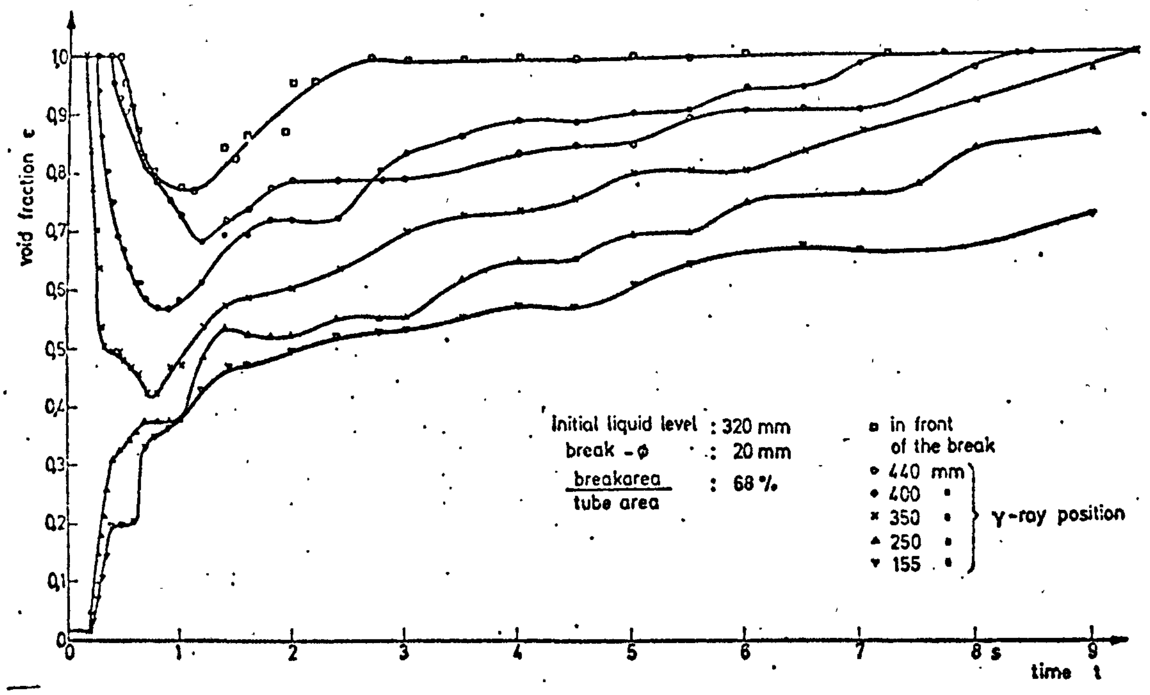


Fig. 6: Transient void-fraction distribution at different heights of the vessel and in front of the break

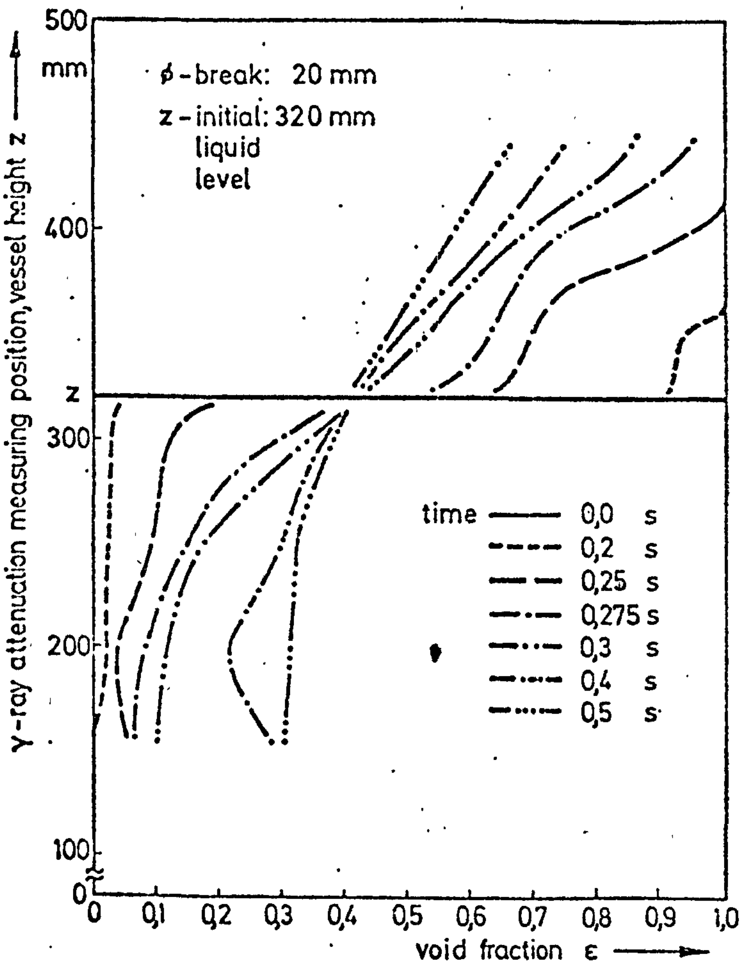


Fig. 7: Time dependent void fraction distribution versus vessel height

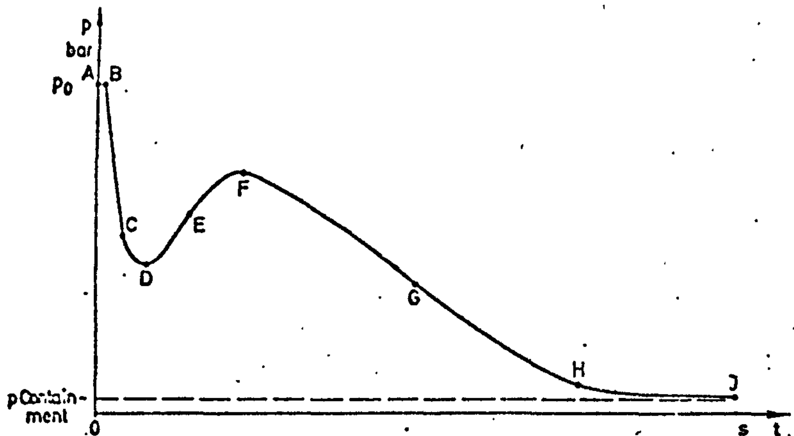


Fig. Qualitative pressure decrease of a BWR during blowdown initial liquid level below the outlet pipe

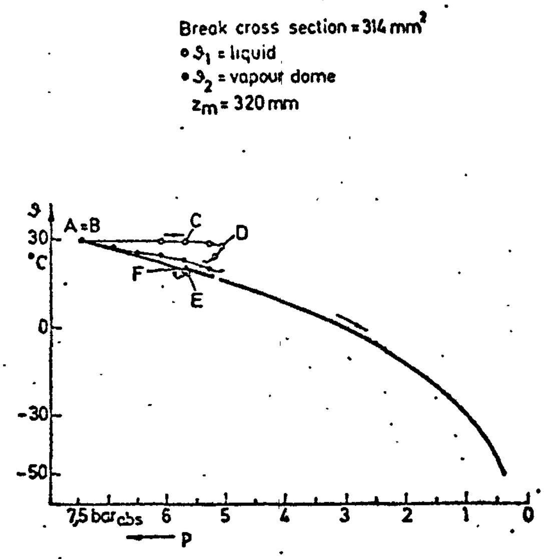


Fig. Non equilibrium of temperature and pressure behaviour during onset of the blowdown

$\boxed{A}$: initial conditions

$\boxed{B-C:}$ evaporation delay in the mixture onset of vapour decompression

$\boxed{\text{At point } C:}$ onset of bubble growth in the mixture due to flash-evaporation

$\boxed{C-D:}$ Flashing mixture moves upwards

$\boxed{\text{At point } D:}$ Flashing boundary reaches the outlet pipe at the top of the vessel. Onset of two-phase outflow.

$\boxed{D-F:}$ Due to the non-equilibrium effects the flash-evaporated volume prevails the outstreaming volume

$\boxed{\text{At point } F:}$ Flash evaporated volume = outstreaming volume

$\boxed{F-H:}$ Continious pressure decrease due to homoge-nous outflow

$\boxed{\text{At point } H:}$ Mixture level drops below the altitude of the outlet pipe

$\boxed{J\ :}$ Pressure equilibrium is reached

Fig. 8: Principle description of the blowdown process

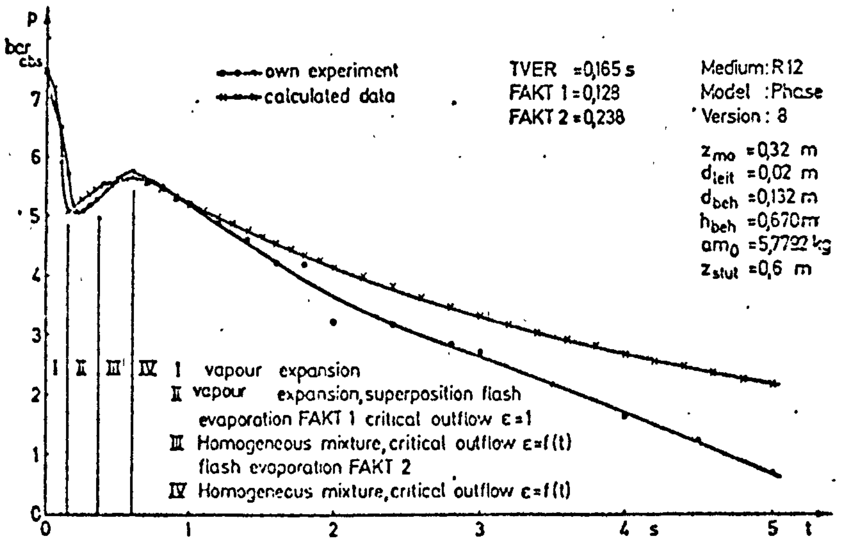


Fig. Pressure course of a vessel blowdown without core structure Comparison: own experiments R12 calculated data JONAS Version 8

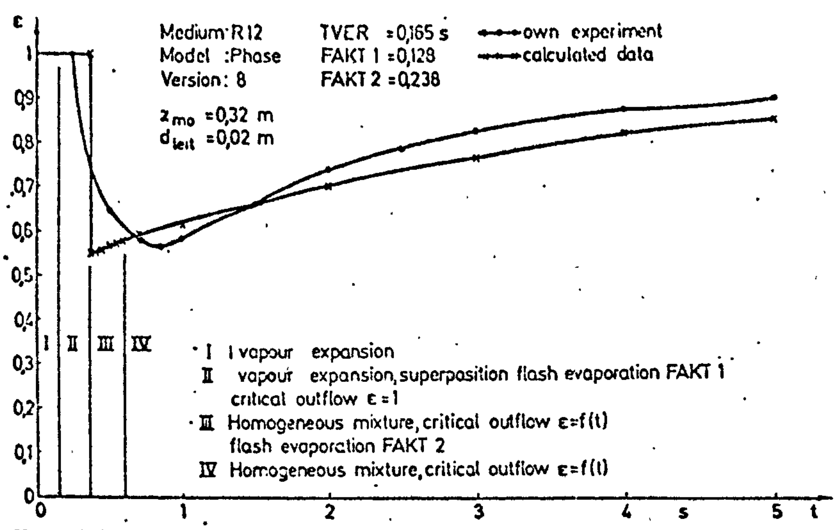


Fig. Void fraction before hot leg junction Comparison: own experiments R12 calculated data JONAS Version 8

Fig. 9: Comparison of own R 12 experiments with calculated data.

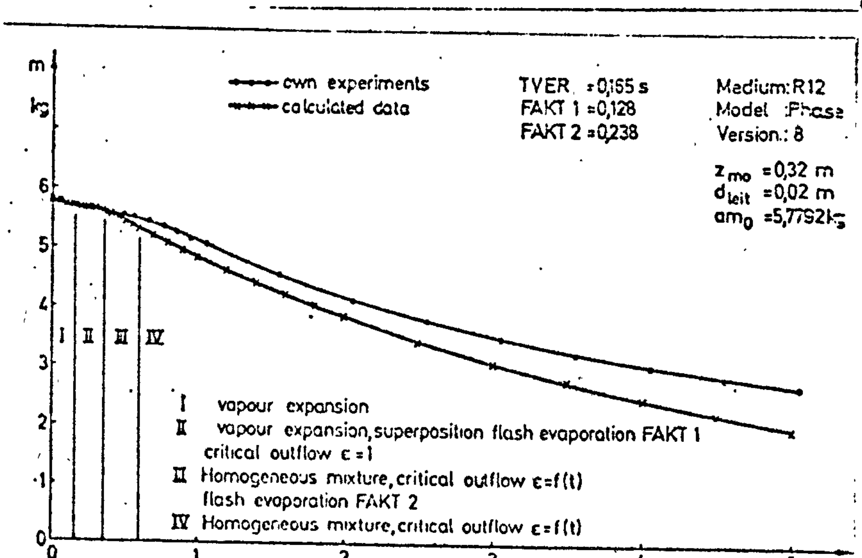


Fig. Mass contents in the vessel Comparison: own experiments R12 calculated data JONAS Version 8

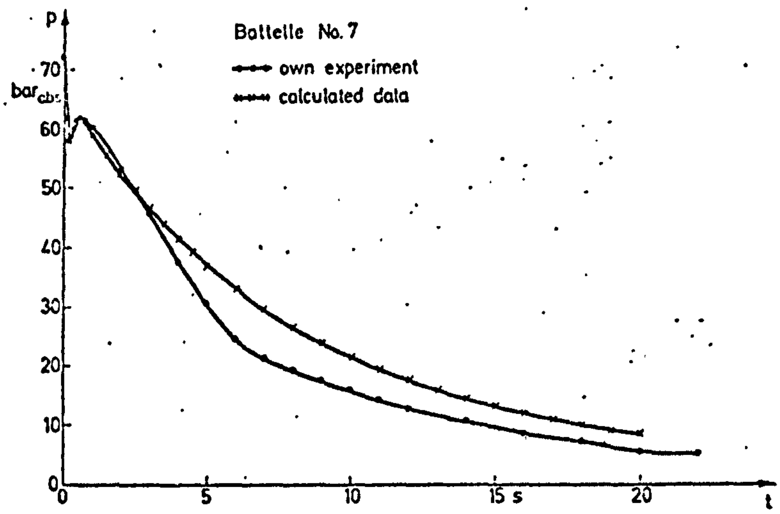


Fig. Pressure course of a vessel blowdown without core structure Comparison : experimental Battelle Frankfurt data medium H_2O calculated data JONAS Version 8

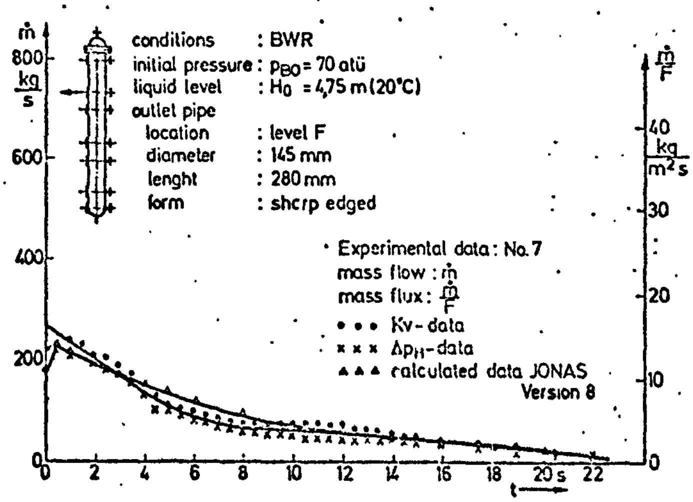


Fig. Mass contents in the vessel Comparison: experimental Battelle Frankfurt data medium H_2O calculated data JONAS Version 8

Fig. 10: Recalculation of H_2O data from the Battelle Institut Frankfurt with the own computer code

'145-1 -03/4160-04		1.1.2
Titre Thermohydraulique du LOCA. Etude des débits critiques en double phase : Programmes MOBY-DICK et SUPER MOBY-DICK.		**Pays** FRANCE **Organisme directeur** CEA/DgCS - EDF/SEPTEN
Titre (anglais) LOCA thermohydraulics. Critical two phase flow studies : MOBY-DICK and SUPER MOBY-DICK projects.		**Organisme exécuteur** CEA/DTCE/STT (GRENOBLE) **Responsable** STT - Grenoble
Date de démarrage 01/01/72	Etat actuel en cours	**Scientifiques**
Date prévue d'achèvement 31/12/80	Dernière mise à jour 1/78	

1 - Objectif général :

Développer et qualifier des modèles d'écoulement en double phase à partir d'expériences analytiques où les déséquilibres entre phases sont importants.

2 - Objectifs particuliers :

Etudier la cinétique de vaporisation en écoulement double phase à fort gradient de pression. Réaliser des débits critiques dans des conditions et des géométries variables.

3 - Installations expérimentales et programme :

MOBY-DICK : Boucle dans laquelle est réalisée un mélange double phase par autovaporisation. Plusieurs géométries de section d'essai sont prévues :

.../

1) Tube de section de 20 mm ID, terminé par un divergent de 7 degrés.
2) Tube de section de 14 mm ID, terminé par un divergent de 7 degrés.
3) Tube de section de 14 mm ID, terminé par un divergent de 7 degrés ; essai en eau-azote.

Ces essais sont réalisés à basse pression et à faible titre ($P < 7$ bars, $X < 2\%$). La boucle a été modifiée afin d'atteindre des titres plus élevés.

SUPER MOBY-DICK : Même type d'expérience mais à des niveaux de pression pouvant aller jusqu'à 100 bars.

4 - Etat de l'étude :

1) Avancement à ce jour

MOBY-DICK : essais 1,2 et 3 terminés. Dans la série 2 des difficultés ont été rencontrées dans la reproductibilité des essais.

Essai en eau-air avec un divergent de 7° terminé.
Essai en eau-vapeur avec un élargissement brusque de la section d'essai terminé.
SUPER MOBY-DICK : Boucle en construction.

5 - Prochaines étapes :

MOBY DICK : Rédaction du rapport final.
SUPER MOBY-DICK : début des essais en Septembre 1978.

6 - Relation avec d'autres études :

C.F.T. : Expérience suédoise de débit critique pour des sections de brèches importantes (200 à 500 mm).
REBECA : Etude de débits critiques d'un mélange à 3 composants : eau, air, vapeur.

7 - Documents de référence disponibles :

" Contribution à l'étude de débits critiques en écoulement diphasique eau-vapeur ", M.REOCREUX - Thèse de l'Université de GRENOBLE, 1974.

" Etudes expérimentales de débits critiques en écoulement diphasique eau-vapeur ", M.GUIZOUARN - Note DTCE-STT 501, décembre 1975.

145-1-04/4160-01		1.1.2
Titre Thermohydraulique du LOCA. Etude de la phase dépressurisation d'un réacteur à eau pressurisée : Programme OMEGA.		Pays FRANCE Organisme directeur CEA/DgCS - EDF/SEPTEN
Titre (anglais) LOCA thermohydraulics. P.W.R. Blowdown studies : OMEGA Projects.		Organisme exécuteur CEA/DTCE-STT (GRENOBLE) Responsable M. COURTAUD STT-Grenoble
Date de démarrage 01/01/72	Etat actuel en cours	Scientifiques R. RICQUE J.C. ROUSSEAU
Date prévue d'achèvement 31/12/78	Dernière mise à jour 1/78	

1 - Objectif général :

Etude des transferts de chaleur durant la phase de dépressurisation d'un réacteur à eau pressurisée afin d'établir des corrélations d'échanges de chaleur et de flux critiques.

2 - Objectifs particuliers :

Développer des modèles physiques pour l'interprétation des expériences : étude des corrélations de RELAP 4, ainsi que des modèles d'écoulement, validation de modèles physiques pour les codes 2ème génération.

3 - Installations expérimentales et programme :

- Boucle OMEGA : Pression 170 bars, débit max. 20 kg/s , puissance 4,5 MW.
- Dépressurisation d'une section d'essais tubulaire puis d'une grappe 36 barreaux :
  1) Section tubulaire de 12 mm de diamètre et 3,65 m de long avec un flux axial uniforme ; taille de brèche : 50 et 15 mm2 : brèche amont, aval, et aux deux extrémités.

.../

1) Tube de section de 20 mm ID, terminé par un divergent de 7 degrés.
2) Tube de section de 14 mm ID, terminé par un divergent de 7 degrés.
3) Tube de section de 14 mm ID, terminé par un divergent de 7 degrés ; essai en eau-azote.

Ces essais sont réalisés à basse pression et à faible titre (P < 7 bars, X < 2%). La boucle a été modifiée afin d'atteindre des titres plus élevés.

SUPER MOBY-DICK : Même type d'expérience mais à des niveaux de pression pouvant aller jusqu'à 100 bars.

4 - Etat de l'étude :

1) Avancement à ce jour

MOBY-DICK : essais 1,2 et 3 terminés. Dans la série 2 des difficultés ont été rencontrées dans la reproductibilité des essais.

Essai en eau-air avec un divergent de 7° terminé.
Essai en eau-vapeur avec un élargissement brusque de la section d'essai terminé.
SUPER MOBY-DICK : Boucle en construction.

5 - Prochaines étapes :

MOBY DICK : Rédaction du rapport final.
SUPER MOBY-DICK : début des essais en Septembre 1978.

6 - Relation avec d'autres études :

C.F.T. : Expérience suédoise de débit critique pour des sections de brèches importantes (200 à 500 mm).
REBECA : Etude de débits critiques d'un mélange à 3 composants : eau, air, vapeur.

7 - Documents de référence : disponibles:

" Contribution à l'étude de débits critiques en écoulement diphasique eau-vapeur ", M.REOCREUX - Thèse de l'Université de GRENOBLE, 1974.

" Etudes expérimentales de débits critiques en écoulement diphasique eau-vapeur ", M.GUIZOUARN - Note DTCE-STT 501, décembre 1975.

145-1 - 10		1.1.2
Titre MARVIKEN CFT (Etude de débits critiques)		Pays SUEDE Organisme directeur PAYS NORDIQUES USA (N.R.C. + EPRI) FRANCE (CEA+ EDF) ~~PAYS BAS (KEMA)~~
Titre (anglais) MARVIKEN CRITICAL FLOW TESTS		Organisme exécuteur MARVIKEN - SUEDE Responsables français J. PELCE (CEA/DSN) J. AZAM (EDF/SEPTEN)
Date de démarrage 1/01/77	Etat actuel en cours	Scientifiques français M. REOCREUX (CEA/DSN) G. HOUDAYER (EDF/SEPTEN)
Date prévue d'achèvement 1/07/79	Dernière mise à jour 1/78	

1 - Objectif général :

1) Etude des débits critiques dans le cas des réacteurs à eau ordinaire.

- Objectifs particuliers :

2) La particularité des essais MARVIKEN réside dans l'étude des ruptures de grosses sections, ce qui n'est jamais le cas dans les installations expérimentales. Les sections de fuite étudiées seront : 200, 300 et 500 mm de diamètre.

3 - Installations expérimentales et programme :

On utilise pour l'expérience, le réacteur désaffecté de MARVIKEN (Suède). La cuve du réacteur est remplie sous pression. La brèche est située au bas de la cuve. Les paramètres étudiés sont :

.../

- pression à la brèche (30,40 et 50 bars)
- diamètre de la brèche (20,30 et 50 cm)
- sous-saturation de l'eau (5,15 et 30°C)
- rapport L/D (L longueur de la section d'essai, D diamètre de la brèche (0, 1, 2,5 et 3)

Le programme a été défini lors d'une réunion d'experts qui d'est tenue à MARVIKEN en avril 1976. La durée des essais est de 18 mois.

L'accord concernant les essais a été signé par :

Les Pays Nordiques (Suède, Danemark, Norvège, Finlande)

Les USA (NRC + EPRI)

La France (CEA + EDF)

Les Pays Bas (KEMA)

La participation française est de 20% du coût total répartis également entre CEA et EDF.

4 - Etat de l'étude :

La phase de modification de l'installation et de mise au point est terminée et la période des essais proprement dits a débuté (durée prévue : jusqu'au 1/7/79)

6 - Relation avec d'autres études :

Essais de débits critiques réalisés en FRANCE : MOBY DICK, SUPER MOBY DICK.
Méthodes analytiques d'interprétation.
N.B. : l'interprétation des essais ne fait pas partie de l'accord.

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 Status Last updating : 20.01.75	Scientists M. BLANC-[illegible] M. SUREAU

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 PROGRAMS

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

a) Essais sur maquettes "froides", Plateforme d'essais Pompes Turbines de CHATOU. La maquette de type 93 D d'échelle géométrique 1/3,2 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 JHPV 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.

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La maquette, d'échelle géométrique 1/10, 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/MTN
Initiated avril 1974	Completed octobre 1975	Scientists M. DAUBERT M. SUREAU
Status 1ère phase en cours	Last updating: 20.01.75	

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 instants (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.

...

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
Initiated Mai 1974 Completed mai 1975 Status Etude en cours Last updating : 20.01.75	Scientists M. PUGNET M. LARRIEAU

I - GENERAL AIM

Déterminer ce que la différence de température des débits primaire 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.

The pressure gradients of all blowdowns were controlled according to precalculated curves. The design of the facility allowed to do this very accurately.

Reproducibility of the dates was studied for blowdowns under different conditions.

For the comparison of computer codes, a standard DNB test was carried out. The data of this tests were send to the IRS for further use.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

RS 36 B	Emergency Core Cooling Program Refilling Experiments with Simulation of the Circulation Loop
RS 36 C	Emergency Core Cooling Program Low Pressure Experiments, BWR Second Cluster

9. References

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10. Degree of Availability

TITLE 1 (original language) Investigation of the transient flow response in a BWR core	Classification 1.1.2
TITLE 2 (english)	Country: ITALY Sponsor: CNEN Organisation: CNEN-CISE
Date initiated 1974 Date completed Dec. 1975 Last updating	Project Leader V. Marinelli (CNEN) G.P. Gaspari (CISE)

General aim

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

Particular objectives

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

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, taken on a 16 rod BWR test section 12 ft long: the plant used is IETI-III (CISE).

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 done.

4.2 Essential results

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

Degree of availability

To a limited extent.

TITLE 1 (original language) Investigation of flow blockage effects in a subchannel array	Classification 1.1.2
TITLE 2 (english)	Country: ITALY Sponsor: Organisation: CNEN-A.B.Atomenergy(Sweden)
Date initiated June 1973 Date completed May 1976 Last updating June 1976	Project Leader V. Marinelli (CNEN) B. Kjollen (A.B.Atomenergy)

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.

5. Degree of availability. To a limited extent.

Title 1 (Original language) Transitori termoidraulici in reattori a tubi in pressione durante lo svuotamento .	Classification 1.1.2
Title 2 (English) Thermohydraulic transients in pressure tube reactors during blowdown	Country ITALY Sponsor: CNEN (mainly) and AECL (Canada) Organisation CISE
Date initiated 1968 Date completed 1978 Last updating April 1977	Project Leader UTMF (CISE)

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 basic 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 circuits; 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.

Title 1 (Original language)	Classification
Transitori termoidraulici in reattori a tubi in pressione durante lo svuotamento	1.1.2

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; check of heat transfer correlations in post-dryout conditions;
- (3.2.3.): several scaled down blowdown tests simulating inlet and outlet failure have been carried out;
- (3.2.4.): integrated tests simulating downcomer, inlet header and steam line failure have been completed.
- (3.2.5.): 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.
- Availability 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.
- Availability 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.
- 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 1971.

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 publication 74 WA/HT-43 presented at the Winter Annual Meeting, New York, November 17-22, 1974.

4) R. Baldassarre, G.P. Gaspari, R. Granzini, V. Pagliari "Predictions of

Title 1 (Original language)	Classification
Transitori termoidraulici in reattori a tubi in pressione durante lo svuotamento	1.1.2

transients CHF using TILT code and the steady state CISE-3 CHF correlation" CISE R-364 (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 (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) V. Agostini, A. Premoli "Valvola di intercettazione rapida per impiego in acqua-vapore" Energia Nucleare vol. 10, 1, January 1971.

8) G. Pierini, C. Sandri "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.

10) A. Azzalin et al. " Blowdown tests on the CIRCE loop under conditions concerning DBA (Design Basic Accident) of the CIRENE reactor" CISE R 362, June 1975.

11) A. Azzalin et al. " Scaled-down blowdown tests concerning voiding rates and thermal transients for inlet end breaks and subcooled initial conditions" CISE R 370, Dec. 1975.

12) A. Azzalin et al. "Scaled down blowdown tests concerning voiding rates and thermal transients for inlet end breaks and boiling initial conditions" CISE R-380, Feb. 1976.

7. Degree of availability: to a limited extent

Title 1 (Original language)	Classification
DNB in loss of flow	I.I.2
Title 2 (English)	Country: ITALY Sponsor: CNEN Organisation: CNEN
Date initiated: April 1976 Date completed: June 1977 Last updating: March 1977	Project Leader G.E. Farello

I - General aim: Experimental determination of faulting conditions due to loss of flow.

2 - Particular objectives: The influence of loss of flow due to pump failure in a uniformly heated channel has been studied.

3 - Experimental facility: Freon I2 loop (400 l/h).

4 - Project status: A final report has been completed and will be available within June 1977.

Title 1 (Original language)	Classification
DNB in damaged bundles (obstructions and bowings)	I.I.2
Title 2 (English)	Country ITALY Sponsor CNEN Organisation CNEN
Date initiated April I976 Date completed December I977 Last updating March I977	Project Leader G.E. Farello

I - General aim	Experimental study for determination of effects of fuel bundle distortions in faulty conditions.
2 - Particular objectives	The influence of single subchannel obstructions and rod bowings in a 4 rod (freon cooled) bundle will be tested.
3 - Experimental facility	IOO kW, IO tons/h freon loop.
4 - Project status	Test section is ready; final checks of the loop are completed.
5 - Next steps	DNB power measurements and comparison of data with existing codes.

TITLE 1 (original language)	Classification
Instabilità connesse con il rilascio del vapore attraverso le valvole di sicurezza	1.1.2 - 7.2
TITLE 2 (english)	Country: ITALY
Instability phenomena related to steam relief through S.R.V.	Sponsor: CNEN Organisation: CNEN
Date initiated 3-1976 Date completed 6-1978 Last updating June 1976	Project Leader D. Pitimada

Description:

1. General aim

   Experimental study of air, water and steam discharge through a single safety relief valve.

2. Particular objectives

   Determination of instabilities connected to air-water clearing, bubble dynamics and to steam flow pulsations.
   Implementation of a computer code for the determination of chief parameters interesting the discharge.

3. Experimental facilities and programme

   Facility consisting of: 2 m^3 boiler (70 kg/cm^2), 2" relief valve, 70 m long, 1.5" SS. discharge pipe, 7 m^3 suppression pool.

4. Project status

   The facility is in advanced building status.
   The computer code is implemented as far as the water clearing phenomenon is concerned.

5. Next steps

   Experimental determination of pressure, temperature and flow rate as functions of steam and water conditions.
   Implementation of the bubble dynamics model.
   Comparison of experimental data with computer codes.

Title 1 (Original language) STUDIO DELL'EFFLUSSO CRITICO BIFASE IN CONNESSIONE CON IL LOCA NEI REATTORI AD ACQUA LEGGERA	Classification 1.1.2
Title 2 (English) STUDIES ON TWO-PHASE CRITICAL FLOW IN CONNECTION WITH LOCA IN LIGHT WATER REACTORS	Country ITALY Sponsor Organisation (Universita' di PALERMO ++
Date initiated 1976 Date completed 1978 Last updating May 1977	Project Leader E. OLIVERI

++ ISTITUTO DI APPLICAZIONI E IMPIANTI NUCLEARI

DESCRIPTION:

The program has been set up with the aim of developing- from basic principles avoiding the use of correlations that are restricted to particular test conditions- a theoretical model for the prediction of steam/water critical pressure and critical flow rate in terms of upstream stagnation properties.

REFERENCE DOCUMENTS:

- F. CASTIGLIA-E. OLIVERI-G. VELLA
  Sulla determinazione della portata nell'efflusso critico bifase. ACCADEMIA DI SCIENZE LETTERE E ARTI DI PALERMO - 24 Giugno 1976 -

Title 1 (Original language)	Classification
PROGRAMMA P.I.P.E.R.: esperienze di blow-down in presenza di strutture interne.	$\boxed{1.1.1,}$ 1.1.2
Title 2 (English)	**Country** ITALY
Blow-down Tests by Piper apparatus-experiments with internal structures.	Sponsor CNEN-CNR Organisation University of Pisa
Date initiated 1972 Date completed 1978 Last updating 1977	Project Leader P. VIGNI

Title 1 (Original language)	Classification
Analisi dei transitori termici ed idraulici a seguito di LOCA nei reattori ad acqua leggera.	$\boxed{1.1.1}$ 1.1.2, 1.1.4, 1.2
Title 2 (English)	**Country** ITALY
Analysis of thermal and hydraulic transients following a LOCA in Light Water Reactors	Sponsor CNEN and CNR Organisation University of Pisa
Date initiated 1974 Date completed 1978 Last updating may 1977	Project Leader N. CERULLO

	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.

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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.

0) Personnel : 3 men/year.

1) Additional Information

			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 Status : progressing		Completed : 31.12.1975 Last updating : March 1975	

	Classification $\boxed{10.3}$ (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

ENERGIEONDERZOEK CENTRUM NEDERLAND		CLASSIFICATION : 1.1.2
TITLE : CHARME: Een computerprogramma ter bestudering van uitstroming		COUNTRY: THE NETHERLANDS
		SPONSOR : ECN ORGANIZATION : ECN
TITLE (ENGLISH LANGUAGE): CHARME: A computer program to study blowdown		PROJECTLEADER : Speelman, J.E.
INITIATED : April 1976	LAST UPDATING : May 1978	SCIENTISTS : Putten, A.P.W.M. van der Bogaard, J.P.A. van den Koning, H.
STATUS : progressing	COMPLETED : 1980	

General aim

Development of a computer code to study the blowdown-process.

Particular objectives

CHARME solves a set of partial differential equations describing the conservation of mass, energy and momentum in a tube as a finction of axial coordinate and time, using the method of characteristics. Notably the pressure, temperature, velocity and void fraction are calculated as function of time and of axial coordinate including the transition to supersonic velocities. The model also includes pressure losses due to friction. A special subroutine has been developed to calculate the fluid parameters in the jet arising after the system opening. This jet model is necessary for the cases in which the blowdown process proceeds with subcritical velocities.

Experimental facilities and program: Not foreseen

Project status

A first model has been developed for the jet; CHARME calculations have been verified on basis of experimental results, performed by others, with encouraging agreement.
The code has been updated, especially the subroutine in which the fluid properties are calculated.

Next steps

Improvement of the jet subroutine. A sensitivety study will be performed with regard to slip and non-equilibrium effects between two phases.

Relation with other projects

Usable to check gross models in other thermo-hydraulic blowdown computer codes.

Reference documents

J.P.A. v.d. Bogaard, H. Koning, A.P.W.M. v.d. Putten: CHARME, A time and space dependent model to predict the discharge rate of single and two-phase fluids through pipes.
NEA-CSNI: Specialist's meeting on transient two-phase flow, Toronto, 3rd-4th August, 1976.

Degree of availability: Upon mutual agreement at ECN-Petten

Budget: -

Personnel:

ENERGIEONDERZOEK CENTRUM NEDERLAND (ECN)	CLASSIFICATION : 1.1.2
TITLE : Reflood experimenten	COUNTRY: THE NETHERLANDS
	SPONSOR : ECN
	ORGANIZATION : ECN
TITLE (ENGLISH LANGUAGE): Reflood experiments	PROJECTLEADER : S.B. van der Molen
INITIATED : 1977 — LAST UPDATING : May 1978	SCIENTISTS : S.B. van der Molen, H. Hoogland, F.W.B.M. Galjee
STATUS : started — COMPLETED : 1980	

General aim

Study of the reflood and rewetting phenomena in bundles.

Particular objectives

a) Investigation of the heat transfer from high temperature fuel pins in bundles by radiation to waterdroplets and vapour convection before rewetting.
b) Study of the influence of a temperature profile over a bundle cross-section on the velocity of the quench front and the cross flow between adjacent channels.
c) Study of the influence of instabilities in parallel bundles on the velocity of the rewetting-front.

Experimental facilities and program

a) Testloops for low and high pressure experiments
b) High speed film camera

Project status

Experiments to study the rewetting phenomena by high speed cinematography have been started in order to study the two phase flow downstream of the quench front. First calculations have been made to obtain an order of magnitude of the convective and radiative heattransfer to the dispersed two phase flow.

Next steps

Development of measuring techniques to determine void fraction and droplet concentration in the two phase flow, downstream of the quench front.

Relation to other projects: no

Reference documents: ECN-78-064, The entrained droplet and vapour velocity and the heat transfer in the dispersed flow region in case of bottom flooding.

Degree of availability: Through E.C.N. library channel

Budget: Hfl. 150.000,--/yr

Personnel: 3 manyears/yr.

Netherlands Energy Research Foundation (ECN)	CLASSIFICATION : $\boxed{1.1.1}$/1.1.2.
TITLE : Mechanisch gedrag van het reactorbinnenwerk tijdens grote ongelukssituaties	COUNTRY: THE NETHERLANDS
	SPONSOR : ECN
	ORGANIZATION : ECN
TITLE (ENGLISH LANGUAGE): Mechanical behaviour of reactor internals during major accident situations.	**PROJECTLEADER :** L.H. Vons
INITIATED : 1977 — LAST UPDATING : May 1977	**SCIENTISTS :** H. van Rij, L.G.J. Janssen
STATUS : in progress — COMPLETED : 1980	

	Classification 1.1.2
Title 1 DEPRESSURISATION DISCHARGE RATE	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION AWRE FOULNESS
Title 2	Project Leader A R EDWARDS
Initiated 1968 Completed : Status : Last updating 1976	Scientists:

Description:

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 wuitable 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 and 140 bar saturation pressure were used for 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 was terminated at the end of March 1975.

Data reduction of a representative sample of tests is in progress and reports should be available in 1977.

Reference Documents

Aldermaston Report AWRE/44/86/97 (SRD R29)

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)	Project Leader Prof. W. B. Hall
Initiated October 1975 Status progressing	Scientists 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.

/...

6. Relation with other projects

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

Ref. Similation 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. General aim

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. Particular objective

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

3. Experimental facilities and programme

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. Project Status

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

/...

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

Berichtszeitraum/Period 1.1.77 - 31.12.77	Klassifikation/Classification 1.1.4	Kennzeichen/Project Number PNS 4234
Vorhaben/Project Title Bestimmung der Nachzerfallswärme von ^{235}U im Zeitbereich 10 bis 1000 sec Decay Heat Measurement of ^{235}U in the Time Region from 10 to 1000 seconds		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KfK Projekt Nukleare Sicherheit (PNS) INR
Arbeitsbeginn/Initiated 1975	Arbeitsende/Completed 1978	Leiter des Vorhabens/Project Leader K. Baumung
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1977	Bewilligte Mittel/Funds

1. General Aim

LOCA-Analysis.

2. Particular Objectives

Providing decay heat data of ^{235}U for short cooling times up to 1000 seconds.

3. Research Program

Irradiation of pellet-like fuel samples and measurement of their adiabatic temperature rise and γ-energy output due to fission product activity.

4. Experimental Facilities, Computer Codes

A pneumatic transfer system was constructed which will be used to bring the fuel samples to the irradiation position in the thermal column of the FR2 reactor and to hold their temperatures near room temperature by aid of an integrated cooling loop. A calorimeter is connected to the transfer system and controlled by a mini-computer which also provides the data recording. The escaping γ-energy will be measured by a Moxon-Rae-type γ-energy flow detector. The control-code as well as several data processing codes, which will be performed by the mini-computer too, were written in Fortran-language.

5. Progress to Date

The experimental facility was installed at the reactor. A delay was caused by new instructions from the reactor operator concerning cooling water provision and use of tested halogene-filters in the exhaust of the transport system.

6. Results

First cold tests showed the proper operation of the facility.

7. Next steps

Before soon, the irradiation experiments will be performed.

Title 1 (Original language)	Classification
Burn-out in reverse flow	I.I.4
Title 2 (English)	Country ITALY Sponsor CNEN Organisation CNEN
Date initiated 1975 Date completed 1977 Last updating March 1977	Project Leader G.E. Farello

I - General aim

Experimental determination of critical heat flux for heated channels at very low flowrates in upward and downward flow.

2 - Particular objectives

DNB measurements in two uniformly heated tubes:

2.1) freon 12 — L = 200 cm, D = 4.75 mm

2.2) water — L = 400 cm, D = 11.5 mm

3 - Experimental facilities

A S.S.water loop (pressure up to 160 kg/cm^2) and a freon loop (pressure up to 40 kg/cm^2).

4 - Project status

Part of the research has been completed. A final report "Burn out in up-flow and down-flow" is available.

Title 1 (Original language) Analisi dei transitori termici ed idraulici a seguito di LOCA nei reattori ad acqua leggera.	Classification $\boxed{1.1.1}$, 1.1.2, 1.1.4, 1.2
Title 2 (English) Analysis of thermal and hydraulic transients following a LOCA in Light Water Reactors	Country ITALY Sponsor CNEN and CNR Organisation University of Pisa
Date initiated 1974 Date completed 1978 Last updating may 1977	Project Leader N. CERULLO

CLASSIFICATION 1.2

TITLE 1 FLECHT. Low flooding rate test program (Full length Emergency Cooling Heat Transfer)	COUNTRY Belgium (U.S.A.) SPONSOR ORGANIZATION Westinghouse Nuclear Europe
TITLE 2	PROJECT LEADER SCIENTISTS
INITIATED (date) May 1974 COMPLETED End 1976 STATUS PROGRESSING LAST UPDATING May 9, 1975.	

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.

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

266

7. REFERENCE DOCUMENTS

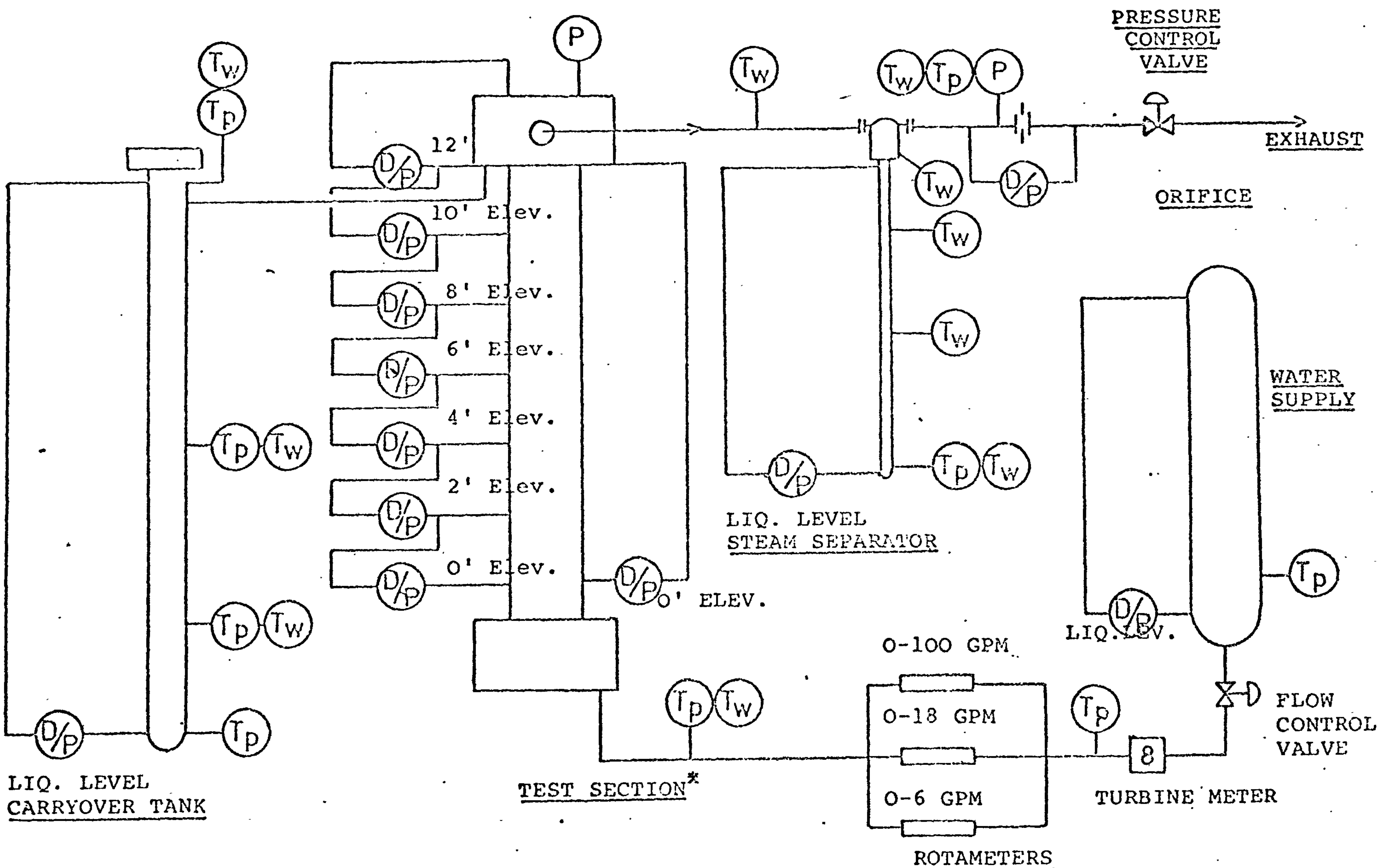
a) WCAP-7665 - PWR FLECHT Final Report, April, 1971.

b) WCAP-7931 - PWR FLECHT Final Report Supplement, October 1972.

8. DEGREE OF AVAILABILITY

Available upon request.

FLECHT [illegible] FLOODING RATE TEST CONFIGU[illegible]ION



* ALL INSTRUMENTATION IS NOT SHOWN

<table>
<tr><td colspan="2">Classification 1.2</td></tr>
<tr><td><u>Title 1</u>
FLECHT SET
Full Length Emercency Cooling Heat Transfer Systems Effect Tests.</td><td>COUNTRY Belgium (USA)
SPONSOR
ORGANIZATION : Westinghouse Nuclear Europe</td></tr>
<tr><td><u>Title 2</u></td><td rowspan="2">PROJECT LEADER
<u>SCIENTISTS</u> :</td></tr>
<tr><td><u>Initiated</u> (date) 7/30/74   <u>Completed</u> :
<u>Status</u> :   <u>Last updating</u></td></tr>
</table>

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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. <u>Essential Results</u>

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.

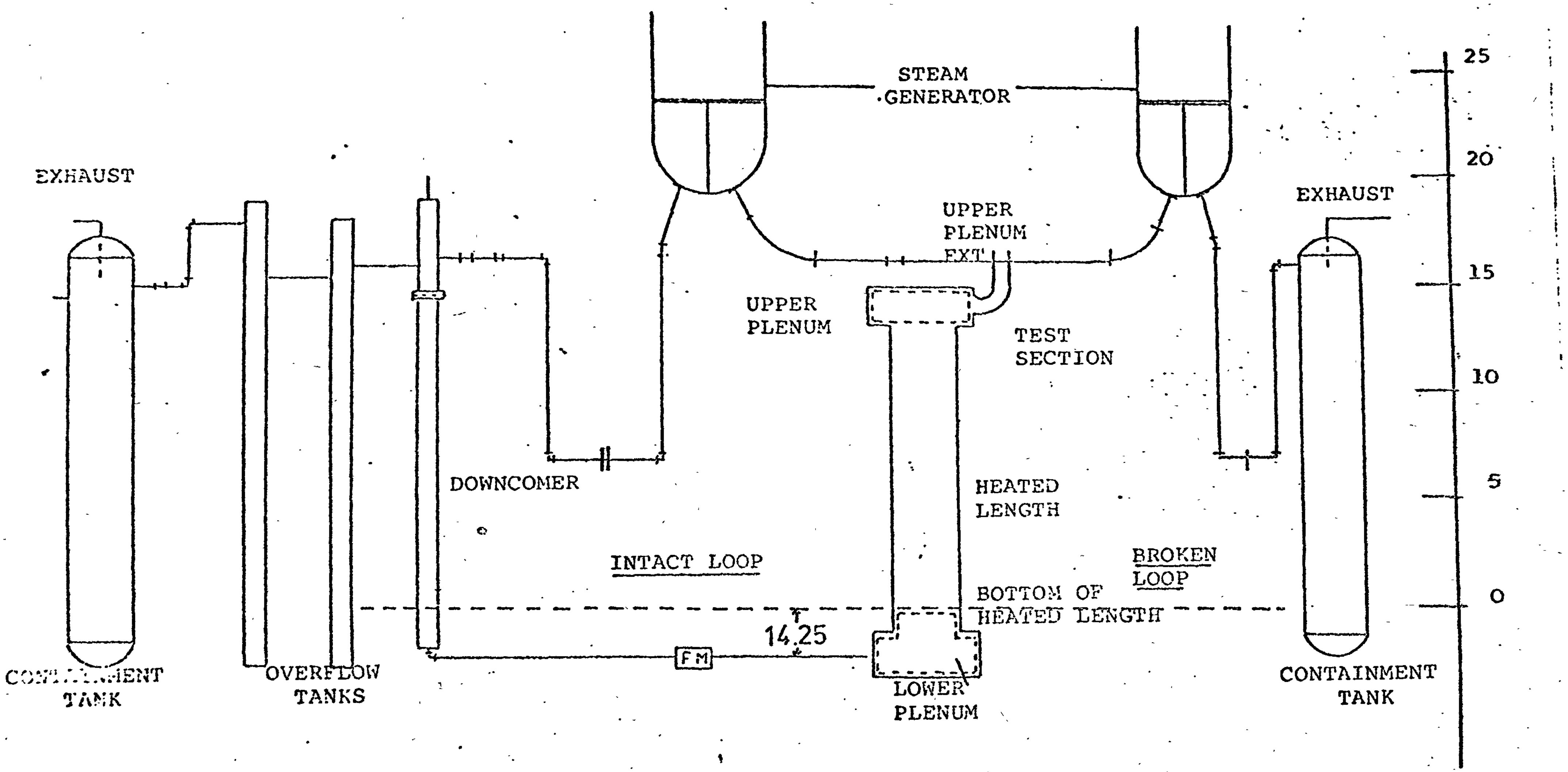

FIGURE 1

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
Initiated Completed Status Last Updating	SCIENTISTS

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°

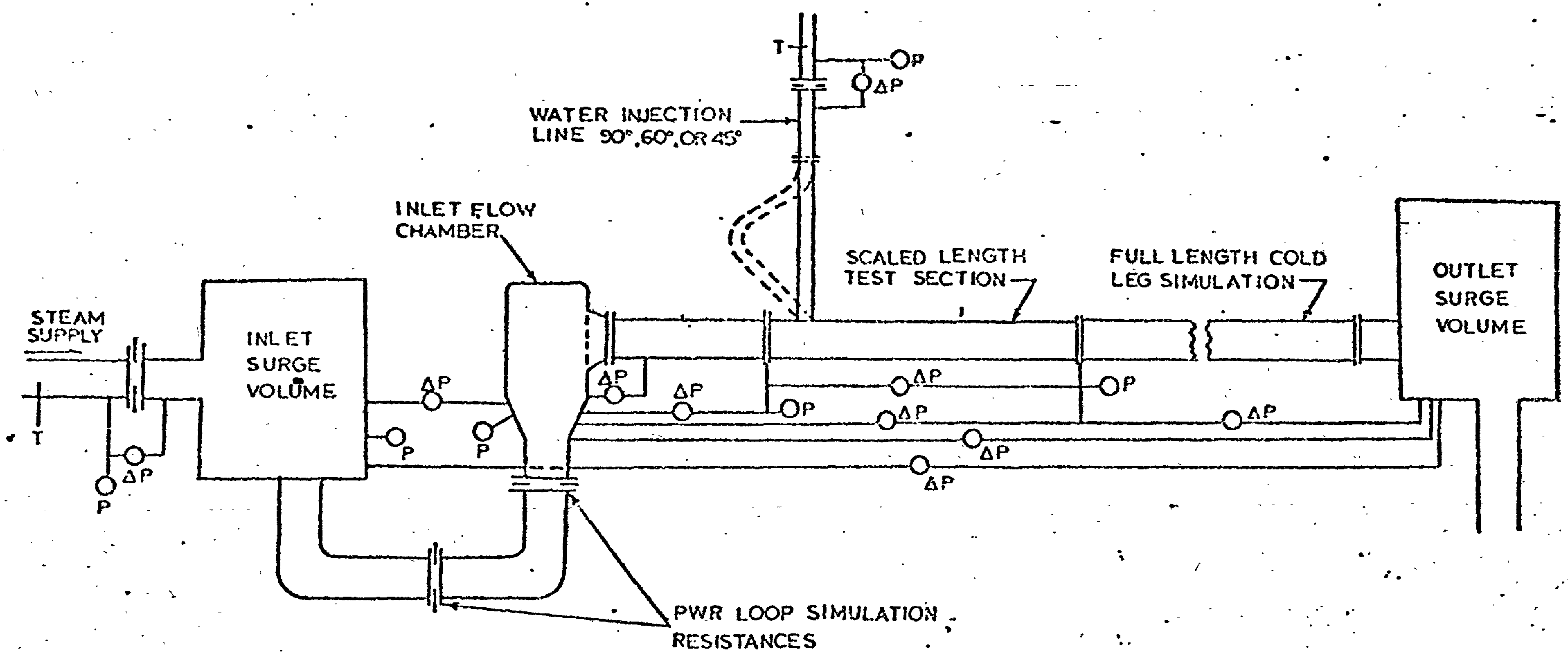

FIGURE 1
STEAM-WATER MIXING TEST CONFIGURATION
SHOWING PRESSURE AND FLOW INSTRUMENTATION

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	1.2	RS 0036 B

Vorhaben/Project Title	Land/Country
Notkühlprogramm - Niederdruckversuche Wiederauffüllversuche mit Berücksichtigung der Primärkreisläufe Emergency Core Cooling Program - Low Pressure Experiments. Refilling Experiments with Simulation of the Circulation Loop	FRG **Fördernde Institution/Sponsor** BMFT **Auftragnehmer/Contractor** KRAFTWERK UNION AG Reaktortechnik R 513, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 1. 73	30. 11. 78	D. Hein
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 77	7'641.190,-- DM

1. General Aim

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

2. Particular Objectives

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

3. Research Program

The test facility is designed to cover the following parametric variations:

- Max. initial clad temperature: 500 to 800 °C
- System pressure: 1 - 6 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: 0/5, 1/4, 2/3, 3/2, 4/1, 5/0
- Time function of injection rates: const, accumulator characteristic
- Break size: 0,25 to 2 F (double ended guillotine)
- Break location: hot leg, cold leg,
- Simulated pump resistance: locked rotor, free rotor
- Residual water in lower plenum: 0 to lower grid plate
- In loop seals: 0 to full

1. 1. 77 - 31. 12. 77

4. Experimental Facilities

In order to investigate the refill and reflood phase in a PWR including the feedback of the complete primary system, a test facility was built. Beside a 340 rod-testbundle it includes three scaled down primary loops with active steam generators.

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

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 acquisation system is capable of handling up to 300 channels at 1 Hz scanning rate.

5. Progress to Date

The test series IA were run with a break in the hot and cold leg. The measured signals were plotted and the mass balance was determined. The heat transfer coefficients were calculated.

During the test series IA the test facility was partially modified. A second test bundle was ordered. The instrumentation is being improved, regarding the advanced instruments of the USNRC, being under development, concerning

- liquid level detector systems
- spool piece measuring equipments
- film and impedance probes
- storz lense videa equipment

- 2 -

1. 1. 77 - 31. 12. 77

6. Results

In test series IA a number of 14 test runs were completed. The main interest was the influence of the loop resistance on the efficiency of cold leg injection compared to combined hot and cold leg injection. The results showed that combined hot and cold leg injection gives lower cladding temperatures, shorter quenchtimes and a smaller influence of the loop resistance in the flooding behaviour of the core.

7. Next Steps

The instrumentation of the heater rods will be completed. The spacers, contact plates and other systems will be adjusted to the extended instrumentation.

8. Relation with Other Projects

RS 36 C, RS 37 C, RS 287 (PKL test series IB and II)

9. References

LA-TR77-39: 1. Techn. Fachbericht RS 0036 B (Febr. 1975)
LA-TR77-44: 2. Techn. Fachbericht RS 0036 B (Jan. 1976)

10. Degree of Availability

The reports were translated into english by the University of California, Los Alamos Scientific Laboratory.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	1.2	RS 287
Vorhaben/Project Title Wiederauffüllversuche mit Berücksichtigung der Primärkreisläufe (PKL) Versuchsphasen I B und II Refilling Tests Considering the Primary Loops (PKL). Test Phases I B and II		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 513, Erlangen
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 9. 77	31. 12. 80	D. Hein
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	31. 12. 77	8'953.050,-- DM

1. General Aim

Experimental investigation of the refill and reflood phase during a LOCA, using a sufficiently large model of the entire PWR primary loop.

2. Particular Objectives

Special aims of this program are

- to perform approx. 15 more test runs in the PKL test facility using the currently installed 340 rod bundle
- selection and upgrading of additional instrumentation, especially for the second bundle
- to replace the current bundle by the new one
- to perform approx. 30 more test runs with the new bundle.

Parallel and subsequent to the test runs, the results will be evaluated and presented in test reports.

3. Research Program

Performance of further refill and reflood tests
Additional instrumentation provided for the second test bundle
Modifications and additions for the PKL test facility
Installation of the new bundle
Start-up

1. 1. 77 - 31. 12. 77

Performance of tests
Evaluation and documentation of test results.

4. Test Facility

The PKL test facility represents as closely as possible the primary loops including active steam generators of a 1300 MW plant, scale 1 : 134 (refering to number of heater rods). The conceptual design of the test facility has the following specific features:

- exact simulation of the core geometry and heating of the bundle
- exact simulation of all reactor elevations, locations of feed nozzles leading into the primary loops and pressure loss sequences
- sufficiently good simulation of the circuit volumes and the thermal capacities of the primary- and the secondary sides.

The test results are to verify the computer codes used for emergency core cooling analysis. In this connection special attention should be paid to the results obtained not only from the German FLUT and WAK programs, but also from RELAP4 MOD6 and TRAC which are developed in the USA as well as the REFLOS program which was developed by GRS from the US-code FLOOD4.

5. Progress to Date

For test series I B a suggestion was made to modify the upper plenum internals in order to provide a phenomenologically correct simulation of the upper plenum behaviour in the reactor.

1. 1. 77 - 31. 12. 77

In test series I A the volume of the lower plenum was found to be too large as compared to the model scale, which either led to excessively long refilling periods or necessitated an adjustment of the cold leg feed injection rate at the beginning of the test. For the test series I B an adjustment to this volume should be made.

The following additional work for test series IB was performed:

- installation of a shock absorber in the NW 200 connecting line between break line and separator tank
- assembly and start-up of the repaired feed pump
- disassembly of the second downcomer since the test series IB will be performed with one downcomer tube only

For the purpose of upgrading and evaluating the series I A measurement results the following work was performed:

- start of a test evaluation with regard to an energy balance for individual sections
- preparation of data tapes and additional information relevant for further evaluation at GRS.

6. Results

Test series IB is being prepared.
The PKL feed injection pump is operating again; this significantly facilitates the test procedure. The overall plant is approved by the TÜV.

7. Next Steps

After approval has been received from "SK Notkühlung" "Safety Commission - Emergency Cooling), the internals for the upper plenum and the displacement bodies for the lower plenum will be fabricated and assembled.

1. 1. 77 - 31. 12. 77

Additionally required measurement transducers for series I B will be installed and calibrated.

Test series I B will be startet during the first quarter 1978.

8. Relation with Other Projects

RS 0036 C

9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	1.2	RS 184

Vorhaben/Project Title	
Untersuchungen zur Hydraulik des Flutvorgangs und zu bisher noch unberücksichtigten Einflußgrößen beim Wiederbenetzen Investigations on the Influence of Hydraulics during Reflooding	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 513, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 10. 75	31. 12. 77	D. Hein
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	31. 12. 77	991.000,-- DM

1. <u>General Aim</u>

In order to improve the reflood models, the hydraulics during reflood will be studied in detail.

2. <u>Particular Objectives</u>

A detailed knowledge of the hydraulics in a channel during the reflood phase and the resulting flow pattern will improve the calculation of heat transfer in the unwetted area. With the help of this experiment, criteria will be worked out for the transition from vapour to fog flow and from fog flow to film boiling in order to get more information on the extent of the 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 prerequisite for a general applicability of theoretical calculations.

3. <u>Research Program</u>

Rewetting experiments will be carried out for tubes with different hydraulic diameters varying the parameters: initial wall temperature, inlet subcooling, feed velocity and system pressure. Also the ratio of stored heat to the water content within the channel will be varied.

A comparison will be made between Zircaloy and stainless steel clad in the annular test section for the progression of the rewetting front.

To improve the rewetting model information is needed on the rewetting temperature and on the effects near the rewetting front.

4. Experimental Facilities

For these experiments the test-rig used for the program RS 62 was modified in order to obtain more detailed information on the hydraulics during the reflood period. For these experiments constant inlet conditions will be stressed.

For special tests an annular test section with a quartz-glass tube for the outer wall will be used.

5. Progress to Date

Modifications of the test facility have been completed. Instrumentation for the first test series was modofied. The inlet and outlet line concept ensures the specified boundary conditions as requested for these tests.

To check the test facility two reproducibility tests were performed with the following parameters:

pressure:	4.5 bar
wall temp.	600 °C
inlet subcooling	75 °C
flooding rate	6 cm/s
heat output	3 W/cm^2

A total of 58 tests was performed on the flooding hydraulics, using two test sections of internal diameters 11.8 and 16.8 mm. Both the measuring probe for determination of the total two phase flow momentum as occurring at the outlet as well as a measurement device for determination of superheated steam temperature were used for these tests. Both measurement devices were subsequently improved following the tests.

the tests were performed under the following conditions:

tube diameter:	11.8 and 16.8 mm
tube length	3 m
tube wall initial temp.	600 °C
heat output	4 and 4,5 kW
pressure	1,5 and 4,5 bar
fluid inlet temp.	25 °C, 75 °C (ϑ_S - 3) °C
flooding rate	2; 6 and 10 cm/s

An existing flow model was chosen as a basis for the HYDROFLUT program.

The YAMANOUCHI model dealing with the progression of the rewetting front was integrated into the HYDROFLUT program. Calculations were performed with both RS 62 tests and tests taken from literature. For proper presentation of the results a plot routine was prepared.

The data tapes of all tests were CDC catalogued and stored in a collecting tape by means of a 2-byt-measurement data program - taped by the data acquisition system - to be adjusted to the CDC-specific word length. The evaluation program MESSDAT provides data which will be compared with the theoretically calculated values by means of the HYDROFLUT program. Some tests from both series were evaluated.

In order to cut down on computer costs, time steps in the HYDROFLUT program were increased. For stability reasons a change-over from differential to integral formulations of the channel mass balances was necessary.

1. 1. 77 - 31. 12. 77

6. Results

The tests showed that -under as-specified test conditions- a thermodynamic non-equilibrium must be expected in the unwetted region during flooding from below. The knowledge of the actual driving temperature gradient will permit the use of generally applicable heat transfer correlations for the unwetted region of the test section.

Tests performed with modified test line inlet components rendered defined, adjustable and temporary constant coolant inlet temperatures. The reproducibility tests conducted, revealed a satisfactory agreement with both the given boundary conditions and the preliminary test procedure. Useful results were obtained on the test evaluation covering the check of the momentum measurement device, including steam mass measurement at the test section outlet.

The YAMANOUCHI model on the progression of the wetting front can also be applied for single tube tests if the wetting temperature is determined as a function of the locally and timewise varying sub-cooling.

Only some of the 11.8 mm and 16.8 mm I.D. tube tests have been evaluated up to now. Thereby special attention was paid to the determination of phase slip since verification of the HYDRFLUT tests revealed a considerable dependency of the wall temperatures on the estimated phase slip during reflooding. The determination of the phase slip is performed by measurement of the outlet momentum. Currently evaluated tests indicate that phase slip in dispersed flow lies between 1 and 3.

7. Next Steps

Evaluation of the tests will be continued. The results gained thereby will be integrated into the HYDROFLUT reflooding program. Subsequently several comparative calculations will be performed with this improved program.

1. 1. 77 - 31. 12. 77

8. Relation with Other Projects

RS 36, RS 62

9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	1.2	RS 268

Vorhaben/Project Title	
Vorprojekt zur experimentellen Untersuchung der Einflüsse mehrdimensionaler Effekte beim Fluten Preliminary Project: On the Experimental Investigations of Multi-Dimensional Effects Influencing Flooding	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 52, Karlstein

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 1. 77	31. 1. 78	Dr. Melchior
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 77	473.432,-- DM

1. General Aim

Preparation of a feasibility study for the "3D-experiment". This experiment will serve to study the 3-dimensional flow effects in the upper plenum of a PWR during the refill and reflood phase after a LOCA. Moreover computer programs describing these effects will be verified with the experimental results.

2. Particular Objectives

This study is expected to render results upon which a decision will be made on whether or not the "3D-experiment" will be performed.

3. Research Program

3.1 Specification of the planned test, estimate of usefulness by the architect engineer.

3.2 Systems-related technical design of the test stand (test stand concept).

3.3 Procedural preparation of the test stand concept.

3.4 Preparation of specifications for the measurements and data acquisition, conceptual design of the data acquisition system.

3.5 Evaluation of expenditures, dates and personnel requirements for the test.

1. 1. 77 - 31. 12. 77

4. Experimental Facilities

Experimental Facilities are not required for the preliminary project.

5. Progress to Date

The complex task, test stand, measurement technique and data acquisition were prepared and data recording was started.

Cost and schedule estimates of the 3D-experiment were made.

Documents on already fixed technical master data of the 2D/3D experiments were prepared for discussions with USNRC which were held in Washington (Nov. 1977). Also, an English description of the 3D test stand as well as an oral presentation were prepared, the latter was then presented during the Review Group Meeting.

At the request of the Advisory-Committee "Emergency Cooling", a rationale for inclusion of the refill phase into the 2D-, 3D-experiments as well as benefit analysis for the 3D experiment were presented.

6. Results

Estimated costs for planning, construction and test performance with a KWU plenum are approx. DM 75 M for an overall test period of 6 years.

7. Next Steps

Completion of test task.
Preparation of a final report.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Classification: 1.2

Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: NORHAV - RHC a core heat-up computer program	Organization: Risø National Laboratory
Initiated date: November 1971 Completed date: 1976 Status: completed	Scientists: J.G.M. Andersen H. Abel-Larsen Preben Hansen

1. General aim

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

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, called NORCOOL-I.

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 transient subchannel computer program TINA and the one dimensional blow down code RISQUE under development at 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 G. Munthe 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

Classification: 1.2

Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: Inverted annular film boiling during the reflooding phase.	Organization: Risø National Laboratory
Initiated date: September 1977 Completed date: Status: progressing.	Scientists: Per Ottosen

1. General aim

Heat transfer on vertical surfaces in tubes under inverted annular film boiling.

2. Particular objectives

An experimental and theoretical work concerning inverted film boiling in glass tubes using N_2 as flow medium and steel tubes using water as flow medium has been started.

Later on a theoretical work based on the experimental results will be started.

3. Experimental facilities

The experimental part consists of development of a visual test section made of glass tubes.

Furthermore, the experimental part consists of development ot various methods of void measurements in a two-phase flow, using x-rays and hot-wire constant temperature anemometry.

4. Status

1. Progress to date

The first visual experiment using N_2 as a two-phase flow in a heated glass tube has been tried.

The instruments for the measurements are still being developed.

5. Next steps

Construction of a test loop, in which it is possible to establish inverted film boiling using water as medium.

145-1 - 07/4160-20		1.2
Titre Thermohydraulique du LOCA. Etude des interactions mécaniques et thermo-dynamiques dans l'injection de secours d'un réacteur PWR : Programmes EPIS I et II.		**Pays** FRANCE **Organisme directeur** CEA/DgCS - EDF/SEPTEN
Titre (anglais) LOCA thermohydraulics. Steam-water mixing studies for PWR : EPIS I and II projects.		**Organisme exécuteur** CEA/DTCE (Saclay) **Responsable** (SEEN) Saclay
Date de démarrage 01/01/75	**Etat actuel** en cours	**Scientifiques**
Date prévue d'achèvement 31/12/78	**Dernière mise à jour** 1/78	

1 - Objectif général :

Etude des phénomènes se produisant lors de l'injection d'eau de secours par accumulateurs et pompes au cours d'un accident de dépressurisation d'un réacteur pressurisé.

2 - Objectifs particuliers :

Développer des modèles physiques pour interpréter les expériences.

3 - Installations expérimentales et programme :

EPIS 1 : Etude de l'interaction mécanique par injection d'eau dans un débit d'air (échelle 1/11).
EPIS 2 : Etude des interactions mécaniques et thermodynamiques par injection d'eau dans un débit de vapeur (échelle 1/25).

.../

4 - Etat de l'étude :

1) Avancement à ce jour :

Essais EPIS 1 terminés. Essais complémentaires EPIS 1' terminés. Installation EPIS 2 opérationnelle.

2) Résultats essentiels :

- EPIS I :

Evolution des pertes de charge au niveau de l'injection en fonction des paramètres principaux :
Angle de piquage de l'injection, rapport des vitesses de l'eau et de l'air, niveau de pression dans la cuve.
Test de différents modèles physiques (en cours) qui permettraient de retrouver soit le ΔP total soit la ligne piézométrique.

- EPIS II :

Observation de phénomènes oscillatoires dans des domaines précis de températures et de titres.

5 - Prochaines étapes :

- EPIS I :

Continuer la modélisation.

- EPIS II :

Nouvelle reconnaissance des phénomènes.
Définition des paramètres d'essais.
Campagnes d'essais.

7 - Documents de référence : rapports internes non disponibles

	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 device 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.

Title 1 (Original language)	Classification
Sistema di raffreddamento di emergenza per allagamento dal basso	1.2
Title 2 (English) Bottom flooding ECCS	Country : ITALY Sponsor : CNEN Organisation : CISE
Date initiated 1971 Date completed 1978 Last updating April 1977	Project Leader UTM (CISE)

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 N. 1.1.2) 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.): Test completed;
- (3.2.2.): Constant flowrate tests completed;
- (3.2.3.): TRAFEM code for constant flowrate and uniform axial pressure condition completed. Comparison with experimental results in progress.

4.2. Essential results

- basic understanding of the physical phenomena involved;
- set up of the experimental procedures and techniques;
- set up of a calculation procedure

5. Next steps

- Full scale experiment with controlled pressure drop (parallel channels simulation)

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TITLE 1 (original language)	Classification
Sistema di raffreddamento di emergenza per allagamento dal basso	1.2

- Completion of comparisons with TRAFEM code
- Computation of pressure drops in TRAFEM code
- Flow restrictions investigation with full scale bundle (cladding-ballooning)

6. Reference documents

1) R. Martini, A. Premoli " A simple model for predicting E.C.C. transients in bottom flooding conditions" CREST Meeting - Munich, October 18-20, 1972.

2) R. Martini, A. Premoli "Bottom flooding experiments with single geometries under different E.C.C. conditions" Energia Nucleare, vol. 20, n. 10, pp.540-553 October, 1973

7. Degree of availability: to a limited extent

Title 1 (Original language) Apparecchiatura sperimentale per lo studio della termoidraulica nella refrigerazione di emergenza	Classification 1.2
Title 2 (English) An experimental facility to study thermohydraulic aspects of emergency core cooling by bottom flooding	Country ITALY Sponsor / Organisation Politecnico di Torino (^)
Date initiated January 1976 Date completed December 1977 Last updating April 1977	Project Leader M. De Salve

(^) Istituto di Fisica Tecnica e Impianti Nucleari

1. - General aim and particular objectives

This experimental and theoretical study is to improve the knowledge of the emergency core cooling by bottom flooding.

The particular objectives are: to measure the rewetting time and the wall temperatures, to estimate the heat transfer coefficient during flooding.

2. - Experimental facilities and programme

An experimental facility with an inner heated annular test section has been built. The inner circular tube wall temperatures are measured by several thermocouples and it is possible to see the climbing liquid level by two glass windows. Investigations are restricted to atmospheric pressure, small flooding rates, high initial wall temperatures ($T \approx 800$ °C) and high subcooling. Some tests have been performed.

3. Reference documents and degree of availability

The reference documents are the usual and open bibliography about the emergency core cooling. All the results will be available.

4. - Resources

The expected budget is about one million of Lire (1.000.000.= Lit.). Manpower consists of two man-years.

Title 1 (Original language)	Classification
Scambio Termico in condizioni di raffreddamento di emergenza	1.2
Title 2 (English)	**Country** ITALY
Heat Transfer in ECCS conditions	Sponsor / Organisation: Calabria University
Date initiated January 1977 Date completed December 1978 Last updating March 1977	Project Leader Prof. Valerio Marinelli

1) General aim
Study the heat transfer between rods and coolant during the ECCS conditions

2) Particular objectives
Optimization of engineering correlations and models to predict the thermal behaviour of rods during ECCS; development of a computer code.

3) Experimental facilities and programme.
Experiments of bottom flooding in rod-annular geometry at low pressure, starting from different initial temeperatures of the rod, and experiments of spray mode of cooling in a second step.

4) Project status
At present a survey of literature is in progress and the conceptual design of the experiments is under way.

5) Next steps
Planning of experiments.

8) Degree of availability
Full availability for the parts of the programme completely sponsored by University and not supplied by external contracts.

Title 1 (Original language)	Classification
Sviluppo di una catena di programmi per l'analisi del LOCA di un PWR con nocciolo in acciaio.	1.2
Title 2 (English)	**Country** ITALY
Development of a chain of digital programs for the LOCA analysis of a PWR having a SS cladding core	**Sponsor** FIAT-T.T.G. **Organisation** Nuclear Energy Division
Date initiated 1971 **Date completed** 1976 **Last updating** 1977	**Project Leader** G.P. Pozzi

1. General aim

Development of a chain of computer codes for the loss of coolant accident (LOCA) analysis of a pressurized water reactor having a stainless steel cladding core.

2. Particular objectives

Application of the above chain to the LOCA analysis of the TRINO Vercellese reactor. Design of the new safety injection sistem (accumulators and emergency pumps), for the TRINO Vercellese reactor according to the 1974 USA Final Acceptance Criteria.

3. Experimental facilities

A set of core heat up and reflooding experiments was performed near the SORIN (Società Ricerche Nucleari) of Saluggia (Vercelli).
In particular the low reflooding velocities and the linear heat rate of the TRINO V. reactor were tested.

4. Project status

Four different codes have been set up: for the blowdown phase; for the calculation of detailed flow and enthalpy distribution in the core; for the core reflooding phase; for the core heat up and cladding temperature turn-around phase.
A comparison of the prediction of these codes against experimental results was performed.
The main work performed with this chain of digital programs were:

a) LOCA analysis of the SIEMENS ATUCHA nuclear plant;

b) design of the safety injection system of the CLEOPATRA plant to be installed in the ISPRA ESSOR reactor;

c) design of the TRINO V. new safety injection system.

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Title 1 (Original language)	Classification
Sviluppo di una catena di programmi per l'analisi del LOCA di un PWR con nocciolo in acciaio.	1.2

5. Next steps

Project completed

6. Relation to other projects

The developed codes will be utilized for the other project just started: "Development of an advanced procedure for the PWR safety analysis following the probabilistic approach" (see present Safety Research Index)

7. Reference documents

Rapporto FIAT-FN-C-47 "Analisi dell'adeguatezza dal nuovo sistema di refrigerazione di emergenza progettato per la centrale nucleare di TRINO Vercellese" Maggio 1976

8. Degree of availability

To a limited extent

9. Budget, personnel involved

3 engineers for 5 years.

Title 1 (Original language)	Classification
Analisi dei transitori termici ed idraulici a seguito di LOCA nei reattori ad acqua leggera.	1.1.1, 1.1.2, 1.1.4, 1.2
Title 2 (English)	**Country** ITALY
Analysis of thermal and hydraulic transients following a LOCA in Light Water Reactors	**Sponsor** CNEN and CNR **Organisation** University of Pisa
Date initiated 1974 **Date completed** 1978 **Last updating** may 1977	**Project Leader** N. CERULLO

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

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322

4.) Project status

1.) Progress to date : 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.) Essential results : 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.) Next steps : Experimental investigation of the different flooding conditions.

6.) Relation with other projects : The programme has been planned so as to be complimentary to other work in the quenching field

7.) Reference documents :

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 24,(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 21, (2), pp 127-135 1973.

8.) Degree of availability : Freely available

9.) Budget : The expected total investment from the CEC is 65 000 UA which includes the cost of the facility and the running costs.

10.) Personnel : 5 men/year

11.) Additional information :

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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 Completed : Status : progressing December 1977 (BMFT part A) Last updating : March 1975	

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PROJECT TITLE : Loop Blowdown Investigations (LOBI)-Project : Influence of PWR primary loops on blowdown.	LWR $\boxed{1.1}$ 1.2
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : J.R.C. Ispra
DATE INITIATED : January 1974 DATE COMPLETED : December 1976	PROJECT LEADER : W. Riebold

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PROJECT TITLE : Blowdown code assessment	LWR $\boxed{1.1}$ - 1.2
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : JRC Ispra Establishment
DATE INITIATED : Jan. 1974 DATE COMPLETED :	PROJECT LEADER : L. Larsen

323/7.

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

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.

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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
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

Classification

1.2

Title 1 PERFORMANCE OF SPRAY COOLING	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION AEE WINFRITH
Title 2	Project Leader
Initiated 1968 Completed : Status : Last updating 1976	Scientists:

Description:

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.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1977	1.3	PNS 4231

Vorhaben/Project Title	
Theoretische Untersuchungen zum Brennelement-verhalten bei Störfalltransienten Theoretical Investigations of Fuel Behavior under Transient Conditions	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Kernforschungszentrum Karlsruhe KFK Projekt Nukleare Sicherheit IRE und IKE Stuttgart

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1973	1980	Dr.R.Meyder, Prof.H.Unger
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1977	

1. General Aim

The aim of this project is the development of a code system (SSYST) in order to model and calculate the behaviour of Zircaloy clad fuel rods in different phases of a LOCA. Especially the effect of ballooning and its consequences are studied and described with particular emphasis. Also, the theoretical investigation of the influence of ballooning on the effectivness of emergency core cooling is a major aim of the planned activities.

The theoretical studies are performed in close cooperation with the "Institut für Kernenergetik" (IKE), Stuttgart.

2. Particular Objectives

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 the clad, pressure in coolant and fuel rod as well as the thermo- and fluiddynamic conditions in the coolant channel and the primary system of a LWR. Additional theoretical investigations are concerned with cooling conditions in deformed areas of fuel elements and the interactions between rods.

3. Program

The development covers three topics

- Development and verification of a computer code for single rod analysis of a LOCA
- Investigation of interacting fuel rods, behaviour of bundles.
- Investigation of geometrical bundle configurations for longtermcoolability.

4. Computer codes

Code system SSYST

I.I.-3I.I2.I977

5. Progress to date

The verification of SSYST has been continued. Experimental results from PNS 4238, TREAT and PCM have been calculated (standard problems), comparisons with FRAP-T4 calculations have been performed. Additional SSYST-calculations have been performed for a single rod geometry in order to back up the out of pile experimental program with rod bundles anticipated in within the frame of the research project PNS 4238. Several heat transfer parameters have been varied. The simulation of the transient behaviour of the pin has been performed under reflood conditions also.

In order to test steady state fuel rod perfomance codes for use in SSYST comparisons have been made. On behalf of a single fuel rod, for which the centerline temperatures were measured, SATURN-L and FRAP-S calculations have been compared. Additionally, comparisons between CARO-D and FRAP-S were made for a fuel rod model.

In COBRA IIIC extended steam tables have been implemented in order to enable the calculation of fuel rod bundle experiments with superheated steam. The modules FRAPDR and FRASSY have been tested and improved also. Several SSYST-modules have been improved. A second version of the module ZIRKOX has been written. It calculates the heat of the Zr/H_2O- reaction the amount of H_2 released, the Zr volume increase and the reduction of the steam due to oxidation. By means of partially seperate treatment of the two phases steam and water in the equations of the flow model in the module ZETHYF an improved description of the two phase mixture has been achieved. A mathematical model for a transient fission gas pressure module has been established. Special emphasis has been put on the calculation of the initial steady state conditions. The module ODRUSPA for the calculation of the pressure in the gap has been improved. An additional module for the calculation of the heat transfer in the gap has been developed. The deformation module STADEF2 has been improved. The module ZWERG for the conditioning and processing of data has been improved for use in SSYST-2.

For the statistical analysis of rod behaviour according to the response surface method a number of SSYST modules have been developed:
- generating of input data vectors for the statistical variables in RELAP and SSYST according to "experimental design" of "hypercube sampling".
- evaluation of the response surface with a linear regression analysis to find an interpolation function.

- evaluation of the interpolation function with Monte Carlo to find the probability density function e.g. peak cladding temperature.

For a hot fuel rod a master input for a LOCA analysis with SSYST was compiled and tested.

For the development of a Zircaloy cladding material model the data base for ID creep and tensile tests was set up and analysed in a first step.

A proposal was made how to evaluate the amount of damage on the fuel rods of a reactor after a LOCA.

6. Essential Results

The calculations for the verification of SSYST-I with respect to temperature and expansion of the clad showed satisfactory results which were in good agreement with the experiment PNS 4238. The verification on behalf of TREAT and PBF experiments (FRF-2 and PCM) was also satisfactory although several numerical problems had to be solved in order to obtain results of acceptable quality.

The coupling of the program FRAP-S with SSYST by means of two auxiliary modules FRAPDR and FRASSY now allows a detailed calculation of the initial state of the fuel pin (as a function, for example, of the operation conditions and the power history) as well as the following transient calculation with SSYST. A reduction of the input set for the transient calculations with SSYST has been achieved also. By means of an improvement of the basic equations for the calculation of the transient fission gas pressure occationally occuring instabilities could be eliminated.

The variation of fuel pin heat transfer parameters showed a strong dependence of the clad temperature on the gap width and an extremely sensitive dependence of the ballooning on the clad temperature.

SSYST calculations of a TREAT experiment and a comparision with FRAP-T4 were in good agreement with respect to the clad temperature rise during heatup phase.

SATURN-L and FRAP-S calculations were in good agreement with respect to the centerline temperatures of the pins. The gap heat transfer coefficients showed larger deviations which disappeared however when gap closed. The contact pressure values were different, especially at high power. Comparisons between CARO-D and FRAP-S showed good agreement for reactor relevant power levels.

I.I.-31.I2.I977

The analysis of the data on Zircaloy cladding behaviour in ID tensile and creep tests had the following results:

- the onset of creep in the secondary creep regime can be described with Norton type relations where the coefficients are a function of temperature
- for some temperature intervals strains up to 30% respectivly 70% can be described.
- the presence of oxigen during deformation increases the rupture strain for $870K \leq T \leq 1070K$ and reduces rupture strain for higher temperatures.

Sample calculations with the regression analysis module showed, that it is possible to handle more than 20 variables with this method in a satisfactory manner.

By means of extended steam tables in COBRA IIIC a substantial reduction of the input data set for the coolant could be achieved. A new version of the module ZIRKOX allows a realistic calculation of the influence of the steam limitation on clad temperature and pin diameter. Measured and calculated data agree well.

7. Next Steps

The steps for the near future are:

- investigations and improved simulation of the refill phase.
- supporting calculations for the experimental programs PNS 4236 to 4239
- investigation of cooling conditions in deformed areas.
- generating a response surface for c.ad deformations for a hot bundle.
- constructing a data base for burst data.

8. Relation with other Projects

This project is part of the major project PNS 4230 of the Kernforschungszentrum Karlsruhe. It is supported by the other parts of the major project as well in the development of models as in their verification.

9. References

Gulden, W. et al.
Dokumentation SSYST-I. Ein Programmsystem zur Berechnung des Brennstabverhaltens bei einem LWR-Kühlmittelverluststörfall.
KFK 2496, August I977.

I.I.-31.12.1977

Meyder, R., Unger, H.
Beiträge zum I. Halbjahresbericht 1977 des Projekts Nukleare Sicherheit des Kernforschungszentrum Karlsruhe.
KFK 2500

Meyder, R., Unger, H.
Beiträge zum 2. Halbjahresbericht 1977 des Projekts Nukleare Sicherheit des Kernforschungszentrum Karlsruhe.
KFK report in preparation

IO. Degree of Availability

Unrestricted ditribution

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	1.3	RS 177

Vorhaben/Project Title	
Vorläufige empirische Beschreibung des Verhaltens von Brennstäben bei hypothetischen Kühlmittelverluststörfällen Preliminary Empirical Description of the Fuel Rod Behaviour during LOCA	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik RB 2, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 9. 75	30. 9. 77	Dr. Wunderlich
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	31. 12. 77	168.200,-- DM

1. <u>General Aim</u>

Deformations of fuel rod tubes during LOCA are to be described empirically. The experimental data of the ballooning and fracture tests gained from RS 107 are used for the calibration of a material law, which describes analytically the process of fuel rod ballooning during a LOCA.

2. <u>Particular Objectives</u>

For ballooning the creep law derived by Norton must be improved because constant Norton parameters are not applicable in the total range of stresses and temperatures during a LOCA.

3. <u>Research Program</u>

- development of the empirical material law
- adjustment of the parameters to the data of the RS 107 tests
- discussion of the influences of indirect heating, oxidation and hydriding on the strain behaviour.

4. <u>Experimental Facilities</u>

No experimental facilities were necessary.

5. <u>Progress to Date</u>

The modofied Norton equation to describe the prior to rupture strain behaviour as shown in [1] and [2], was fitted to burst-strains. It led to a not yet fully satisfactory description of the strain behaviour.

Therefore a new fitting of the parameters of this equation to prior to rupture strain results has been performed. Maximum strains of unburst tubes from transient tests and from creep rupture tests with direct heating have been used for this purpose.

To verify this new equation, the clad strains at all elevations of temperature measurement (4 Pyrometer measurements at different axial positions) have been calculated for each tube.

The modified Norton equation itself cannot predict burst strains. Therefore it must be combined with a burst criterion. The burst criterion is simply taken as the experimentally derived dependence of the strain at rupture on the burst temperature. By fitting the burst strain versus burst temperature with a least squares method, it was found empirically, that the burst strain is a function of temperature and heating rate.

Experimental results from internally heated tubes and from pre-heated (oxidized and oxidized + hydrided), directly heated tubes have been compared with the predictions of the above described models for the prior to rupture strain and for the rupture strain.

6. Results

The verification of the creep formula through the calculation of the strains at the elevations of temperature measurement showed good agreement.

The combination of the prior to rupture strain model (modified Norton-equation) with the burst criterion allows the calculation of the strain as a function of time until rupture. This model is calibrated against data from directly heated tubes, i.e. tubes with circumferentially and axially uniform temperatures.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

RS 107: Behaviour of Zry-4 Canning Tubes under Loss-of Coolant Accident Conditions

9. References

[1] B. Brzoska et al
"A New High Temperatur Deformation Model for Zircaloy Clad Ballooning under Hypothetical LOCA Conditions".
Vortrag auf der 4th International Conference on: "Structural Mechanics in Reactor Technology", San Franzisko vom 15. - 19.8.1977

[2] B. Brzoska et al
"Ein neues Modell zur Beschreibung des Dehnungsverhaltens von Zircaloy-Hüllrohren während hypotehtischer Kühlmittelverluststörfälle"
Reaktortagung 1977, Mannheim vom 29.3. - 1.4.77

10. Degree of Availability

Compacts available.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.77 - 31.12.77	1.3	PNS 4236

Vorhaben/Project Title	
Untersuchungen zum Brennstabverhalten in der Blowdownphase eines Kühlmittelverluststörfalles Investigations of the Fuel-Rod-Behavior during the Blowdown-Phase of a Loss-of-Coolant Accident	Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS) IRE/RBT

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.7.1972	1979	Dr. G. Class, IRE/ K. Hain, RBT
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	December 1977	

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. Research Program

The experimental program so far includes the simulation of hot and cold leg breaks with sizes of 1F and 2F. In each case experiments will be carried out at different rod powers and internal pressures.

4. Experimental Facilities, Computer Codes

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.

1.1.77 - 31.12.77

5. Progress to Date

After the difficulties encountered in commissioning the electronic data recording system and the interface for the computer controlled development of blowdown had been overcome, test operation started at the end of the first half-year.

Also the special measurement techniques allowing pyrometric measurement of the cladding tube temperature during the transient and, above all, the true-mass-flowmeter (TMFM) for direct measurement of the transient two-phase mass flow had been developed so as to be ready for operation under blowdown conditions.

The computer program intended for evaluation was completed and tested. Additional programs were set up for data conversion and graphical representation of the measured values. The first blowdown tests were evaluated to the extent possible.

6. Results

In the first 12 computer controlled tests a valve actuating program was iterated to simulate a 2F rupture in the cold line between the pump and the reactor pressure vessel. In these tests the fuel-rod-simulator containing Al_2O_3 annular pellets proved to be fully performing. Fuel-rod powers exceeding 700 W/cm were managed without difficulties.

The results measured in these tests show that the plant and control concept chosen allows to set specified reactor typical test conditions in the test section. The pressure and cladding tube temperature transients were satisfactorily adapted to the reference plots.

7. Next Steps

The test rod for measurement of heat transfer coefficients will be completed and put in operation. Parallel with test operation the development of the simulating fuel rod containing ThO_2 pellets will be continued. In addition to the evaluation of current tests theoretical work will concentrate on the elaboration of control programs for different break-parameters.

1.1.77 - 31.12.77

8. Relation with other Projects

PNS 4231, 4237, 4238, 4239

9. References

1st semiannular Progress-Report PNS 1/1977
2nd semiannular Progress-Report PNS 2/1977

10. Degree of Availability

Unrestricted distribution

Berichtszeitraum/Period 1.1. - 31.12.1977	Klassifikation/Classification 1.3	Kennzeichen/Project Number PNS 4238
Vorhaben/Project Title Studies of the Interaction between Ballooning Zircaloy Claddings and the Emergency Core Cooling (REBEKA Program) Untersuchungen zur Wechselwirkung zwischen aufblähenden Zirkaloy-Hüllen und einsetzender Notkühlung (REBEKA-Programm)		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KfK Projekt Nukleare Sicherheit (PNS) Institut für Reaktorbauelemente (IRB)
Arbeitsbeginn/Initiated 1973	Arbeitsende/Completed 1981	Leiter des Vorhabens/Project Leader K. Wiehr
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating December 1977	Bewilligte Mittel/Funds

1. General Aim

The general aim of the project is the development of experimental information about the ballooning of zircaloy claddings during the refill and flooding phases of a hypothetical loss-of-coolant accident which is to verify and further developing the SSYST code system.

2. Particular Objectives

The deformation behavior of zircaloy claddings is studied under various transient boundary conditions in single rod and bundle experiments. For this purpose, full length fuel rod simulators with axial power profiles and arranged in bundles under approximately representative thermohydraulic flooding conditions are being used. The project serves the particularobjectives outlined below:

- Assessment of ballooning of single rods as a function of time.
- Assessment of the influence of emergency core cooling on the ballooning event.
- Studies of the thermal and mechanical interactions between adjacent rods during ballooning in a rod bundle.
- Generation of information about possible failure propagation.
- Studies of the extent and distribution of cooling channel blockages.

3. Research Program

The experimental program is carried out in a number of consecutive series of experiments starting from single rod experiments in a steam atmosphere and finishing with bundle flooding experiments of bundle assemblies including at least 5x5 rods.

All experiments are accompanied by basic experiments on separate effects and by theoretical studies.
The experiments are carried out as parameter experiments. The experimental parameters are based on the assumed power generation and the thermohydraulic boundary conditions accompanying a loss-of-coolant accident:

- rod power	10 - 25 W/cm
- axial power profile	step and cosine shaped, respectively
- internal rod pressure	20 - 100 bar
- systems pressure	1 - 4.5 bar
- cladding temperature at onset of flooding	700 - 900° C
- flooding rate (cold)	1 - 30 cm/s
- flooding water temperature	25° C - saturation.

4. Test Facilities

Test rigs are available for single rod experiments using shortened fuel rod simulators in air and steam atmospheres, respectively, and there is also a test loop for bundle experiments involving flooding of full length fuel rod simulators with axial power profiles.
In the single rod experiments on shortened fuel rod simulators it is possible to simulate temperature transients of the type calculated in the refilling and flooding phases of a loss-of-coolant accident for rods of various power levels by combining the external heat transfer conditions with the rod power of the fuel rod simulator. - In the test loop for full length bundle experiments under flooding conditions a representative cladding temperature curve is generated largely automatically as a consequence of the good simulation quality of the fuel rod simulator and of representative emergency cooling conditions.

The deformation of the zirkaloy cladding tubes is recorded by X-ray filming and evaluated.
Some 130 measuring data (temperature, pressure, level, power, etc.) are recorded 10 times per second each with a cycle frequency of 10 kHz by means of a fast data acquisition system and the CALAS process computer.

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5. Progress to Date

In the period under review most of the effort was concentrated upon the implementation of the following activities:

- Experiments on shortened fuel rod simulators in a steam atmosphere
  - . under stylized temperature transients for rods of normal to maximum rod powers,
  - . under varied isothermal starting temperatures, in order to indicate the influence upon the burst strain of the azimuthal differential temperature.
- Preliminary experiments on the first bundle experiment with flooding
  - . to determine the heating rates,
  - . to determine the isothermy of the test section,
  - . to set the flooding rate
  - . to determine the onset of flooding.
- First bundle experiment with flooding in a 5x5 rod configuration with full length fuel rod simulators of an axial power profile.
- Evaluation of the experiments.

6. Results

The single rod experiments conducted on shortened fuel rod simulators in a steam atmosphere were based on temperature transients of the type computed for a double ended cold leg break. The temperature curves were simulated by means of power control. One typical characteristic of the series of experiments was the balancing out of azimuthal differential temperatures on the cladding at 300°C and 600°C, respectively, before the initiation of heating.

The temperature curves as recorded showed that even if all differential temperatures are equalized on the cladding tube at approx. 300°C, considerable azimuthal differential temperatures will yet form again under the set power and cooling conditions in the course of the strain exerted on the cladding. This is true even at an isothermal starting temperature of 600°C. This limits the area of the maximum decrease of the wall thickness to a small part of the circumference of the cladding tube, which prevents major mean strains.

The experiment indicated a systematic relationship between the azimuthal differential temperatures and the resultant burst strain: large azimuthal differential temperatures resulted in small burst strains,

small differential temperatures in large strains.
In all experiments strains in excess of 33 %, which would mean contacts with adjacent ballooning rods, were limited to a relatively short axial section around the point of rupture. Axially extended cladding tube deformations were not observed in any case.
In the first bundle experiment fuel rod simulators with 3.90 m of heated length and a step type axial power profile were used in a bundle of 25 rods under approximately representative thermohydraulic flooding conditions. The experiment was based on rod power of medium rated rods with a rod power of 20 W/cm and a flooding rate of approx. 3 cm/s. This led to a temperature plateau of a maximum cladding tube temperature at approx. 800° C in the flooding phase.
The first bundle experiment carried out under these experimental conditions furnished the following results and trends:

- The generation of typical cladding tube deformations is decisively influenced by the correct simulation of the following parameters as a function of time and position:
  . the amount of heat generated inside the fuel rod simulator and heat transport over the gap to the cladding;
  . the external cooling conditions in the different phases of a loss-of-coolant accident.
- The changing gap greatly influences the development of the cladding tube temperature throughout the flooding phase.
- Pronounced lifting of the cladding leads to earlier quenching, less pronounced lifting to later quenching.
- Under the experimental conditions chosen, only 2 out of the 9 zircaloy claddings maintained under internal pressures of 70 bar were ruptured.
- The asymmetrical shape of deformation of the ruptured cladding tubes seems to indicate major azimuthal differential temperatures at the rupture point, which lead to the low rupture strains of 25 and 31 %, respectively.
- The maximum hoop strains of the unruptured cladding tubes were between 8 and 32 %.
- Because of increased local cooling acitivity, the spacers prevent major strains in the area of the spacers.

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- In the bundle experiment described above the maximum of the strains was shifted between the spacers in the direction of the flow towards the next spacer.
- The cladding tube deformations generated under the set boundary conditions of the experiment indicated a pattern of hardly disturbed cooling channel geometries in rod bundles and did not show a negative impact on the cooling capability.

7. Next Step

- Further single rod experiments to be applied to equations of state for plastic deformation of zircaloy in r-geometry.
- Studies of the influence of anisotropy effects and of bowing of zircaloy cladding tubes in the α-phase.
- Further bundle experiments with later onset of flooding and higher rod power, respectively.

8. Relations with other Projects

KFK: PNS 4231, 4235, 4236, 4237, 4239
KWU: RS 107, RS 36
USNRC: MRBT program

9. References

/1/ F. Erbacher, J.J. Neitzel, M. Reimann, K. Wiehr, "Ballooning in Zircaloy Fuel Rod Claddings in a Loss-of-Coolant Accident", ANS-Topical-Meeting on "Thermal Reactor Safety", July 31 - August 7, 1977, Sun Valley, Idaho, USA.

/2/ K. Wiehr, H. Schmidt, "Out-of-pile Versuche zum Aufblähvorgang von Zirkaloyhüllen - Ergebnisse aus Vorversuchen mit verkürzten Brennstabsimulatoren", KFK-2345 (Oktober 1977).

/3/ K. Wiehr, F. Erbacher, U. Harten, W. Just, H.J. Neitzel, P. Schäffner, H. Schmidt, "Brennstabverhalten in der Wiederauffüll- und Flutphase eines Kühlmittelverluststörfalles", Jahreskolloquium 1977 des Projektes Nukleare Sicherheit, KFK-2570.

/4/ PNS-Halbjahresbericht 1977/1, KFK-2500 (September 1977).

10. Availability of Reports

Available as KFK reports or conference reports.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	1.3	RS 185

Vorhaben/Project Title	
Parameteruntersuchungen über die Beeinflussung der Hüllrohre durch Nachbarstäbe beim Kühlmittelverluststörfall Investigations on the Influence of Neighbouring Fuel Rods during LOCA	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik, RB 32, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 10. 75	31. 3. 79	Dr. Weidinger
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 77	1'142.330,-- DM

1. 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 interaction are to be investigated.

2. 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. Especially 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.

3. Research Program

The influence of thermal and mechanical interaction between neighbouring rods will be investigated with convection cooling (air) and forced cooling for the following cases:

a) tests with homogenous temperature distribution:
Of special interest are the ballooning of one specimen in a non-disturbed surrounding and the ballooning of a specimen towards another deformed specimen in an non-disturbed surrounding.

b) tests with inhomogenous temperature distribution:
The specimens are heated to higher or lower temperatures compared with the surrounding dummies.

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Planned are also tests with two specimen with equal and different temperatures.

The tests will be run in air-atmosphere. The fuel rods are pressurized with helium.

4. Experimental Facilities

The test apparatus consists of the following equipments:

a) Test bundle consisting of 3 x 4 rod specimen with internal heaters. The two central rods are pressurized internally by helium.

b) Energy supply for heating of the bundle.

c) Pressure supply.

d) Measuring and recording equipment for temperature and pressure control.

5. Progress to Date

In total about 20 tests were run without forced cooling with following test procedures:

- One pressurized rod with higher or lower maximum temperature compared to the neighbouring rods

- Two pressurized rods with equal or different internal pressures with homogeneous temperature distribution into the bundle.

6. Results

Preliminary results are summarized in the following statements:
Tests with one pressurized specimen (65 bar)

- Specimen burst in the area of maximum temperature (axial and azimuthal)

- The more homogeneous the temperature distribution in the bundle, the larger the burst strain.

- Test rods, whose temperature are lower than the surrounding dummy rods, have the burst opening facing towards the hottest neighbouring dummy rod.

- Test rods, whose temperature are higher than the surrounding dummy rods will have burst opening in an undeterminate direction.

Tests with two pressurized specimen (50/65 bar)

- Test rods with equal internal pressure burst at rather the same moment.

- Test rods with different internal pressure burst with temporal interval according to different stresses.

- Test rods with equal internal pressure but with a shifted axial temperature profile burst again at about the same moment but in different axial positions according to the temperature maximum.

- Neither test rods with equal internal pressure nor with different internal pressure showed any interaction on the burst behaviour.

Preliminary conclusions from the test results are:

- The obtained burst strains of multi rod tests correspond to that of single rod tests.

- Thermal interaction is observed if neighbouring rods are on a higher temperature level.

- Mechanical interaction was not observed in any case.

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7. Next Steps

Preparation of multi rod tests with forced cooling by air.

8. Relation with Other Projects

RS 107

9. References

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10. Degree of Availability

Berichtszeitraum/Period 1. 1. 77 - 31. 12. 77	Klassifikation/Classification 1.3	Kennzeichen/Project Number RS 309
Vorhaben/Project Title Innendruckversuche an Einzelstabproben zur experimentellen Absicherung des Brennstabverhaltens in der Notkühlanalyse Single Rod Internal Pressure Tests to Experimentally Verify Fuel Rod Performance for Emergency Cooling Analysis		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik RB 32, Erlangen
Arbeitsbeginn/Initiated 1. 11. 77	Arbeitsende/Completed 31. 3. 80	Leiter des Vorhabens/Project Leader H.-J. Romeiser
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 77	Bewilligte Mittel/Funds 1'563.005,-- DM

1. General Aim

Further experimental parameter investigations of clad swelling under approximated LOCA conditions in single rod geometry.

2. Particular Objectives

2.1 Investigation of the specific- and overall influence of test parameters obtained on swelling shape and distribution

- cladding tube geometry (wall thickness variation)
- axial and azimuthal temperature variations
- cladding tube properties (mechanical properties, structure and microstructure)
- superimposed influencing factors

2.2 Further investigations of the refill phase are planned on the effects of:

- low heat-up rates
- extended hold times
- steam atmosphere
- pre-oxidation

The test results obtained are to verify over an extended parameter range the present empirical correlations of the expansion and burst performance analysis, and to improve knowledge of the natural scattering of results.

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3. Research Program

Consistent with the key points, the program will be carried out in the following test groups. That this program does not overlap with the scope of tests planned under PNS (project nuclear safety) was confirmed with GfK.

3.1 Investigations of the reasons causing scattering of the expansion data

- influence of cladding
- influence of thermal eccentricity
- influence of cladding properties

3.2 Investigations on the "2 peak" (refilling) over an extended parameter scope

- with various heat-up rates during transient burst tests
- with various hold times during creep/burst tests
- with various, preceeding thermal conditioning of the cladding corresponding to the blowdown phase of a LOCA by means of transient- and creep/burst tests.

3.3 Investigations of steam influence

Construction, assembly and trial of a test equipment in order to perform internal pressure tests (transient- and creep/burst tests) with directly heated cladding samples in steam atmosphere.

Performance of transient and creep/burst tests to evaluate relationships during swelling and bursting of clad tubes in steam.

Evaluation of the influence of cladding tube pre-oxidation on the swelling- and burst performance of cladding tubes. Determination of correlations existing between burst temperature, burst strain, internal pressure, heat-up rate and hold time by means of transient- and creep/burst tests in steam.

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3.4 Investigations of combined Influences

Transient- and creep/burst tests are planned to be performed on indirectly heated samples in steam with superimposed individual effects, such as steam atmosphere, cladding tube geometry, thermal eccentricity and cladding tube properties in order to verify the combined influence of these individual effects on the swelling and burst performance of cladding tubes.

3.5 Analytical Work

The models for the strain and burst performance of the directly heated single rod will be checked by means of the test results and will be adjusted to the pertinent test conditions in order to achieve a more realistic description of the strain and burst performance.

4. Experimental Facilities

Tests in air will be performed in the same test equipment as was used for RS 107, though modified to meet the specific test requirements. The device consists of burst equipment, heating transformer, programmable temperature control:

electrical requirements:	500 A/60 V
pressure range:	100 - 150 bar
length of test specimens:	50 cm

Tests in steam will be performed in a test device consisting of steam generator heating, transformer, programmable temperature control and burst equipment.

steam generation:	5 kgs/h max.
steam temp.	600 °C max.
steam pressure:	5 bar max.
sample length:	40 cm

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5. Progress to Date

Construction and fabrication of a burst equipment to perform single rod internal pressure tests on Zry-4 cladding tubes with direct heating in steam atmosphere.

Trial and calibration of a temperature control unit in order to reproduce given temperature curves with directly heated single rod internal pressure tests automatically.

Sample fabrication for direct heated creep/burst test to be performed in air in order to evaluate influence of geometry (wall thickness variation).

Performance of creep/burst tests on Zry-4 cladding tubes in air with steps in the heat-up temperature curve.

6. Results

Max. strain is approx. 30 %. The axial strain profile approximates the axial temperature profile of the sample set at the beginning of hold time ($t_H = 0$), i.e. the strain profile shows an almost cosine-shaped curve on the overall sample length.

The axial temperature profile changes during hold time are as follows:

- with constant temp. in the vicinity of max. strain (axial position of temp. control by means of the central thermocouple, approx. 14 mm above the middle of the sample) a temperature increase was observed above and below the swelling,

- with constant temperature remote from max. strain (90 mm below the thermocouple in the middle) temperature decreases at position of max. circumferential strain. Temperature increases, however, below the control thermocouple (smaller circumferential strain).

1. 1. 77 - 31. 12. 77

7. Next Steps

Assembly and trial of the burst equipment in order to perform internal pressure tests on directly heated single rods.

Preparation of tests planned to determine influence of thermal eccentricity.

Preparation of pre-oxidized samples in 80 vol% Argon and 20 vol% O_2 at 420 ^{o}C.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.-31.12.1977	1.3	PNS 4235.1
Vorhaben/Project Title Untersuchungen zum mechanischen Verhalten von Zircaloy-Hüllrohrmaterial bei Störfalltransienten Investigations of the Mechanical Behavior of Zircaloy Cladding Material under Transient Conditions		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KfK Projekt Nukleare Sicherheit (PNS) IMF 2
Arbeitsbeginn/Initiated 01.07.1972	Arbeitsende/Completed 1979/80	Leiter des Vorhabens/Project Leader Dr. M. Bocek
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating December 1977	Bewilligte Mittel/Funds

1. General Aim

Investigation of the plastic behaviour of Zircaloy-4 (Zry-4) during different reactor incidents, especially LOCA-typical temperature- and stress-transients and in a LOCA-typical environment.

2. Particular Objectives

Determination of a mechanical equation of state, containing all the parameters which influence the plastic strain.

3. Research Program

3.1 Tensile testing and creep testing of Zry-4 at high temperatures.

3.1.1 Influence of texture on plastic properties.

3.1.2 Influence of grain structure and phase composition on plastic properties.

3.1.3 Influence of ZrO_2-coating on plastic properties.

3.2 Burst tests with Zry-4 cladding.

3.2.1 Isothermal tests.

3.2.2 Transient tests.

3.2.3 Combined experiments (integral tests).

3.3 Examination on irradiated material.

4. Experimental Facilities

Ad 3.1 Tensile testing is being performed in an INSTRON closed-loop machine.

Ad 3.2 For burst tests in vacuum tubes are pressurized in a radiant furnace. For integral experiments (steam environment) cladding with internal heaters will be used.

5. Progress to Date

Ad 3.1 a)- Stress-strain testing in vacuum have been finished.

- Isothermal creep tests have been performed in vacuum at temperatures between 600 and 1100°C. The uniaxial stress range $\sigma_{uni}(o)$ was 5.0 to 78 MPa what corresponds to an internal pressure P_i from 0.8 to 12.7 MPa for "BIBLIS A"-type cladding.

b)- Comparison of the temperature dependence of the stress-exponent n (from the rate equation $\dot{\varepsilon} = A\ (T)\ \sigma^n$) determined under different experimental conditions.

- Comparison of the temperature dependence of Q (from the rate equation $A(T) = A_o e^{-Q}/kT$) determined under different experimental conditions.

c)- Creep curve calculation on the basis of $\dot{\varepsilon} \sim \sigma^n$.

- Analysis of the tensile creep curves.

Ad 3.1.3 - Influence of ZrO_2-surface layers on the ductility of Zry-4 tensile specimens.

- Examination of the crack morphology of ZrO_2-layers.

Ad 3.2 An analysis of burst test results had been made.

6. Results

Ad 3.1 a)- In essential the stress-strain testing has been finished. In the report /1/ the results are compared for experiments conducted in air and vacuum repectively. The comparison is mainly based on the temperature and strain rate dependence of the yield stress, the stress exponent and the total elongation resp..

- The stress-rupture behaviour of isothermal tensile creep tests is compared with results from stress rupture tests performed on Zry-4 tubing. Using the relation between the hoop stress σ_{hoop} and unaxial stress σ_{uni}

$$\sigma_{hoop} = \frac{2}{\sqrt{3}}\ \sigma_{uni}$$

the agreement for tests at 800°C is very good. The stress rupture behaviour is influenced:

and i) by recrystallisation (for T = 600 and 700^{o}C resp.)
ii) by the environment (ZrO_2 - layer)
from the slope of the stress rupture curves it seems that the temperature dependence of the stress exponent n is less pronounced as that which follows from change in strain rate experiments.

b)- The n-values were determined in two ways.

i) From stress-strain curves: plotting log $\sigma_{0,2}$ vers. log $\dot{\varepsilon}_o$ ($\sigma_{0,2}$ is the 0,2 % proof stress and $\dot{\varepsilon}_o$ is the engineering strain rate) these are called $n_{\dot{\varepsilon}_o}$-values.

ii) From creep tests (constant load): plotting log σ vers. log $\dot{\varepsilon}_{min}$ (σ is the true stress for ε corresponding to the minimum creep rate $\dot{\varepsilon}_{min}$). These are called $n_{\dot{\varepsilon}_{min}}$-values. The dependence $n_{\dot{\varepsilon}_{min}}$(T) are compared for deformation in vacuum (pressure p < 10 nbar) and air (or less good vacuum p > 10 nbar) resp.

For tests performed in vacuum (p < 10 nbar) the $n_{\dot{\varepsilon}_{min}}$ values between 600 and 800^{o}C are less dependent on temperature than the $n_{\dot{\varepsilon}_o}$-values (6,5-5,5). In the two phase region both the values decrease (2,5). More over the $n_{\dot{\varepsilon}_o}$-values decrease in the α-phase region too. However in the ß-phase region $n_{\dot{\varepsilon}_o}$-values remain constant (3,0). For tests in air atmosphere $n_{\dot{\varepsilon}_o}$ (T) is rather complex due to an apparent stress dependence (which was not observed in vacuum tests). Up to now there are only few results from tests in less good vacuum (p > 10 nbar) which indicate that $n_{\dot{\varepsilon}_{min}}$(T) has a maximum at 700^{o}C.

- The Q-values were determined in two ways.

i) From plots: log $\dot{\varepsilon}_o$ vers. 1/T for $\sigma_{0,2}$ = const. (so called $Q_{\sigma_{0,2}}$-value).

ii) From plots: log $\dot{\varepsilon}_{min}$ vers. 1/T for σ = const. (so called Q_{σ}-value).

For tests performed in vacuum (p < 10 nbar) between 600 and 800^{o}C both the values are independent upon temperature (3.5 x 10^2 kJ/mol). In the two phase region irrespective upon test atmosphere the Q-values become temperature dependent in a complex manner. In the α-phase region the Q-values from tests in air are 0.8 x Q-value measured in vacuum. Between 1000 and 1100°C the Q-values ·r vacuum are independent upon temperature (1.5×10^2 k J/mol). The examinations mentioned above serve as a data bank for model calculations of cladding deformation.

c) - Principally the constitutiv equation

$$\dot{\varepsilon}_o = A\ (T)\ \sigma^n \Big|_{\phi_i} \qquad (1)$$

(where $\dot{\varepsilon}_o$ is the engineering strain rate and σ is the true stress resp.) can be easily integrated assuming that the structure parameters ϕ_i (e.g. grain diameter, phase composition etc.) remain unchanged during creep. For creep at constant load the strees σ increases with strain. Assuming constancy of volume creep it follows from eq (1)

$$\varepsilon_o = \frac{1 + \varepsilon_{o,p}}{\left[1 - t/_{t_R}\right]^{\frac{1}{n-1}}} - 1 \text{ for } t < t_R \qquad (2)$$

$\varepsilon_{o,p}$ is the "primary" creep and t_R is the time to rupture. For $\varepsilon_{o,p} << 1$ eq (1) can take the form

$$\dot{\varepsilon}_o = \frac{A\ (T)\ \sigma(o)^n}{\left[1 - \frac{t}{t_R}\right]^{\frac{1}{1-1/n}}} = \frac{\dot{\varepsilon}\ (o)}{\left[1 - \frac{t}{t_R}\right]^{\frac{1}{1-1/n}}} \text{ for } t < t_R \qquad (3)$$

for practical calculations an appropriate maximum value for $\dot{\varepsilon}_o$ can be chosen from the experiment and together with t_R the limiting (maximum) time $t_{max} < t_r$ is obtained from eq (3). This time t_{max} should limit the reliable range of comparison between experiment and calculation (e.g. instability of tensile deformation).

- For the development of a constitutive equation for plastic flow we need information about the work hardening as well as the recovery behaviour of zircaloy. Therefore tensile creep experiments at constant load were conduced in vacuum between 600 and 1100°C. From the analysis of creep curves it follows that:

i) In general the curves (in a plot log $\dot{\varepsilon}$ vers. ε) consist from two regions. The first creep stage is immediatelly followed by the accelerated creep stage. 3-stage creep curves (with a linear intermediate region) were observed only exeptionally. Three diffe-rent types of initial creep appeared. Normal-transient creep (decreasing $\dot{\varepsilon}$), anomalous transient creep (increasing $\dot{\varepsilon}$) and occasionally (as an intermediate type) a stage with $\dot{\varepsilon}$ = const.

ii) The tendency for anomalous creep increases with increasing temperature and load. Thus anomalous creep was predominatly observed in the α/β- and β-phase region resp. (no network hardening during the whole course of deformation).

iii) Applying the stability criterion for a tensile test: if $^{d\ln\dot{\varepsilon}}/d\varepsilon \leq 1$ then deformation is stable to the creep curves obtained, it follows, that the deformation is stable (no necking-occurs) only under conditions of normal transient creep.

Ad 3.1.3 - In the α-phase region ZrO_2-layers increase the ductility of tensile specimens. Principally the same behaviour is observed in burst tests on Zry cladding. Preliminar metallographic investigation about cavitation on tensile specimens have shown that cavitation is supressed even by the presence of very thin ($\sim$ 1 μm) ZrO_2 layer. According to that, the failure is delayed. This phenomenon is explained on the basis of the hydrostatic stress component associated with the ZrO_2 coating. This component compensates the axial stress component (applied stress) which is responsible for the cavitation.

- A method was developed for the investigation of the crack morphology a tensile test on ZrO_2/Zry-4 specimens. Measurements of crack density and crack width on preoxidized samples as a function of temperature, strain and strainrate are on the way. At 400°C blistering of the ZrO_2-layer occurs after low plastic strain (1-2 %). Whereas at 800°C the layer adheres to the metal up to the failure of the specimen (strain ≥ 50 %).

Ad 3.2 On the basis of plots: "maximum circumferential strain vers. burst temperature", the influence of heating modus, heating rate, atmosphere, inventory, constraints and preoxidation is visualized. The extend of "ballooning" is strongly dependent on testing procedure (heating modus: smaller circumferential strains for cladding internally heated as compared to ohmically heated cladding). There is a distinct influence of the heating rate (in the α-phase region larger ballons for the lower heating rate) and the "outer" and "inner" atmosphere resp. (ZrO_2-layer, influence of fission products). Generally the smallest ballons are observed on tubing which failed in the (α + ß)-phase region. In this range the strains observed seems to be less dependent on testing procedure.

7. Next Steps

Ad 3.1 - To close ISO-tensile creep test in vacuum ($p < 10$ nbar).
- To start with temperature transient tensile creep test in vacuum.

Ad 3.1.3 Examinations of the single-fibre composite will continue.

Ad 3.2 ISO-burst tests on short Zry cladding specimens (with continuos strain measurement) will continue.

8. Relation with other Projects

PNS 4235.2

PNS 4235.3

RS 107

9. References

M. Boček et al. in 2nd PNS-Sem.-Annual Report (German with English abstracts), KFK-2435, April 1977, S. 250.

/1/ M. Boček et al. in 1st. PNS-Sem.-Annual Report 1977 (German with English abstracts), KFK-2500, Dec. 1977.

10. Degree of Availiability of the Reports

Unrestricted distribution.

Berichtszeitraum/Period 01.01.-31.12.77	Klassifikation/Classification 1.3	Kennzeichen/Project Number PNS 4235.2
Vorhaben/Project Title Untersuchungen zur Hochtemperatur-Wasserdampf-Oxidation an Zircaloy-Hüllrohrmaterial Investigation of the High Temperature Steam Oxidation of Zircaloy Cladding Tubes		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit(PNS) IMF I
Arbeitsbeginn/Initiated 1973	Arbeitsende/Completed 1979/80	Leiter des Vorhabens/Project Leader Dr. S. Leistikow
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1977	Bewilligte Mittel/Funds

1. General Aim

Cladding Material Behaviour under Accident Conditions

2. Particular Objectives

Investigation of the High Temperature Steam Oxidation of Zircaloy Cladding Tubes.

3. Research Program

Study of Zircaloy 4/Steam Oxidation Kinetics and of Oxidation Related Change in Mechanical Properties.

4. Experimental Facilities

Experimental set-ups for isothermal and temperature-transient oxidation reactions. Facilities for isothermal and temperature-transient stress-rupture testing.

5. Progress to Date

Final isothermal and temperature-transient testing. Further post-test evaluation and documentation. Isothermal/isobaric creep-rupture testing of capsules in argon and steam (unoxidized and preoxidized condition). Temperature-transient/isobaric testing in steam according to LOCA-transients.

6. Results

Post-test evaluation of Zircaloy 4 specimens, oxidized in steam at temperature between 700 and 1000^{o}C, showed that not only oxygen take-up but also the different layer thicknesses can approximately be calculated on the basis of parabolic rate laws which were established recently for the range of high temperature (1000-1300^{o}C).

To determine the oxygen distribution across the oxidized wall of Zircaloy 4 tubing the post-test evaluation of isothermal steam oxidation experiments was performed in parallel by hardness measurements, microprobe analysis, and theoretical calculations based on physical properties and the Zr-O phase diagram. The comparison of the results gained in this way showed that the theoretical assumptions of oxygen distribution are in good agreement with the experimental findings.

Accidental oxidation of nearly equally preoxidized (400-800°C) tube specimens showed to be more dependent on maximum temperature of the following LOCA exposure than on temperature of preoxidation (oxide structure). At 1200°C there was no protection by preformed scales, at 1000° and 1100°C however, a considerable reduction of accidental oxygen consumption, hydrogen release, and heat generation was observed.

The isothermal/isobaric creep-rupture tests of one-sided closed Zircaloy 4 capsule under superimposed steam oxidation were continued at 1000-1300°C. Thus, the mechanical properties in argon and steam could be compared now over the total range of 800-1300°C. The experiments proved again the strengthening and ductility reducing effect of steam oxidation on Zircaloy 4. These effects could be recognized the earlier the higher the test temperature was chosen. For all experiments a nearly equal internal pressure threshold value of 7-8 was found below which no capsule ruptured, which indicates that - with increasing temperature - the decreasing strength of the material was overcompensated by the strengthening effect of steam oxidation. Far higher pressures had to be applied to rupture strongly oxidized capsules. The band of maximum circumferential elongations in steam differed clearly at all temperatures from that in argon. This effect was found to be strongly increased when preoxidized (2 hrs, 800°C) capsules were tested at 900-1300°C.

The scanning electron microscopic and metallographic examination of single specimen creep tests (900°C) showed that crack formation occured at 3-4 % elongation. The crack density increased with pressure, the crack width increased with time-to-rupture. Thus, higher internal pressure produced numerous cracks of medium width, lower internal pressure a smaller number, but broader cracks. By metallography and following planimetric analysis of creep-deformed and ruptured capsule surfaces, the extent of the creep-assisted surplus in oxidation was determined. The extent of oxidation in the α/β-transformation region (900 and 950°C) was at all times not higher than that at 1000°C in the unstressed condition.

First temperature-transient/isobaric creep-rupture experiments were performed in steam under inductive heating. At the beginning, a good agreement was found when isothermal/isobaric experiments were performed and compared to the results of our creep-rupture tests in the resistance heated furnace system. The following experiments were performed by specimen exposure to LOCA similar time-at-temperature conditions (950-750-800/1300°C). Burst-pressures and max. circumferential elongations were established for the whole range of temperatures. All specimens, being pressurized higher than 15-19 at, ruptured already after having reached the blowdown peak (950°C).

7. Next steps

Creep-rupture testing at temperatures < 900°C. Construction of facility for testing of fuel rod simulators (FABIOLA). Post-test evaluation of fuel rods and bundles.

8. Relations to other programs

All other of KFK/PNS, KWU, NRC.

9. References

S. Leistikow, R. Kraft
"Creep-Rupture Testing of Zircaloy 4 Tubing under Superimposed High Temperature Steam Oxidation at 900°C"
6th Europ. Congr. Metallic Corrosion, London 19-23.09.77, p. 577-84

10. Degree of Availibility of the Reports

Unrestricted distribution

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1977	1.3	PNS 4237
Vorhaben/Project Title Untersuchungen zum Brennstabverhalten in der 2. Aufheizphase eines Kühlmittelverluststörfalles - In-pile-Versuche mit Einzelstäben im DK-Loop des FR2 Investigations and Fuel Rod Behavior in the 2nd Heatup Phase of a LOCA. In-pile Experiments with Single Rods in the DK Loop of the FR2 Reactor		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS) RBT-IT
Arbeitsbeginn/Initiated July 1972	Arbeitsende/Completed 1980	Leiter des Vorhabens/Project Leader B.Räpple/E.Karb
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating December 1977	Bewilligte Mittel/Funds

1. General Aim

The in-pile experiments performed in the DK loop* of the FR2 reactor aim at investigating the influence of the "nuclear parameters" on the mechanism of fuel rod failure. The nuclear parameters include: thermal, mechanical and chemical behavior of the fuel and cladding, above all after irradiation, the presence and, if applicable, additional release of fission products during the transient, true nuclear heat generation.

The tests are performed with short length single rods; they concentrate on the second heatup phase of a loss-of-coolant accident (LOCA).

2. Particular Objectives

The investigations are performed under two individual tasks:

- PNS 4237.1: Tests with nuclear rods, variation of internal pressure and burnup: Nuclear tests
- PNS 4237.2: Comparative tests with electrically heated fuel rod simulators (BSS), variation of internal pressure: Reference tests.

3. Research Program

PNS 4237.1: It is planned to carry out 42 nuclear tests with fresh and preirradiated specimens. Steps of burnup: 0/ 2,500/ 5,000/

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* DK = Dampf-Kontamination = Steam-Contamination

10,000/ 20,000/ 35,000 MWd/t_U. Range of internal pressures: 45 to 100 bar.

PNS 4237.1: 25 non-nuclear BSS tests are planned. Range of internal pressures similar to 3.1

Both types of tests are performed in the DK loop of the FR2 reactor.

4. Experimental Facilities, Computer Codes

4.1 Test Facilities

4.1.1 Test Loop

The DK loop is operated with superheated steam of 60 bar, at 300 - 350 °C in the test section, and with a mass flow of 120 kg/h. By means of a quickly closing valve upstream of the test section the coolant flow can be interrupted abruptly thus initiating the heatup phase. When the maximum temperature is reached, the specimen power is reduced by reactor scram or interruption of the power supply respectively.

A special waste-gas system is available for the fission products escaping from the burst nuclear rod; it decontaminates the waste gases from volatile halogens and retards noble gases; solid and liquid isotopes are retained by filters.

4.1.2 Preirradiation and test rod assembling

The test rods are preirradiated in the FR2 up to the desired target burnup. For each burnup step six rods are assembled to form a preirradiation rig similar in its structure to that of an FR2 fuel element. For safety monitoring and determination of the degree of burnup each rig has been provided with the following instruments: One TC each at the D_2O inlet and D_2O outlet, a vanadium detector for neutron flux density measurement, a turbine type flow rate transducer, and a TC for D_2O temperature monitoring.

The measurement signals are processed in a separate measurement console. In this console the respective power is determined from ΔT and flow rate, and the flux density is determined from the detector signals and summed up to give the fluency. The burnup is obtained from this and the power history.

Specimen withdrawal from the preirradiation rig, rod carrier assembly and specimen instrumenting are carried out in the FR2 shielded cell. This requires special equipment, such as coupling and welding devices, 2 cutting devices, leak testing equipment, handling and transport devices.

4.1.3 Post-Irradiation Examination PIE

Upon completion of the test the specimen is subject to examination in the neutron radiography facility (NERA) of the FR2. The radiographs serve as a preliminary information about the position and shape of balloonings at the fuel rod etc.

In the FR2 shielded cell the fuel rod is separated from the specimen carrier and transported to the Hot Cells for post-irradiation examination. Here, instruments are available for

- specimen photography,
- dimension measurement,
- γ-scanning,
- X-ray testing,
- metallographic examination,
- sample fabrication for the radiochemical burnup determination.

4.2 Computer Programs

For precalculation of the tests the following programs are available at IKE Stuttgart (see cooperation, point 8):

- RELAP 3 (Thermohydraulic program)
- WALHYD 2D (heat conduction program taking into account the radiation).

The results of this task are to be used also to verify the SSYST program system (see PNS 4231).

1.1. - 31.12.1977

5, Progress to Date

5.1 The experimental program was continued according to schedule:

- Six tests with unirradiated specimens completed the experiment series B.
- A modified fuel rod simulator (BSS 07) was exposed to a number of transients. Objectives: Testing of BSS design and loop hardware modifications, additional investigations on accuracy of temperature measurement, comparison runs with steam bypass.
- With 5 tests of the F series (20,000 MWd/t_U) transient testing of preirradiated rods was started.

5.2 The remotely operated devices for hot cell assembling of preirradiated test rods, including the facility for TC welding, were completed, tested and successfully used with five F rods.

5.3 Preirradiation of two rigs with six rods each was continued towards a target burnup of 35,000 MWd/t_U. Preirradiation of the rods of the F series was terminated after they reached the desired burnup of 20,000 MWd/t_U.

5.4 Comparison calculations were continued at IKE Stuttgart using data obtained in the BSS 05 test series.

5.5 Post-irradiation examination of rods continued. Nondestructive examination and sectioning for metallographic investigation was completed with 8 rods out of the total of 14 rods transferred to the Hot Cells.

6. Results

6.1 Burst temperatures and burst pressures of all B tests lie well within the scattering band of the failure data obtained in out-of-pile tests performed by other workers. For the specimens already examined in the Hot Cells a similar statement can be made with respect to maximum circumferential elongation.

6.2 Burst-data of the F testswith preirradiated rods fit exactly to those obtained with fresh fuel (B tests). Neutron radiography shows that the fuel pellets, already heavily cracked during previous steady-state irradiation, disintegrated during transient tests into small particles at those rod sections where major clad lifting or ballooning occured.

6.3 The cladding temperature correction of 75 K as determined earlier for a linear rod power rate of 50 W/cm (electrical) was confirmed by the tests with BSS 07. The scattering band amounted to only ± 15 K with these tests as compared to the ± 35 K used up to now. The circumferential temperature variation at the undeformed simulator was found to be less than 10 K.

6.4 The minimum required mass flow rate via the steam bypass had been determined by oxidation kinetics evaluation to 0,02 kg/h. This rather low rate could not be realized technically, the effective rate amounted to 0,3 - 0,5 kg/h. As this flow had a pronounced influence on the cladding heatup rate during the transient, operation of the bypass was discontinued.

6.5 A modification of the gap heat transfer model in the WALHYD-2D code resulted in an improved agreement between calculated and measured clad temperature histories for BSS. The new model uses a lower heat transfer coefficient in the annular gaps during steady-state and during the first seconds of the transient phase followed by a step increase of this quantity simulating the gap width reduction by differential thermal expansion during the heatup phase.

6.6 An internal report was distributed on the methods and results of dimensional characterization of the cladding tubes used for fabrication of the nuclear rods (9.3).

The data of test # A 1.1 and the results of the non destructive post-test examinations were compiled in a preliminary test result report (9.4).

1.1. - 31.12.1977

7. Plans for the near future (1978)

7.1 Assembling and testing of preirradiated samples will be continued with five G.1 rods in the first quarter, two G.2 and three G.3 rods later in the year, all with a target burnup of 35,000 MWd/t_U.

7.2 A BSS test series will be performed to test a modified TC attachment and to further develop the BSS design.

7.3 Hot Cell post-test examination and evaluation of the results will be continued with increased efforts.

8. Relation with Other Projects

The thermodynamic precalculations and verifications of events taking place in the in-pile test section are performed together with IKE Stuttgart.

The task makes part of the PNS 4230 program and thus it is closely linked to the PNS 4235, 4236, 4238 and 4239 tasks tackled by IMF, IRE and IRB.

The metallurgical evaluation of the test rods is performed jointly with IMF.

IRB develops and provides the heater rods for the fuel rod simulators.

9. References

Publications:

9.1 Karb, E.; Sepold, L.: In-pile-Experimente zur Untersuchung des Brennstabversagens, KFK 2102 (1974) S. 94 - 112

9.2 E.Karb, G.Harbauer, W.Legner, L.Sepold, K.Wagner: "Theoretische und experimentelle Untersuchungen zur Gasströmung in LWR-Brennstäben bei Kühlmittelverluststörfällen", KFK 2411, December 1976

1.1. - 31.12.1977

Reports:

9.3 Sepold, L., Prüßmann, M.: Vermessung von frischen DWR-Brennstabhüllen. (1977) (unpublished)

9.4 Karb, E., Harbauer, G., Prüßmann, M.: Teilbericht über Nuklearversuch A 1.1. (1977) (unpublished)

10. Degree of Availability of the Reports

9.1, 9.2 Unrestricted distribution

9.3, 9.4 internal distribution only

Title 1 (Original language)	Classification
Valutazione del comportamento del combustibile di LWR in condizioni di LOCA e PCM	1.3
Title 2 (English)	**Country** ITALY
Investigation LWR fuel behaviour under LOCA and PCM conditions	**Sponsor** Italian Government **Organisation** C.C.R. Ispra
Date initiated June 1975 Date completed 1981 Last updating April 1977	**Project Leader** C.C.R. Ispra - Essor Division CNEN Italy - Thermal Reactor Department

1. General aim

In pile investigation of fuel behaviour (pins and bundles) in operating conditions representative of malfunctioning in Water Reactors, by an experimental loop named SARA to be built in the ESSOR reactor.

2. Particular objectives

2.1. Assessment of the loop performances

2.2. Production of experimental data on in-pile fuel behaviour

2.3. Verification of the computer codes for accident analysis

3. Experimental facilities and programme

Besides the operation in steady conditions for both the boiling and the pressurized mode, the SARA loop can attain a wide variety of abnormal conditions by suitable transient control actions on loop system pressure and flow distribution. The thermohydraulic performance of the loop to a good precision is insured by carrying out tests in an identical electrically heated test section.
The loop is designed to insure operation in a wide parameter range for each accident type and can accomodate bundle size up to 16 pins with reentrant test section (650 kW) and 36 PWR pins in the once through test section (1.5 MW); the active height is in both cases 1.5 m, which seems to be enough for the analysis of the reflood case of a LOCA. In the SARA loop fuel failure propagation can be investigated, an item which is expected to be of great importance at the time when the loop has become operating.

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Title 1 (Original language)	Classification
Valutazione del comportamento del combustibile di LWR in condizioni di LOCA e PCM	1.3

4. Project status

4.1. Conceptual design completed; detailed design running

4.2. Definition of the test matrix under-way

4.3. Acquisition of existing codes (in particular FRAP-T and RELAP completed) and updating under way; development of specific submodels carried on; parametric and sensitivity analysis being completed to support item 4.2.

5. Next steps

Mid '77 Safety authority licence to proceed to construction

PROJECT TITLE : Transient boiling heat transfer in emergency core cooling conditions	LWR 1.3
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : JRC - Ispra
DATE INITIATED : 1974 DATE COMPLETED : 1976	PROJECT LEADER : W.Hufschmidt/H. Lauer

Description :

1. General aim

Investigation and visualization of transient boiling conditions.

2. Particular objectives

To study the transient boiling conditions in the pressure range of 1-5 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 : 1-5 bar
- cooling water temperature : 20-150°C
- initial surface temperature : 200-800°C

4. Project status

4.1 Progress to date

The study has been completed at the end of 1976.

5. Next steps

Evaluation of the experimental data and preparation of the final report.

6. Relation with other projects

The programme has been planned so as to be complementary to other work in the quenching field.

7. Reference documents

LAUER H.
Untersuchung des Wärmeübergangs und der Wiederbenutzung beim Abkühlen heisser Metallkörper.
EUR. 5702.d

8. Degree of availability

Freely available

9. Budget

The expected total investment from the CEC is 65.000 ua which includes the cost of the facility and the running costs.

2. CORE MELTDOWN

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1977	2.1	PNS 4321 (4241)

Vorhaben/Project Title	
Experimentelle Untersuchung der Abschmelzphase von UO_2-Zircaloy-Brennelementen bei versagender Notkühlung Experimental Investigations of the Meltdown Phase of UO_2-Zircaloy Fuel Rods under Conditions of Failure of Emergency Core Cooling	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Gesellschaft für Kernforschung Projekt Nukleare Sicherheit, RBT/IT

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1973	1978	Dr. S. Hagen
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1977	

1. General aim

Investigations of the course of the melting process including the resolidification of the melt on the colder parts of single rods, single rods with spacers, bundles and bundles with ballooned cans. The influence on the melting process from different parameters like surrounding atmosphere and temperature gradients will be investigated.

2. Particular objectives

Meltdown Behaviour of bundles in non-oxidizing and oxidizing atmosphere.

3. + 4. Researchprogramm and experimental facilities

The research programm and the experimental facilities are given in earlier reports.

5. Progress to date

In the reporting period we have done experiments on bundles in non oxidizing (He) and oxidizing atmosphere (He/20 % O_2; steam). The bundles consisted of an central arrangement of 3 x 3 fuel rod simulators surrounded by a double circle of 36 reflector rods or a fibre ceramic reflector of ZrO_2. We have done experiments with lack of steam. The experiments with steam in excess were performed with different heating rates (0,25 - 4 ^{o}C/sec). Some experiments were carried out with spacers. The central rod of the bundle was used as a heated rod or was set up of solid pellets.

6. Essential results

The bundle experiments should give us information on the movement and the refreezing of the molten material. The experiments show, that the melt tends to refreeze between the rods, rather then to drop down. This tendency is more pronounced, the more the rods are oxidized.

In He, when large amounds of melt are produced, the molten material tends to solidify on the bottom electrode as a lump. The vertical range of this lump is larger then the horizontal dimensions. The melt refreezing between the outer rods of the bundle formes "walls". In steam, the lump is formed much higher in bundle at the lower end of the hotter middle region - in axial direction - with out dropping down remarcable amounts of melt to the bottom electrode.

If we try to translate these results to the behaviour in the core, it is imaginable that there may be the formation of a solidified bottom layer above the risidual water of a core flooded in the beginning.

The bundle experiments in oxidizing atmosphere with lack of oxygen show that there exists a strong axial dependency of the oxidation. During streaming up along the rod the oxygen is consumed. The oxidation begins on the lower end and limits itself to a region which corresponds to the total amount of oxygen available.

So in a bundle experiment only the lower half of the central rod was oxidized so much that melting of the can was avoided when the melting temperature of Zircaloy was axceeded. For this experiment only half of the necessary oxygen was available.

Additional supply of oxygen on one side of the central rod enlarged the oxidized region only on this side. From this we see that we can have the formation of quite different oxide layers close together, when only the oxygen content of atmosphere passing by is varying analogously.

In the experiments on central rods with solid pellets and fibre ceramic isolation we have reached temperatures over 2000 °C for the central rod. These experiments confirm that the melting attack of the Zircaloy on the UO_2 is the same for solid pellets as for the ring pellets in the heated fuel rod simulator.

Concerning the stability of the rods with solid pellets the experiments gave the following results. For all heating rates (0,25 to 4 °C/sec) as well in the shorter bundles (30 cm) as in the longer bundles (50 cm) the pellet collum stayed intact during heat up. For the shorter bundles we had the undestroyed pellet collum also during the cool down, in contrary to the longer solid pellet rods, which broke down at the end of this period.

In the bundle with a spacer at the central rod the break off of the rod toke place in the region of the spacer.

Although we have the protection of the can from the attack of the spacer material by the oxidation process, this shows, that the region of the spacer is the weakest part of the rod.

7. Plans for the near future

In the next reporting period the bundle experiments are continued. In particular we want to see the influence of the conditions in the emergency core cooling region on the beginning of core melting. We have planed experiments with ballooned cans.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.-31.12.1977	2.1	RS 205
Vorhaben/Project Title Nachrechnung von Stabexperimenten und Absicherung von MELSIM Calculation of Fuel Pin Meltdown Experiments and Application to MELSIM		**Land/Country** FRG **Fördernde Institution/Sponsor** BMFT **Auftragnehmer/Contractor** Universität Stuttgart Institut für Kernenergetik u.Energiesysteme
Arbeitsbeginn/Initiated June 1, 1976	**Arbeitsende/Completed** November 30, 1978	**Leiter des Vorhabens/Project Leader** Prof.Dr.H.Unger/Dr.Schmidt
Stand der Arbeiten/Status continuing	**Berichtsdatum/Last Updating** December 31, 1977	**Bewilligte Mittel/Funds** 346.900,-- DM

1. General Aim

Within the frame of the research project RS 73 of the BMFT the program system MELSIM has been developed in order to describe the meltdown of a light water reactor core. In parallel, experiments on the meltdown behavior of fuel pins are performed within the "Projekt Nukleare Sicherheit" (PNS) at the "Kernforschungszentrum Karlsruhe" (GfK).

The present research project is aimed at the calculational verification of the results obtained at the GfK. An experimentally verified model of the meltdown process is to be developed later in order to describe the integral course of the core meltdown. This is of importance for the slumping modul of MELSIM.

2. Particular Objectives

The computer model is aimed at the solution of the following problems

- Fuel pin heatup in oxidizing (H_2O, air), reducing (H_2) and inert atmosphere
- Oxidization of the clad (zirconium) dependent on a differing oxigen supply, H_2-generation
- Influence of the interaction between UO_2 and molten zircaloy on the fuel pin behavior
- Pin failure depending on the radiation history
- Meltdown process in bundles of fuel pins
- Influence of the tungsten heater on the results of the single pin experiment

- Formation of channel blockage
- Geometric interactions during the slumping process

3. Research Program

3.1 Calculation of the Pin-Heatup Until Clad and Fuel Interact

The pin-heatup has to be simulated according to the different experimental conditions in different atmospheres. Temperature distribution, Zr-H_2O-reaction including heat production, formation of a zirconium oxide layer and the formation of H_2 has to be taken into account.

3.2 Simulation of the Pin-Behavior up to Clad-Melting-Temperatures

The interaction between Zr and UO_2 has to be described by a model.

3.3 Simulation of the Pin-Failure and the Meltdown Process

Pressure differences between the interior of the pin and the coolant have to be calculated and the behavior of the molten material has to be described.

3.4 Supporting Calculations with Respect to Rod-Bundle-Experiments

Bundle experiments are carried out to support the development of MELSIM. Therefore these experiments have to be calculated and the results of theory and experiments have to be analyzed. They have to be investigated with respect to consequences for the development of MELSIM's slumping modul and the modelling of the meltdown of the reference reactor.

4. Experimental Facilities, Computer Codes

No experiments are carried out within this project, the necessary experimental information is supplied by GfK (PNS 4240). Computer codes developed at IKE, modelling fuel pin behavior (e.g. STT, ZET-1D, ZET-2D, WUEZ) as well as the modular program system MELSIM are employed to the extend necessary. The experimental and theoretical knowledge is used to develop an improved computer code.

5. Progress to Date

to 3.1 The model for the calculation of the pin-heatup has been finished.

to 3.2 The interaction between Zr and UO_2 was modeled in a special melting model. Melting of Zr, Zr-UO_2-mixtures or UO_2 may be described.

to 3.3 The pin-failure has been simulated in a special modul which uses statistical failure criteria. The experiments proofed that a detailed description of the behavior of the molten material is not necessary for single pin experiments.

to 3.4 Calculations were performed for the first bundle experiments. Questions to further experiments were formulated.

6. Results

to 3.1 -3.3 The single pin model is finished. A good agreement between calculations and various measured parameters like power, temperature, oxidation, rod dimensions, and the composition of the molten material has been achieved.

7. Next Steps

to 3.1 -3.3 A report on the single pin model is in preparation.

to 3.4 A detailed program for the modeling of rod bundles will be worked out. Design calculations for further experiments are in preparation. The meltdown of pins in a bundle will be modeled.

8. Relations with Other Projects

The experimental data for the performed and intended investigations are provided by GfK (Research Project PNS 4240). There is a strong dependence on the experimental program. In return, the experimental program of GfK will be supported by the theoretical investigations.

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.-31.12.1977	2.1	RS 211

Vorhaben/Project Title	
Analyse der Zwischenphase Kernversagen-Schmelzsee und Integration von MELSIM in BILANZ zur Berechnung der Energiebilanzen nach hypothetischem RDB-Versagen Investigation of the Phase Between Failure of the Core and the Assembling of the Molten Material in the Pressure Vessel; Integration of the Program MELSIM into BILANZ in Order to Calculate the Energy Balances	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Universität Stuttgart Institut für Kernenergetik u.Energiesysteme

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
July 1, 1976	March 31, 1978	Prof.Dr.Unger/DP Körber
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	December 31, 1977	177.500,-- DM

1. General Aim

The physical behavior of a reactor core under hypothetical core meltdown conditions is to be investigated starting after the failure of the lower pin-supporting structure of the core and ending if the core is molten and assembled on the lower plenum of the pressure vessel. The vaporization process of the water contained in the lower plenum will be calculated as well as the heatup of the dry reactor vessel assuming different configurations of the core debris which fall into the plenum. A computer program (LÜCKE) will be developed. In order to obtain improved information on the heat balances involved in core meltdown, this program will be integrated into the energy balance program BILANZ I (PWR) and BILANZ II (BWR) of KWU together with the computer code MELSIM-1.

2. Particular Objectives

This project is closely connected to the research project RS 183 (energy balances after hypothetical pressure vessel failure) of KWU. Within this framework, KWU and IKE will couple the program systems listed above.

The particular objectives are as follows

- Investigation of the core meltdown accident by means of MELSIM-1 until the core falls into the lower plenum of the pressure vessel. Integration of MELSIM-1 in BILANZ I and II.

- Investigation of the behavior of the remaining core after partial failure of the lower core supporting structure.

- Modelling of the sequences of events in the reactor pressure vessel until the molten core is assembled in the lower plenum of the pressure vessel.
- Completion of the computer program LÜCKE which simulates these events.
- Coupling of LÜCKE into the program system BILANZ - MELSIM-1 and investigation of the entire accident sequence.

3. Research Program

3.1 Search and processing of the PWR and BWR data required for computer calculations which are carried out with MELSIM-1

3.2 Integration of MELSIM-1 in BILANZ
- Coupling of MELSIM-1 and BILANZ
- Heatup and slumping of the remaining core after a first partical failure of the pin-supporting structure.

3.3 Development of a simple model to describe the sequence of events from the failure of the core supporting structure until the formation of core melt at the bottom of the pressure vessel
- Analysis of the accident sequence
- Development of the computer code LÜCKE
- Performance of the calculations

3.4 Integration of LÜCKE into BILANZ

3.5 Simulation of the accident in applicating the complete model to a standard PWR- ord BWR-type and evaluation of the results.

4. Experimental Facilities, Computer Codes

Within the research project, the computer program LÜCKE is developed. It will be integrated together with MELSIM into the code BILANZ of KWU.

5. Progress to Date

to 3.1 Relevant PWR and BWR data have been collected and computer calculations have been performed.

to 3.2 MELSIM-1 has been integrated into BILANZ. A coupled computer run has been carried out.

to 3.3 The investigation and analysis of the accident sequence has been continued. The moduls for the program system LÜCKE have been developed and the system has been tested under various problem configurations.

to 3.4 and 3.5 The integration of LÜCKE into BILANZ is underway.

6. Results

The first phase of a core melt accident is able to be simulated for PWR and BWR using MELSIM-1. It can be seen, that the time from the beginning of the accident to the first core destruction is 2 to 3 times longer in BWR compared with PWR. Other time constants are in the same magnitude. In case of an initially dry core (water level in the vessel at the lower core edge) only 10% of the clad metal are oxidized to ZrO_2. A computer run using the coupled MELSIM-1 and BILANZ has been carried out successfully.

7. Next Steps

to 3.3 Testing of the program system LÜCKE will be continued and terminated. Then the second phase of the accident will be calculated.

A final report will be prepared.

8. Relation with Other Projects

There is a strong dependence to the project RS 73 (Development of the Computer Code MELSIM) and a close coupling to the investigation program RS 183 (Energy Balances after Hypothetical Failure of the Reactor Vessel). Further on this project is connected to the RS 316 program (Develpment of a Core Melting System on the Basis of RSYST).

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period 1.1. - 31. 12. 1977	Klassifikation/Classification 2.1	Kennzeichen/Project Number RS 200 /PNS 4245
Vorhaben/Project Title Technology and Properties of Corium Technologie und Eigenschaften von Coreschmelzen		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor PNS-IMF I KfK
Arbeitsbeginn/Initiated 1. 2. 1976	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader Dr. G. Ondracek/Dr. S. Nazaré
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1977	Bewilligte Mittel/Funds RS 100.000,--

1. General aim: Production and properties of corium.

2. Special aim: Development of a producing "recipe" to get corium EX1 including its quality characterisation.

3. Experimental programme: - Induction melting of core components to obtain corium EX1 using inert atmosphere.
   - homogenisation of the four-phase material by crashing and hot pressing.
   - chemical as well as microstructural analysis of the corium EX1 samples.

4. Experimental equipments: Induction melting apparatus; chemical and microstructural analysis equipments.

5. Work performed: 1.5 kg corium EX1 has been produced and characterized. The procedure works reproducable and is described in a final report.

6. Results: The production "recipe" has been developed and checked by producing a stored amount of corium EX1; homogeneous samples are available to perform property measurements and compatibility tests with concrete.

7. Planned futural work: The technological development work is full filled; property measurements are on the way.

8. Related projects: PNS 4244

9. References: KfK 2435 (1977) 432
   Nuclear Technology 32 (1977) 239
   KfK 2375 (1976) 440

   Final Report BMFT-RS 200

10. Degree of Availability:

Kernforschungszentrum Karlsruhe

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01. - 31.12.1977	2.1	RS 166

Vorhaben/Project Title	
Theoretische und experimentelle Untersuchung des Verhaltens eines geschmolzenen Kerns im Reaktorbehälter und auf dem Betonfundament Theoretical and Experimental Investigation of the Thermohydraulic Behaviour of a Molten Core in the Reactor Vessel and on the Concrete of the Basement	Land/Country: FRG Férdernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Institut für Verfahrenstechnik der T.U. Hannover Callinstr. 36 3000 Hannover 1

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
May 1975	December 1978	Prof.Dr.-Ing. F. Mayinger
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1977	DM 1.371.600,-

1. General aim

The aim of this research project is to investigate the thermal interaction of the molten core with various components of the reactor. Computer codes are established, which describe the attack of the melt to several reactor components. In order to test the reliability of the codes and to get necessary empirical informations for the codes, a number of experiments with modeling fluids are performed.

2. Particular objectives

The investigation of the spread out of the molten core material on the concrete of the reactor containment and the penetration of the concrete by the molten material is of particular interest. In order to get this information, the heat transfer between the melt and the concrete has to be determined by considering the behaviour of the molten concrete and the vapour and gas escaping from the concrete. Using this heat transfer relations, the penetration of the melt into the concrete and the formation of a cavern can be predicted.

Another object of the research project is the calculation of the transient interaction between molten core material and the wall of the reactor pressure vessel, i.e. the instationary thermal penetration of the vessel wall by the hot melt.
Furtheron the heat transfer and the crust formation at the top of a layer of melt cooled by boiling water from above is of particular interest. This case is important to determine the consequences of a subsequent flow-in of water on the top of a pool of molten core material.

3. Research program

3.1 Investigation of the heat transfer from an internal heated fluid to a gas liberating wall by the help of modeling experiments.

3.2 Numerical calculation of the thermohydraulic behaviour near growing and rising gas bubbles in an internal heated fluid.

3.3 Numerical computation of the formation of a cavern into concrete by molten core material. In the calculations the heat conduction and the gas liberation in the concrete must be taken into account. The heat transfer coefficients determined in 3.1 and 3.2 shall be used.

3.4 Extension of an existing thermohydraulic code by a modul calculating the thermal penetration of the reactor pressure wall simultaneous with the transient thermohydraulic behaviour in the melt.

3.5 Experimental investigation of the crust formation at the top of an internal heated pool of liquid, cooled from above by a boiling fluid. The solidification temperature of the heated liquid is lower than the Leidenfrost-temperature in the boiling fluid. Additionally, the heat transfer from the lower fluid to the upper one will be determined quantitatively.

4. Experimental facilities and computer codes

To 3.1 The experimental investigation of the heat transfer between a gas liberating wall and an internal heated fluid was performed by the help of the holographic interferometry. Additionally, the velocity in the fluid round the growing and rising gas bubbles was measured with a Laser-Doppler-Anemometer. The principal design of the test chamber used in the experiments is shown in figure. 1.

A rectangular cavity contains as test fluid water. At the lower boundary a number of tubes penetrates the bottom to inject air into the water. The side walls of the test cavity serve as electrodes to allow the electrical current passing through the water and such generating volumetrical heat sources. The bottom cooling channel can easily be removed to examine another arrangement of air injection tubes.

To 3.2 To calculate the thermohydraulic behaviour round the gas bubbles formed at the interaction front between melt and concrete the computer code BETON was established. The basis of this code is the numerical solution of the conservation laws for mass, momentum on energy by finite differences. It can handle two seperated phases including the effect of their surface tension.

To 3.3 The penetration of the molten core material into concrete can be determined by the code BETSI. This code computes the local melting rate of the concrete using an empirical correlation for the heat transfer between melt and concrete. The variation of the mean temperature of the melt is calculated by an energy balance for the melt. Additionally the code determines the shape of the cavern molten into the concrete.

To 3.4 The code THEKAR is extended to enable it to calculate the penetration of the pressure wall simultaneous to the computation of the thermohydraulic behaviour in the molten material.
At each time step an energy balance considering the local heat flux from the melt to the pressure vessel wall and the heat stored and conducted in the wall yields melting velocity of the wall. The aim of this extended code THEKAR is to determine the time necessary for the penetration of the vessel wall and the thermohydraulic behaviour in the melt at the moment of the vessel failure, considering the fact,that in this moment the melt is still in an instationary state.

5. Progress to date

To 3.1 A large number of experiments are performed with various gas flow ratios and several inclination angles of the gas liberating interface . These experiments were all done with the same arrangement of air injection tubes and with the model fluid water.

To 3.2 The code BETON was completed and a number of calculations were done with various fluid combinations and several initial and boundary conditions.

To 3.3 The computer code BETSI is established and almost completely tested, and parameter calculations are performed.

To 3.4 The extension of the code THEKAR K is completed and tested and this extended code was used to calculate some typical examples.

To 3.5 This part of the research project was started in the last quarter of the past year. Up to now the construction of the test facility and the theoretical preparation of the experiments are partly done.

6. Results

To 3.1 The mean Nu-numbers, describing the heat transfer between an internal heated fluid and a gas liberating wall, are plotted in fig. 2 as a function of the Re-number. This figure shows the values measured at a horizontal wall. Measurements for inclined areas are already performed, but not yet evaluated completely. The measured Nu-numbers can be correlated by the equation.

$$Nu = 3.2 \cdot Re^{0.5} \cdot Pr^{0.42} \cdot (F_g(1-F_g))^{0.5} \qquad (1)$$

In equation (1) the term describing the effect of the Pr-number and the bubble density factor Fg was not measured, but a result of theoretical considerations.

To 3.2 A typical example of the isotherms predicted by the code BETON is shown in figure 3 under the condition of just growing gas bubbles. Furtheron the figure shows the local heat flux to the bottom. Between the gas bubbles the heat flux density is high, but beneath the bubbles where the relatively cold gas covers the wall the local heat flux is nearly zero. Another example of results computed with BETON is shown in fig. 4. In this example, the heavier and internal heated fluid lies at the bottom of the considerated area covered by a layer of lighter fluid without internal heat sources. It can be observed that in each layer seperate convection patterns occure.

To 3.3 An example of the calculation done with the code BETSI is shown in fig. 5 In this picture the shape of the cavern molten into the concrete by the liquid hot core material is plotted for several time step. In these calculations equation (1) was used to determine the heat flux between melt and concrete.

To 3.4 The calculations done with the extended version of the code THEKAR gave the result that the pressure vessel is already penetrated before the thermohydraulic conditions in the melt are stationary. The mean consequence of this result is, that the maximum temperature in the melt may be considerably lower than calculated for stationary conditions.
The typical distribution of the local heat flux, known from the stationary calculation, can be seen under transient condition at a very early moment, too.

7. Next steps

To 3.1 The experimental investigation of the heat transfer between an internal heated fluid and a gas liberating wall shall be continued with several test fluids and different arrangements of the air injection tubes.

To 3.2 With the code BETON some calculations are planned to get more informations about the heat transfer between two stratified fluid layers.

To 3.5 During the next months the test facility for the experimental investigation of the crust formation shall be completed, and the performance of the experiments will start.

8. Relation with other projects

RS 183
Energy balance after a hypothetical reactor pressure vessel failure under consideration of the concrete destruction.
KWU Erlangen 01.09.75 - 31.05.77

9. Reference documents

Quarterly reports in the series: GRS Forschungsberichte (German)

Report period:	Jan. 1977 - Mar. 1977	GRS F - 41
	Apr. 1977 - Jun. 1977	GRS F - 43
	Jul. 1977 - Sep. 1977	GRS F - 44
	Oct. 1977 - Dec. 1977	GRS F - 45

Annular report 1976. (German)

10. Degree of availibility

The reports are available at the GRS, Cologne.

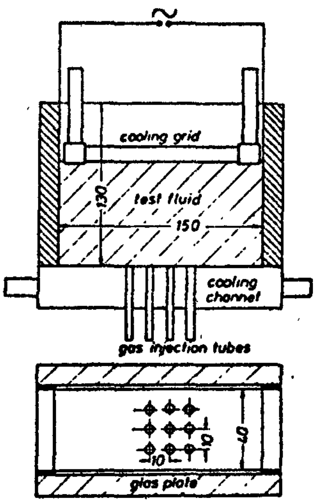

Fig. 1: Test chamber for the investigations of bubbles

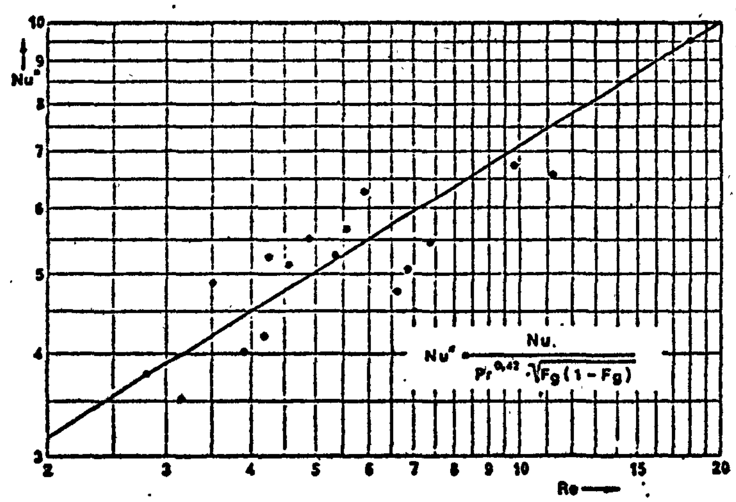

Fig. 2: Mean Nu-number at a gas-liberating wall

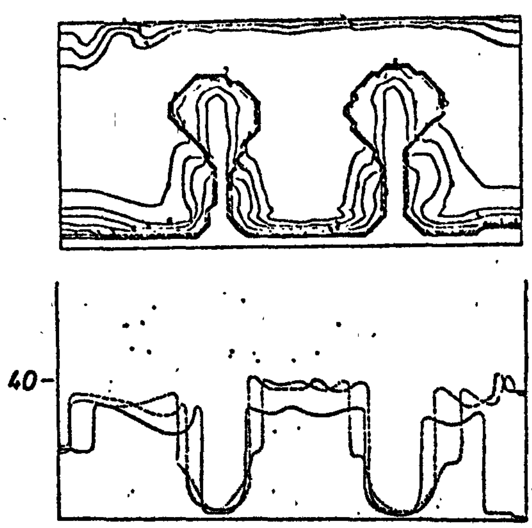

Fig. 3: Calculated isotherms and Nu-numbers with rising bubbles

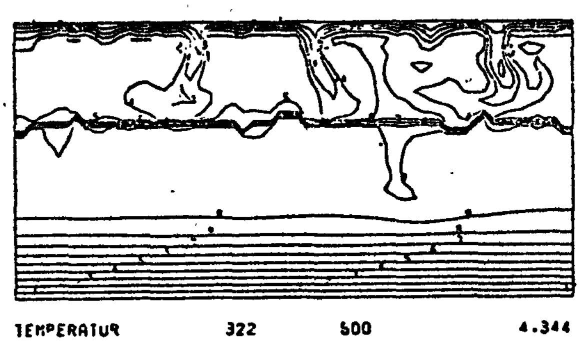

Fig. 4: Calculated isotherms in two layers upper layer not heated ($Ra_u = 2 \cdot 10^8$, $Ra_o = 5 \cdot 10^5$)

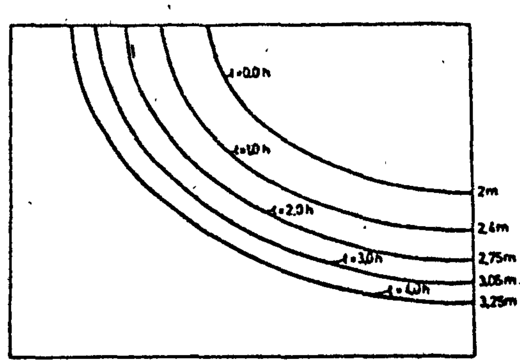

Fig. 5: Penetration of the melting front into concrete with time

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
Jan.1, - Nov.30,1977	2.1	RS 214

Vorhaben/Project Title	Land/Country / Fördernde Institution/Sponsor / Auftragnehmer/Contractor
Messung der Hochtemperaturviskosität von ausgewählten Substanzen, die im Zusammenhang mit einem möglichen Kernschmelzunfall wichtig sind Measurement of the High-Temperature Viscosity of Selected Substances Which are Important in Connection With a Possible Core-Melting Accident	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Battelle-Institut E.V. Frankfurt am Main Metals and Composites Division

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
June 1, 1976	Nov.30,1977	R.Skoutajan
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
completed	Nov.30,1977	305.810,--

1. General Aim

At the present state of progress achieved in the research project Core Meltdown, the viscosity of the melts that may form during a hypothetical core meltdown accident is one of the factors determining the course of such an accident. The investigations were focused on the oxidic mixtures of core melt and concrete melt that may form during the time-determining fourth accident phase termed "concrete penetration". The viscosity of molten concrete can be estimated on a theoretical basis from the composition of the melt, however, the additional components originating at the reactor inventory change the viscosity in a still unknown way. An experimental determination is, therefore, indispensable. A high-temperature rotary viscometer will be used.

2. Particular Objectives

Determination of the compatibility of selected crucible materials with oxidic corium-concrete mixtures; preparatory work necessary for the experimental determination of the viscosity of oxidic corium-concrete mixtures.

3. Research Program

3.1. Identification of a material for those components of the viscometer which during a measurement will come into contact with the oxide melt.

3.2. Preparation of design specifications for constructing the measuring systems and for assembling the viscometer.

3.3. Preparation of a working program for performing the measurements.

4. <u>Experimental Facilities</u>

An experimental setup containing parts of the high-temperature viscometer constructed under Project RS 71 was used for the compatibility measurements.

5. <u>Progress to Date</u>

Ad 3.1. Tungsten, tantalum, boron nitride and thorium oxide were taken into consideration. The two last mentioned materials could be excluded from experimental work according to results obtained at Euratom-Ispra /1/ and GfK /2/. Compatibility measurements were performed by melting a sample in a crucible with planeparallel ground bottom and by determining the thickness decrease of the bottom at a polished cross-section. In addition, the solidified melt was analyzed chemically and metallographically. Samples were prepared by sintering pressed pellets of the intimately mixed oxides (solid state reaction).

Ad 3.2. Preparatory work necessary for the use of measuring systems made of the four selected crucible materials was performed (Phase I). According to schedule and in accordance with the experimental results of the compatibility tests, the work was focused on tungsten (Phase II). Design specifications in the form of dimensional workshop drawings for the construction of the measuring systems from tungsten and specifications for the assembly of the measuring systems and the other subunits were worked out. The high-temperature rotary viscometer was modified for the use of tungsten measuring systems and made ready for operation.

Ad 3.3. A detailed working program and a flowchart were prepared for performing the measurements.

6. Results

Table 1: Compatibility Tests

Melt	Crucible	Melting time at temp. min/°C	Thickness decrease of crucible bottom µm	Melt behavior
Fe_3O_4	W	2/2020	crucible destructed	
corium E3	W	2.5/2010	290	phase separation
Mixtures of corium (A+R)3 and concrete W-%				
40:60	W	5/1630	7	homogeneous
50:50	W	5.5/1620	13	homogeneous
60:40	W	5.5/1640	25	onset of phase separation
60:40	Ta	0/1640	crucible destructed	

The conducted experiments substantiate the following generally significant conclusions:

- Mixtures of oxidic corium and concrete prepared by reacting water-free oxides in the solid state exhibit a smooth melting behavior in an inert gas atmosphere.
- Within the range of corium-concrete mixtures investigated, specimens of the above-described type are completely melted at 1640°C. The experiments performed so far do not allow any statement to be made on the lower temperature limit of the melting range.
- The molten mixtures are homogeneous, behave like glass, and solidify as a glass.
- The compatibility measurements led to the selection of tungsten as a crucible material for oxidic corium-

concrete mixtures. The stability limit of tungsten is determined approximately by the ratio 60 w-% corium (A+R)3 : 40 w-% concrete and a melting time of 5.5 minutes at 1640°C. For higher concrete contents and/or milder melting conditions, it can be anticipated that viscosity or other types of measurement can be conducted without disturbance.

The three requirements for starting viscosity measurements have been fulfilled: (i) a suitable crucible material has been identified, (ii) specifications for the construction of the measuring systems have been worked out and (iii) a detailed working program has been prepared. The particular objectives of Project RS 214 have thus been reached.

7. Next Steps

Viscosity measurements will be conducted within Project RS 214a.

8. Relation with Other Projects

9. References

/1/ Palinski, R.: Kompatibilitätsstudie. Euratom-Ispra. Presented at the 24th session of the expert committee on core melting.
February 1, 1977

/2/ Institut für Reaktorsicherheit, Köln: Quarterly Report V 76/3 on Project BMFT-RS 200. Gesellschaft für Kernforschung.

10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.Nov. 77 - 31.Dec.77	2.1	RS 310

Vorhaben/Project Title	Land/Country; Fördernde Institution/Sponsor; Auftragnehmer/Contractor
Emergency Core Cooling Analysis within the Core Meltdown Research Program Kernnotkühlanalyse im Rahmen des Forschungsprogramms Kernschmelze	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Babcock-Brown Boveri Reaktor GmbH-Mannheim BBR - TPR/AAB

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. Nov. 77	31. Dec. 78	Dr. G. Haury
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1977	750.859,48 DM

1. General Aim

The objectives of the Core Meltdown Project within the Reactor Safety Research Program of the Federal Ministry of Research and Technology so far encompassed the experimental and theoretical investigation of core meltdown by means of a hypothetical accident (under the boundary conditions of the postulated complete failure of all emergency cooling systems in the event of a double-ended rupture of the reactor coolant line of a pressurized water reactor with given remaining water levels in the reactor vessel).

The results of research within the scope of the Core Meltdown Project show that the meltdown of a fuel rod need only be expected when a cladding temperature of ca. 1 850°C is exceeded. However, the design of the emergency core cooling systems according to a RSK guideline shall ensure that the maximum cladding temperature after a loss-of-coolant accident will not rise beyond 1 200°C under any circumstances.

Thus an interesting region whose safety-engineering margins have not been investigated sufficiently to date lies between the region of safe emergency cooling of the reactor core in the event of an accident as defined by the RSK guidelines and the core meltdown which will occur in case of a hypothetical total and simultaneous failure of the multiple redundant emergency core cooling systems.

The region may be described such that it will only be reached in case of a given leakage spectrum in the primary system of the pressurized water reactor if an additional system failure is postulated

within the last available design redundancies.
The starting conditions for hypothetical core meltdown accidents are to be assessed by means of emergency cooling analyses of a primary system characterized by these system failures and be realistic initial conditions.

2. Particular Objectives

The system behavior for two rupture magnitudes (2 F rupture and rupture of approx. NW 100) is to be analized if, starting with the remaining redundancies stated in the licensing procedure, the various emergency cooling systems (core flooding tanks, high pressure and low pressure injection systems) are systematically further reduced or activated with a time lack.

This analysis is to be performed for the BBR pressurized water reactor (1 300 MWe) whose main feature is a two-loop system (2 hot legs, 2 steam generators, 4 cold legs with one pump to each).

3. Research Program

3.1 Preparation of "best estimate" input data and code adaptations
3.2 Analysis of a 2 F rupture, reduced number of emergency cooling systems
3.3 Analysis of a rupture NW 100, reduced number of emergency cooling systems
3.4 Analysis of a 2 F rupture, delayed injection from emergency cooling systems
3.5 Analysis of a NW 100 rupture, delayed injection from emergency cooling systems.

4. Computer Codes

The codes below are being used for the analyses:

CRAFT for the thermohydraulic behavior of the reactor cooling system during the blowdown phase

REFLOOD for the thermohydraulic behavior of the reactor cooling system during the refill and reflood phase

THETA 1 B for calculating the transient temperature distribution in a hot channel fuel rod in the course of an accident

CONTEMPT for calculating pressure and temperature in the containment atmosphere.

5. Progress to Date

Ad 3.1
- Preparation of "best estimate" parameters for the analyses.
- Preparation of a core power distribution by application of the peaking factor 2.0 (licensing runs 2.65).
- Preparation of the temperature, pressure and flow distribution in the primary system for load factor 100% (licensing runs 105%).
- Incorporation of realistic pump characteristics into the CRAFT code.

Ad 3.2 Repetition of the blowdown analysis of the licensing run (2 F guillotine rupture in the cold leg, 2 core flooding tanks, 1 high pressure system, 1 low pressure system, on the basis of realistic response time lags of active systems) with "best estimate" parameters.

6. Results

Ad 3.1 The axial peaking factor is 1.29.
The radial peaking factor is 1.55.

Ad 3.2
- The radially averaged fuel temperature in the hottest axial zone of the average load fuel rod in steady state under "best estimate" conditions is 215°C lower than the corresponding temperature under conservative licensing conditions.
- Towards the end of blowdown both the maximum cladding temperature and the average fuel temperature of the average load fuel rod in the "best estimate" case are approx. 145°C lower than for the licensing case.
- A lowering of the discharge coefficient (C_D-Moody) from the conservative figure in the licensing case (1.0) to

the more realistic values 0.8 or 0.6, respectively, results in an extension of blowdown time from 23.0 to 24.7 or 28.4 secs., respectively, and to a lowering of the maximum cladding temperature of the average load fuel rod of 12° or 45° C, respectively.

7. Next Steps

Continuation and completion of the work acc. to para. 3.2; start of work acc. to 3.3.

8. Relation with Other Projects

The paramount objectives of the project are identical with those of parallel projects of GRS-K (RS 311) and KWU (RS 306). Starting conditions, leak sizes and scope of work have been coordinated accordingly. The intermediate results of the work part of which is carried out in parallel will be compared by the Core Meltdown Committee of Experts even before conclusion of the projects; further work will be coordinated.

9. References

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.10.1977 - 31.12.1977	2.1	RS 311
Vorhaben/Project Title Best estimate-Rechnungen für das erweiterte Arbeitsprogramm des Projektes Kernschmelzen Best Estimate Calculations for the Extended Research Program of the Project Core Meltdown		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH Köln
Arbeitsbeginn/Initiated 1.12.1977	Arbeitsende/Completed 31.3.1979	Leiter des Vorhabens/Project Leader Dr. J. Keusenhoff
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1977	Bewilligte Mittel/Funds 672.580,-- DM

1. General Aim

The aim of the project is to analyse in better detail the occurences of possible failure combinations of emergency core cooling systems in the range of defined system failures in the licensing procedure and defined initial conditions for core meltdown events.

Particular emphasis will be given to:

- Determination of the failure combinations of partial ECC-systems under which the core remains coolable in spite of large damages

- Conditions where partial core meltdown will occur

2. Particular Objectives

- Analysis of the blowdown for two defined break sizes and two typical PWR-plants for different failure combinations of ECC-systems

- Analysis of the refilling and flooding phase

- Analysis of the temperature time history of fuel-rods up to 2200° C

- Analyses of the expected cladding deformations and cooling channel blockages as feedback of the temperature time history

3. Research Program

3.1 Data acquisition

3.2 Definition of the computer runs

3.3 Blowdown analyses for large breaks (KWU-type)

3.4 Analysis of the refilling and flooding phase for large breaks (KWU-type)

3.5 Core heatup calculations for large breaks (KWU-type)

3.6 Comparing analyses for large breaks of the BBR-reactor type

3.7 Analysis of thermohydraulics for small breaks (KWU-type)

3.8 Comparing analysis for small breaks of the BBR-reactor type

3.9 Interpretation of the calculation and final report

4. Experimental Facilities, Computer Codes

The work to be performed is completely theoretical and numerical in nature. So experimental facilities are not relevant.

For blowdown analyses of large and small breaks the computer code RELAP-4-GRS will be used. This code is a modification of the code RELAP 4, sponsored by the USNRC.

For the analysis of the refilling and flooding phase for large and small breaks the computer code REFLOS will be used. This code has been developed on the basis of FLOOD 4 and extended on the calculation of the refilling phase and the hot leg injection. For calculations regarding the core heatup the computer code TEMPA will be used. The code TESPA, developed from TEMPA, is available for the calculation of plastic cladding swelling.

5. Progress to Date

To 3.1: For a standard nuclear power plant of the KWU technical data have been compiled for the above mentioned computer codes.

6. Results

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7. Next Steps

Conduction of blowdown calculations by means of RELAP-4-GRS for large breaks for a KWU standard plant.

8. Relation to Other Projects

RS 182, LOFT-Program

RS 36B, Refilling Experiments with Simulation of the Circulation Loop

RS 310, ECC-Analysis within the Frame of the Project Core Meltdown

All new knowledges obtained so far from these projects will be taken into consideration in the calculation by means of RELAP-4-GRS and REFLOS.

9. References

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10. Degree of Availability

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Classification
2.1.

Title 1	
Coremelting - Measurement of physical properties of molten reactor core materials	Country : JRC
	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 1 2

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> :

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.01.77 - 31.12.1977	2.2	RS 76A
Vorhaben/Project Title: Experimentelle Untersuchungen der Dampfexplosion Experimental investigation of the vapour explosion		Land/Country: FFG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Euratom CCR Ispra Dept. Engineering
Arbeitsbeginn/Initiated: 1.01.1976	Arbeitsende/Completed: 31.12.1978	Leiter des Vorhabens/Project Leader: H.Kottowski, H.Hohmann
Stand der Arbeiten/Status: Continuing	Berichtsdatum/Last Updating: 31.12.1977	Bewilligte Mittel/Funds: 200.000,-- DM

1. GENERAL AIM

Theoretical and experimental investigation of the process of thermal interaction of molten reactor core materials and coolant as well as the estimation of the consequences of vapour explosions are the aims of this research programme.

2. PARTICULAR OBJECTIVES

In particular it is intended to study the thermal interaction between molten UO_2, molten structural materials 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 volumetric ratio of molten material and coolant.

3. PROGRAM

Theoretical and experimental work is underway to study the thermal interaction process of:

- stainless steel and water,
- UO_2-granulates and water,
- UO_2 and water.

3.1. Theoretical studies

Theoretical studies are underway to develop 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. Experimental studies

3.2.1. Shock tube experiments

These experiments are performed in order to provide data in a mono-dimensional test device to check and to adjust physical interaction models and calculation codes. The quantities 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 g),
- system pressure (up to 25 bar).

3.2.2. 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 molten material and coolant in a confinement simulating the reactor vessel. The quantities 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,
- visualization of the interaction history, using a high-speed camera.

4. EXPERIMENTAL FACILITIES

Because of the difficulty of modelling the fuel/coolant interaction, two test facilities have been built:

a) a mono-dimensional shock tube test rig for the study of the thermodynamic factors and
b) a tank facility for the simulation of reactor-like conditions at least as far as the volumetric ratio of fuel and coolant is concerned.

5. PROGRESS TO DATE

5.1. Theoreitcal studies

The development of the KAMIWA-Code (channel geometry) for melt/water interactions using temperature and pressure dependent physical properties is nearly terminated. The testing of the code is initiated.

5.2. Channel experiments

- Waterhammer tests measuring the impact momentum as a function of the driving pressure (pressure difference between blanket pressure in the expansion vessel and interaction crucible)
- Interaction test with molten steel at driving pressures of 1 bar, 5 bar and 25 bar.

5.3. Tank experiment

- Interaction experiments with molten stainless steel and water at melt to coolant volume ratios from 1:30 to 1:8 have been terminated.
- Interaction experiments with UO_2-granulates and water have been terminated
- Interaction experiments with molten UO_2 and water at small melt to coolant volume ratio ($<$ 1:1000) have been performed
- Experiments with melt to coolant volume ratios from 1:5 to 1:2 are in preparation.

6. RESULTS

6.1. Theoretical work

A thermodynamic consistent equation for the calculation of the specific enthalpy and an equation for the heat conductivity covering the temperature and pressure range between ambient conditions to 2500°C and 10000 bar respectively have been developed. These equations have been checked with the data available from the water skeleton table. The equations accord with the skeleton data within the experimental tolerances. These equations have been built into the KAMI-Code. Numerical testing is initiated.

6.2. Channel experiments

Before the real tests with molten material waterhammer tests were executed. The waterhammer pressure pattern was then compared with the interaction pattern and the fragmentation. In the following, three characteristic test results are discussed. The tests were performed at the same conditions but the pressure in the blanket vessel which determines the impact momentum. The pressure was changed from 1 bar to 5 bar and 25 bar respectively. The common test conditions were: (a) coolant temperature: 300^{o}K, (b) melt temperature: 1730^{o}K, (c) pressure inside the crucible: 1Pa, (d) weight of melt: 110 gr ÷ 120 gr. Water hammer test conditions: (a) crucible temperature: 300^{o}K, 1730^{o} respectively, (b) falling heigth of the liquid column: 12 cm. Measured quantities: F = reaction force of the interaction chamber, P_2 = pressure measured 15 cm above the interaction zone, P_3 = pressure measured 130 cm above the interaction zone, Δp = dynamic pressure due to the motion of the coolant measured downstream the test section, v_1 to v_6 = void sensors.

Fig. 1 and Fig. 2 show the measured waterhammer and the interaction pattern at 1 bar blanket pressure. The interaction (Fig. 2) does not show any violent pressure generation. The impact momentum seems to be attenuated by evaporation before the liquid column reachs the melt. This is supported by the fact that nearly no fragmentation occured. The vapour produced during the interaction condenses when travelling downstream the test section only reaching the void sensor Nr. 2. None of the experiments performed under these initial conditions showed high pressure peaks or remarkable fragmentation.
Quite different is the interaction pattern when only changing the blanket pressure to 5 bar. Fig. 3 shows the interaction recording. The waterhammer test shows a similar pattern as in the previous test series. For the interaction however two phases can be observed (a) the impact which leads to penetration of coolant into the melt and (b) the violent evaporation stage which can be identified as vapour explosion. The pressure peak measured downstream the interaction chamber (P2) at the second phase was 80 bar with a relatively long pressure tail until test section voiding. During the impact the coolant penetrated up to 2 cm into the melt and provoked high fragmentation. The vapour explosion expelled liquid melt-droplets up to 25 cm into the test section which melt thermocouples and destroyed void sensors. Increasing the system pressure to 25 bar the interaction becomes more violent. The interaction (Fig. 4) occurs again in various phases. Registration of P2 shows a pressure excursion up to 750 bar over a time of about 15 msec followed by another pressure burst of about 2msec duration. A pressure tail of up to 250bar

of about 30 msec duration follows the pressure bursts. The pressure peak before the 15 msec pressure excursion could be the recording of the impact momentum. The oscillations preceding the impact signal are caused by mechanical difficulties when opening the valve between the test section and the interaction chamber. During the impact the coolant penetrated up to 4cm into the melt. The vapour explosion expelled fragments up to the expansion vessel. The melt was highly fragmented. The interpretation of these experiments leads to the following preliminary conclusion:

- Free surface contact seems not to lead to vapour explosion. The surface boiling provokes rapid surface freezing which prevents esalation of fragmentation
- Coolant impact on liquid melts, even at moderate impact momentum, provokes fragmentation, penetration and quasi entrapment which results into violent vapour explosion.

6.3. Tank experiments

- Stainless steel/H_2O experiments

  Up to now about 50 interaction experiments with molten stainless steel and H_2O were performed in the tank facility. No vapour explosion has been observed. The main experimental parameters were:

melt: 1.5 to 3 kg	
melt/coolant volume ratio	$<$1:1000 to 1:8
melt coolant contact mode:	free contact mode (pouring of melt into H_2O)
melt temperature:	1800^oK to 2100^oK
H_2O temperature:	300^oK to 600^oK
H_2O gas content:	normal and degassed
system pressure:	1 bar

  The maximum observed overpressure in the reaction vessel did never exceed 2 bar, which diminished to the system pressure in about 10 sec after the first contact. An example of a pressure signal is given in Fig. 5. The solidified melts showed normally a very poor fragmentation. Motion of the melt fragments in the reaction vessels and the boiling pattern have been visualized by high speed cinematography (up to 4000 frames/sec).

- UO_2-granulates/H_2O experiments
  Seven interaction experiments with UO_2-granulates and water have been performed under different experimental conditions. The particle size distributions of the granulates (from some μ to about 1000 μ) were nearly the same for all experiments. The temperatures of the granulates were 1500°K, 1800°K and 2100°K, the H_2O-temperatures 300°K and 360°K. The coolant volume was about 200 l for all experiments. The hot material was partially poured in the coolant or brought directly into contact with the coolant by destroying the bottom of the crucible.
  A maximum pressure of 1.6 bar with a ramp rate of ~ 3 bar/sec was measured. An example of a pressure signal is shown in Fig. 6.
- UO_2-H_2O experiments
  All experimental trials to bring molten UO_2 at ~ 3200°K directly into contact with water failed due to technical problems of destroying the Tungsten crucible.
  Finally 1,5 kg of molten UO_2 was poured in ~ 200 l water of about 300°K. Also in this experiment no violent interaction was observed. Due to the evaporation of the H_2O an overpressure of about 1 bar was measured in the reaction vessel.

7. FUTURE STUDIES

7.1. Theoretical work

Improvement of KAMI-WA

7.2. Channel experiments

- UO_2-granulates/water tests,
- molten UO_2-water tests.

7.3. Tank experiment

- Interaction experiments, molten UO_2 and water at melt/coolant volume ratios between 1:5 and 1:2. This experimental programme has to be terminated on April 30th, 1978.
- The interpretation of the experimental results is intended to be performed at the IKE (TU-Stuttgart).

8. RELATIONS WITH OTHER PROJECTS

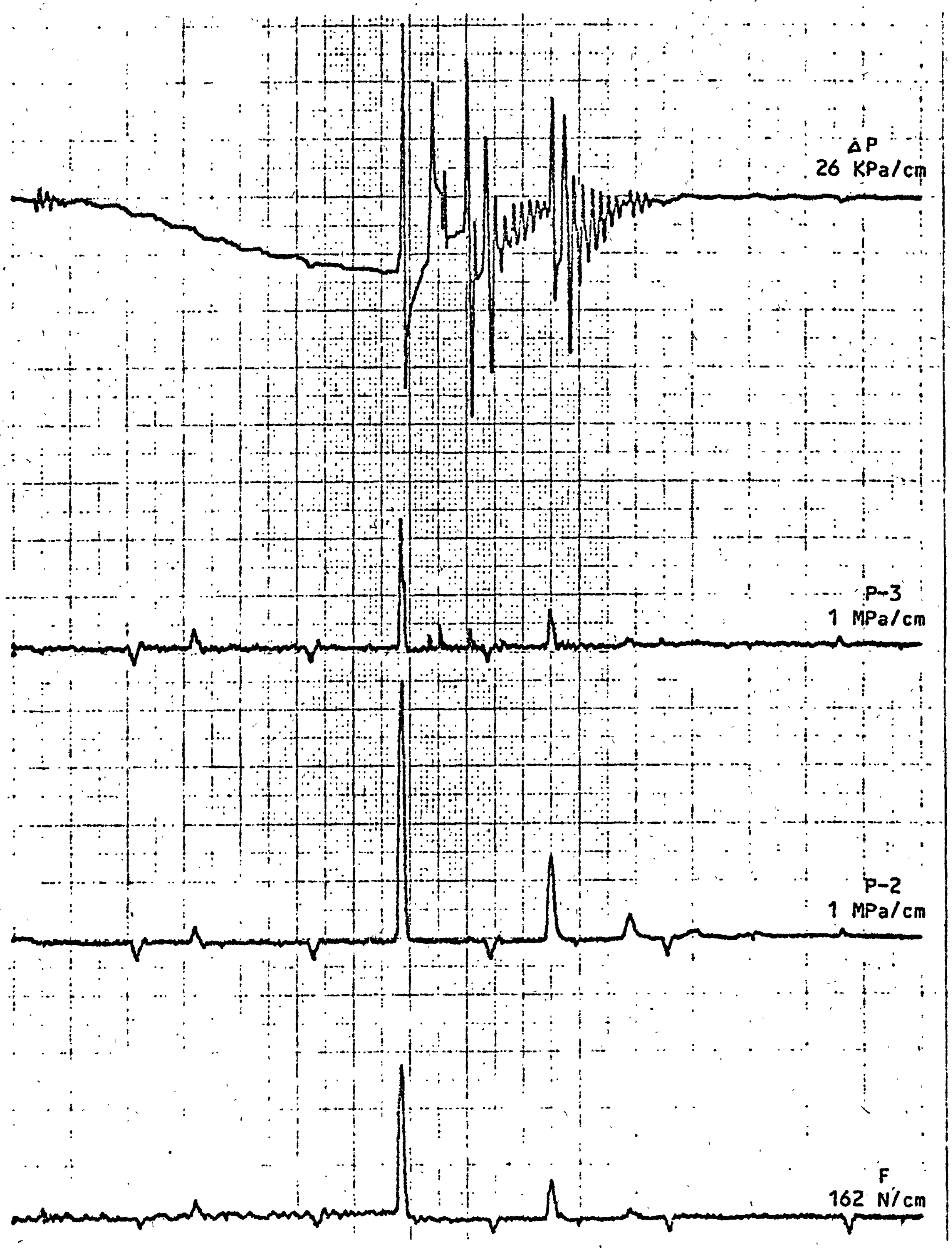
No new relations with respect to 1976 Annual Report.

9. REFERENCE DOCUMENTS

- GRS 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 availability)
- Fachbericht des IKE und des CCR Ispra über den Stand der begleitenden theoretischen Arbeiten zur Dampfexplosion; R. Benz, G. Fröhlich, H. Goldammer, H.M. Kottowski
- Experimentelle Untersuchung zur Dampfexplosion; H. Hohmann, H.M. Kottowski, F. Toselli; 3. Technischer Fachbericht zum Forschungsvorhaben BMFT-RS76; Technical Note 161/293/77 - March 1977
- Theoretical and Experimental Investigation of the Similation of Fuel Coolant Interaction in a Shock Tube Configuration; H.Goldammer, H.M. Kottowski, International Meeting Fast Reactor Safety and Related Physics, Oct. 5-8-76, Chicago
- Literaturstudie zur Dampfexplosion, 1. Technischer Fachbericht zum Forschungsvorhaben BMFT-RS76 IKE (Febr. 1976), R. Benz, G. Fröhlich, H. Unger
- Theoretische Studien zur Dampfexplosion; 2. Technischer Fachbericht zum Forschungsvorhaben BMFT-RS76, IKE (Apr. 1976), R. Benz, Fröhlich G., Goldammer H., Kottowski H.M., Unger H.
- Theoretische und experimentelle Untersuchung zur Dampfexplosion Abschlußbericht zum Forschungsvorhaben BMFT-RS76, EUR/C-IS/116/77 d, R. Benz, Fröhlich G., Goldammer H., Hohmann H., Kottowski H.M., Lazarus J., Toselli F., Unger H.

10. DEGREE OF AVAILABILITY OF THE REPORTS

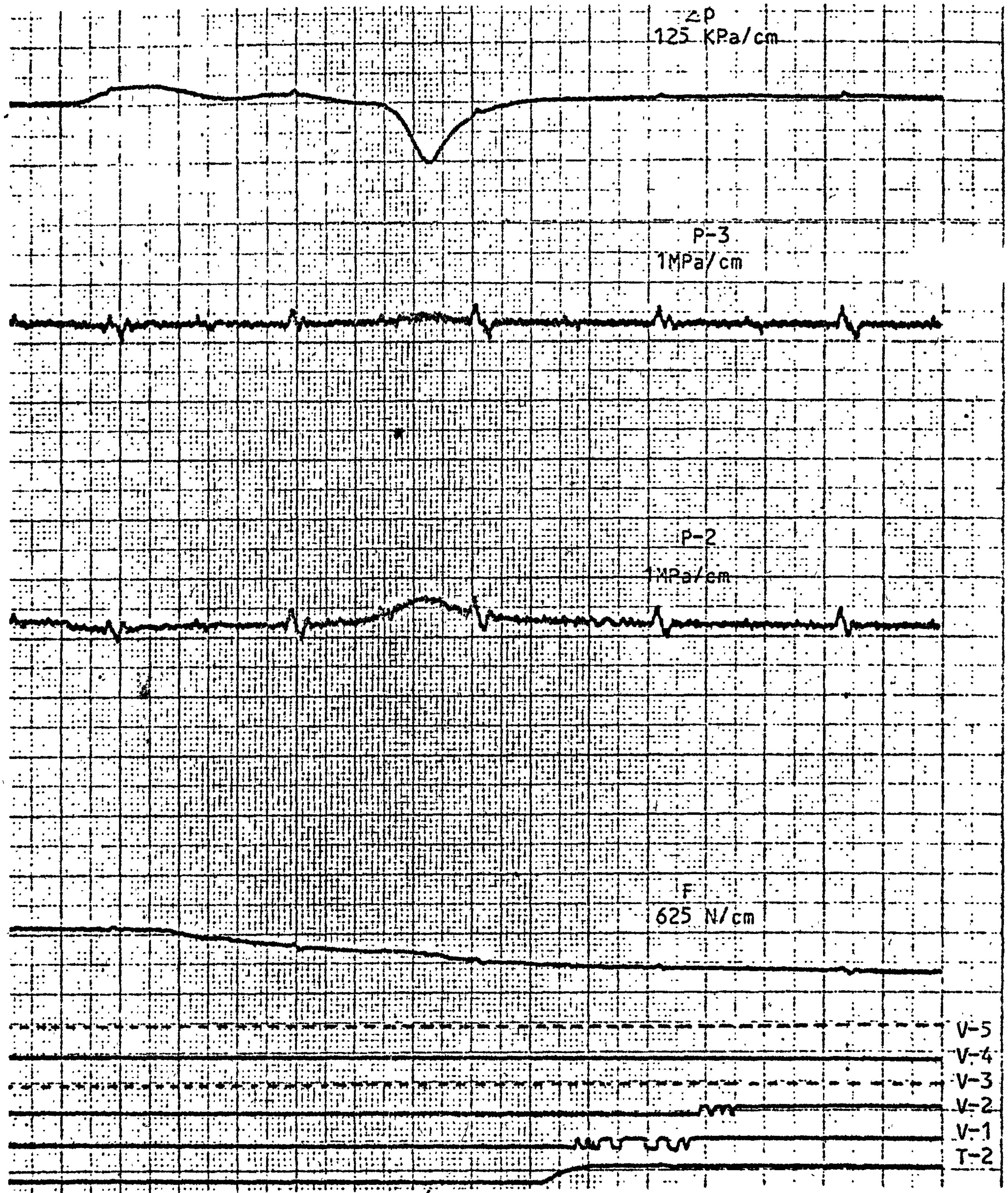
All reports available.



F.C.I. Channel experiment. Waterhammertest. Fig. 1

P_c = 100 KPa T_c = 300 K Timescale: 31.25 ms/cm

P_f = 1 Pa T_f = 300 K Crucible: Mo, normal geometry

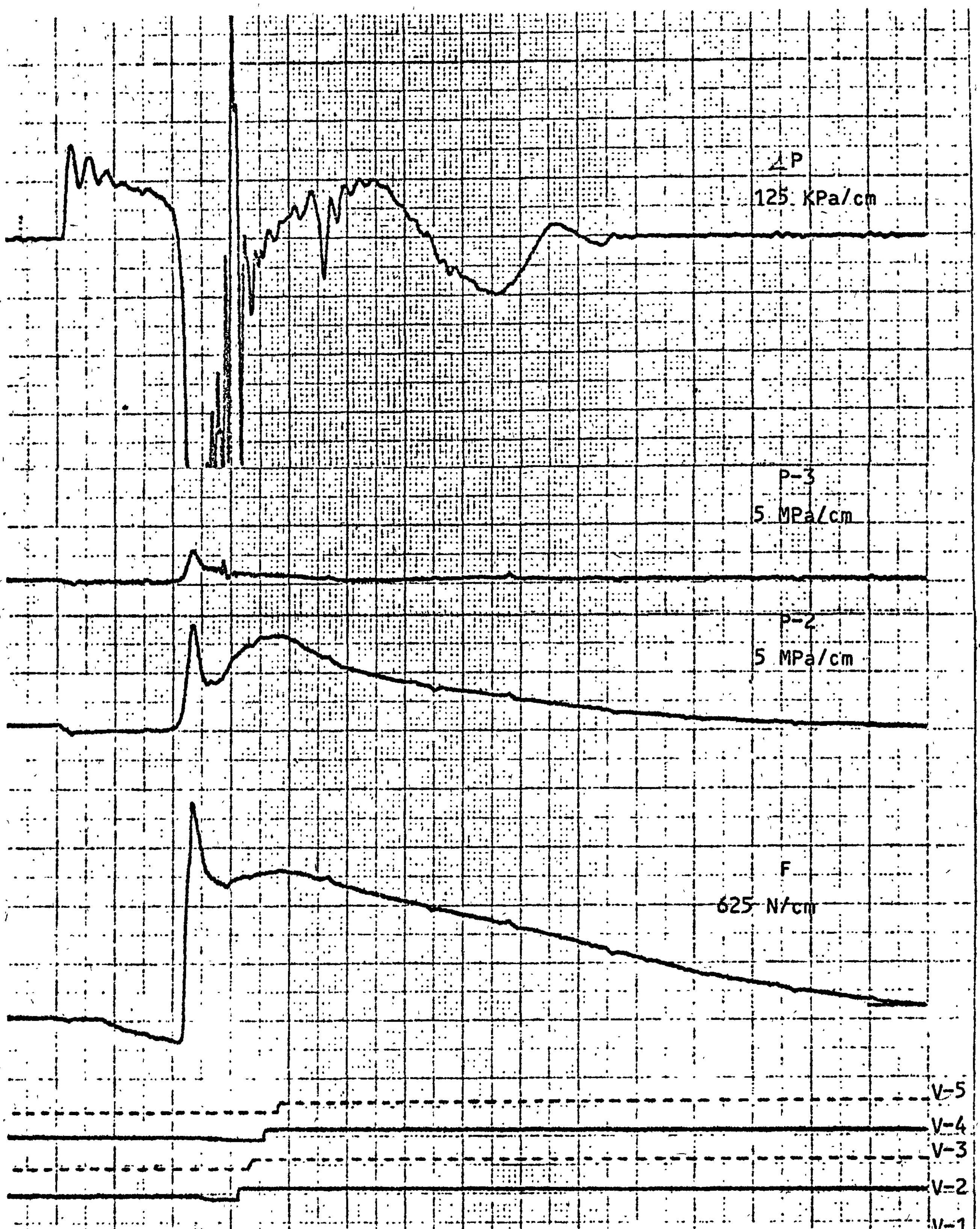


FCI Channel Experiment H_2O/Steel

P_c = 100 KPa T_c = 293 K Timescale: 31,25 ms/cm

$P_f \leq$ 1 Pa T_f = 1730 K

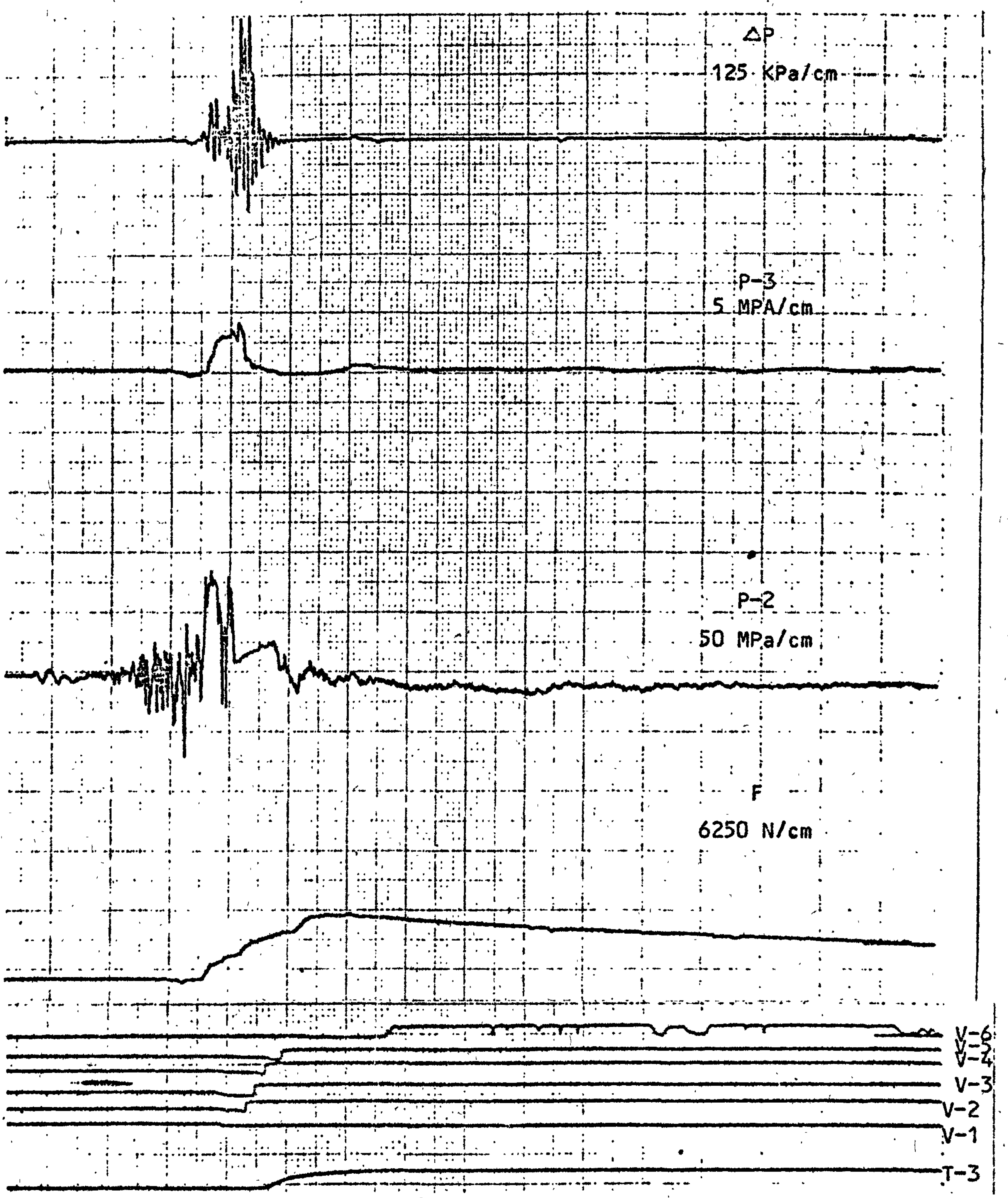
Fig. 2



FCI-Channel Experiment H_2O/Steel. Fig. 3

P_c = 500 KPa T_c = 293 K Timescale: 31,25 ms/cm

P_f 1 Pa T_f = 1730 K

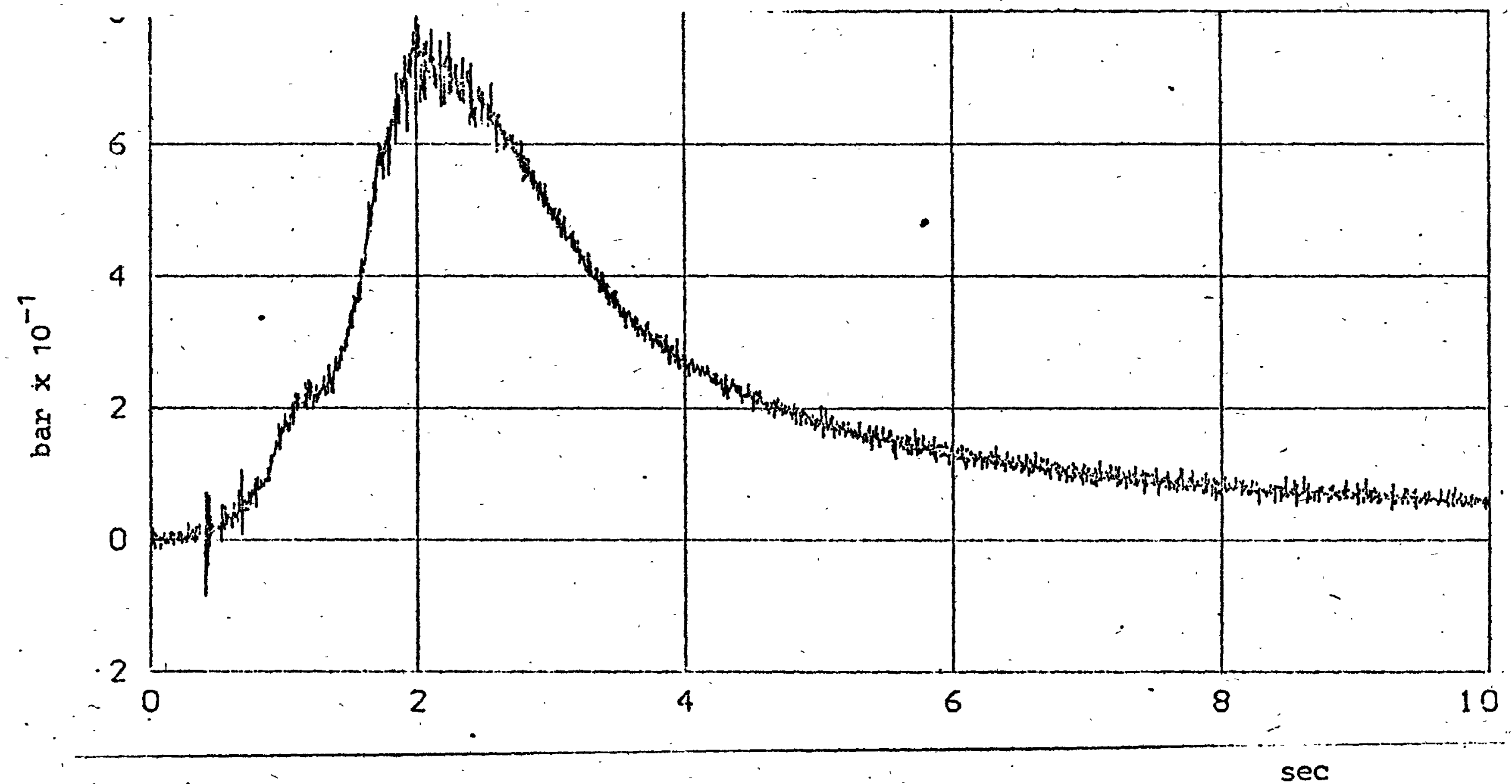


FCI-Channel Experiment H_2O/Steel

P_C = 2,5 MPa T_C = 293 K Timescale: 31,25 ms/cm

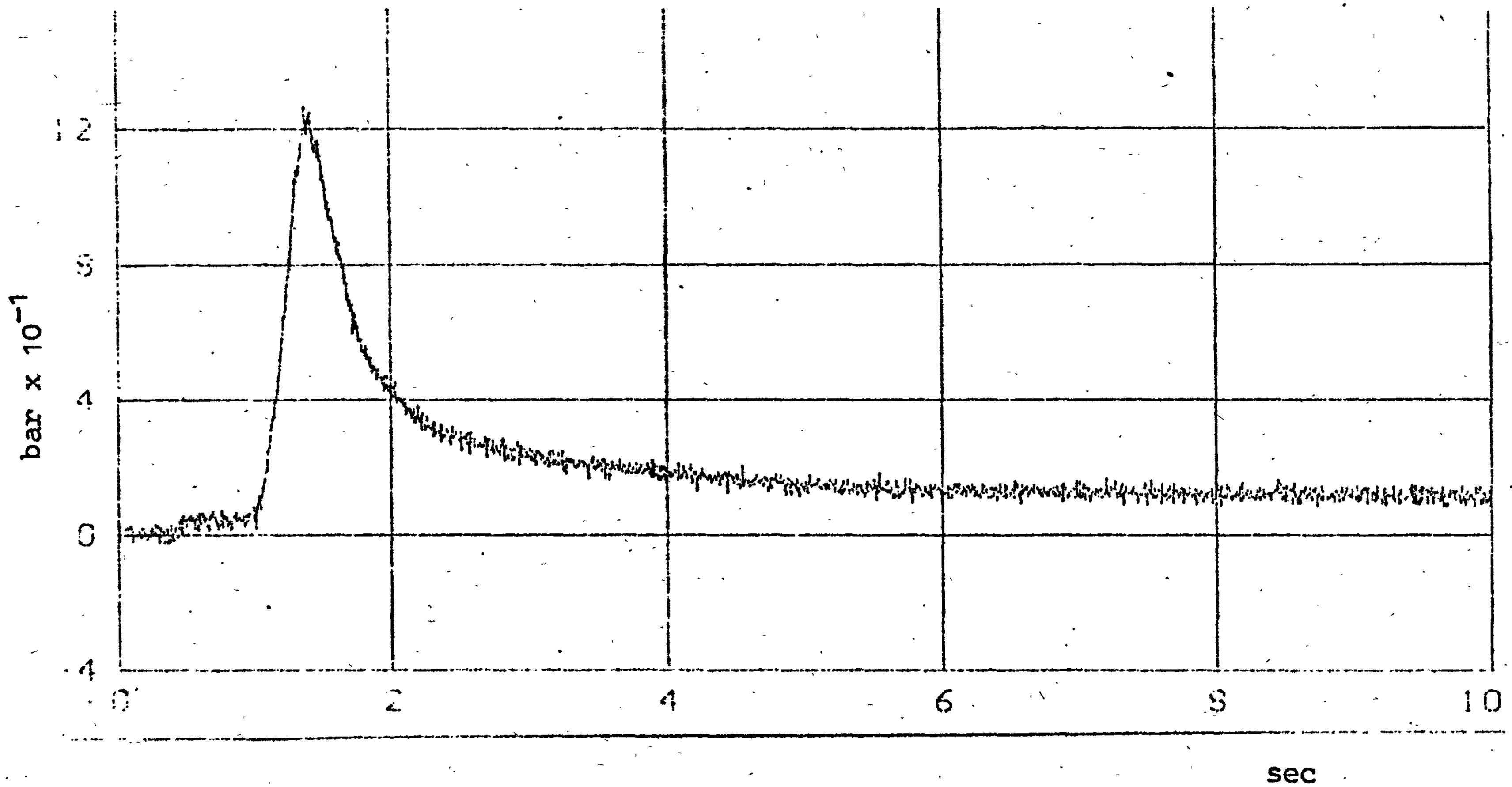
$P_f \leq$ 0,02 Pa T_f = 1730 K

Fig. 4



FCI- Tank Experiment (Steel/Water); T_f = 2050 K, T_c = 300 K

Fig. 5



FCI-Tank Experiment (UO_2-Granulate/Water); T_f = 2075 K, T_c = 350 K

Fig. 6

Berichtszeitraum/Period 01.01.-31.12.1977	Klassifikation/Classification 2.2	Kennzeichen/Project Number RS 206
Vorhaben/Project Title Theorie zur Dampfexplosion in Tankgeometrie, Entwicklung von Fragmentationsmodellen, experimentelle Untersuchung stark transienter Siedezustände Theoretical Investigation of Vapor Explosions in Pool-Type Geometry, Development of Fragmentation Models, Experiments Related to Highly Transient Boiling		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Universität Stuttgart Institut für Kernenergetik u.Energiesysteme
Arbeitsbeginn/Initiated May 1, 1976	Arbeitsende/Completed June 30, 1979	Leiter des Vorhabens/Project Leader Prof.Dr.Unger/DI Benz
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating December 31, 1977	Bewilligte Mittel/Funds 196.000,-- DM

1. General Aim

Within the frame of the research project investigating core meltdown problems of light water reactors experimental and theoretical calculations on hypothetical vapor explosions in light water reactors are performed. Both activities initiated are intended to lead to a broadening of the knowledge on conditions, course and extend of vapor explosions possibly occuring during a hypothetical core meltdown.

2. Particular Objectives

- Estimation of the upper limits of the energy release and the pressure buildup during a reaction between molten materials and water in reactor geometry under reactor-relevant conditions with respect to specific courses of the accident after the beginning of the meltdown
- Clarification of the conditions and the course of the fragmentation processes which might lead to vapor explosions
- Development of calculational models in order to describe the fragmentation based on relevant physical mechanisms
- Development of a calculational model in order to describe vapor explosions in pool-type geometry
- Theoretical calculations of the experiments performed in pool-type geometry at the EURATOM-research-center Ispra
- Experimental research on highly transient boiling, particularly with respect to the fragmentation of melts
- Experimental research on the trigger conditions for a coherent fragmentation by means of entrapment.

3. Research Program

3.1 Performance of an Investigation Using Engineering Methods

Estimates on upper limits for energy release and vapor pressure buildup during hypothetically postulated vapor explosions in reactor geometry.

3.2 Development of Fragmentation Models

Theoretical investigation of fragmentation models for various materials and reactor conditions. Selection of mechanisms which may occur during core meltdown and development of calculational models in order to describe course and extend of the reaction (e.g. bubble collapse model, shock wave model).

3.3 Development of a Model for Pool-Type-Geometry and Calculation of Experimental Results

Coupling of the fragmentation models within a computer code describing the course of a vapor explosion. Collection of data obtained from the experiments performed in pool-type-geometry, calculation of partial results from experiments, e.g. surface increase of the melt, fragmentation time, pressure distribution.

3.4 Experimental Research on Highly Transient Boiling, Particularly with Respect to Fragmentation

Experiments to the highly transient boiling with large temperature differences will be performed in order to get data to the heat transfer and a picture of the phenomena under these conditions. Included are measurenents of the direct contact during and after the stable film boiling.

3.5 Experimental research of trigger conditions for a coherent fragmentation by means of entrapment

Trigger conditions for a coherent fragmentation will be verified by means of entrapment of water in different melts. Temperatures, layer thickness and material of the melt will be varied as well as the temperature and the mass of the water entrapped.

4. Experimental Facilities, Computer Codes

To estimate the strongly transient pressure distribution, and the quasi-static pressure-buildup in reactor geometry a computer program has been developed. Another program estimating fragmentation of the melt based on the vapor bubble collapse has been worked out and improved.

5. Progress to Date

Development of a vapor bubble collapse model. Description of fragmentation by shock waves. Test status of the computer code describing interactions in the Ispra-tank-facility. Initiation of measurements of boiling data. Development of heat transfer correlations on subcooled film boiling around spheres.

6. Results

The vapor bubble collapse model estimates the surface area increase of the melt as a function of time. The model predicts the fragmentation time and fragmentation degree of low melting materials within the right order of magnitude as well as the dependency of the fragmentation on the subcooling. A heat transfer correlation on subcooled film boiling around spheres is developed.

7. Next Steps

Development of a fragmentation model based on shock waves and of a code for interactions in tank geometry. Experiments to transient boiling at free surfaces and to the entrapment as a trigger mechanism of vapor explosions will be continued.

8. Relation with Other Projects

For pool-type-experiments performed at CCR-ISPRA (research project BMFT RS 76A) theoretical calculations are done.

9. References

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10. Degree of Availability of the Reports

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Classification
2.2.

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°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°C (pressure 25 atm). Instrumentation is provided for the measurement of the pressure and temperature excursions accompanying interaction.

6

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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

128

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 : -

PROJECT TITLE : Fuel-Coolant Interaction - Out of pile studies in appropriate geometries	LWR 2.2.
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : JRC Ispra
DATE INITIATED : 1973 DATE COMPLETED : 1980	PROJECT LEADER : H. Kottowski H. Hohmann

Description :

1. General aim

Aim of the theorethical and experimental programme is the investigation of the thermal interaction process of molten reactor core materials and coolant, as well as estimation of the consequences of vapour explosions in a core.

2. Particular objectives

Collection of experimental data on thermal interaction of molten fuel (UO_2), or reactor structural materials with water in a tank facility. Reactor-like conditions are simulated with respect to geometry and quantity of molten materials.
The experimental results will be compared with theoretical model predictions to acquire a better understanding of the interaction phenomena.

3. Experimental facilities and programme

3.1 Facility

The core-melt material (up to 4 kg) contained in a crucible is heated in a furnace (up to 3000°C) and dropped through a fall-guide into a reaction tank (diameter: 60 cm, height: 160 cm, operation pressure 40 bar) with variable amounts of water (temperature up to 230°C). Smaller vessels (simulating PWR or BWR pressure vessels) can be installed in the reaction tank to get fuel/coolant volume ratios, representative for core melt-down accidents. Instrumentation is provided for:

- measurement of pressure and temperature excursions, accompanying the interaction
- measurement of strains in the vessel walls and support structure
- visualization of the fragmentation process using high speed cinefilms

- determination of the specific surface area of the debris by means absorption technique.

3.2 Experimental Programme

The experimental programme, agreed with the German BMFT in the frame of a collaboration contract, consists of two parts:

A – Experimental studies of the factors influencing the interaction process at coolant/fuel volume ratios of about 500. The molten material is poured in the reactions tank containing ~ 200 l of H_2O.

B – Experiments in water reactor simulated geometries at coolant/fuel volume ratios from 2 to 5. For this purpose smaller round bottomed vessels will be supported inside the reaction tank. In addition to this, experiments with UO_2-granulates of particle diameters in the order of 100 microns and known surface areas will be performed.

The experimental programme is listed in the following tables:

Experimental Parameters for programme A

melt material	stainless steel DIN 1.4550	UO_2
mass of melt	3 kg	4 kg
temperature of melt	1500 °C	2900 °C
gas contents of melt	unknown	unknown
H_2O temperature system pressure	20 °C 1 bar	20 °C 1 bar
H_2O temperature system pressure	20 °C 25 bar	20 °C 25 bar
H_2O temperature system pressure	80 °C 1 bar	80 °C 1 bar
H_2O temperature system pressure	220 °C 25 bar	220 °C 25 bar
gas contents of H_2O	saturated at H_2O temp. degassed	saturated at H_2O temp. degassed

Experimental Parameters for Programme B

melt material	stainless steel DIN 1.4550	UO_2 - granulates	UO_2
mass of melt	3 kg	3 kg	4 kg
temperature of melt	1500 °C	1800°; 2200°C	2900°C
gas contents of melt	unknown	---	unknown
H_2O temperature system pressure	20 °C 1 bar	20 °C 1 bar	20 °C 1 bar
H_2O temperature system pressure	80 °C 1 bar	80 °C 1 bar	80 °C 1 bar
gas contents of H_2O	saturated at H_2O temp. degassed	saturated at H_2O temp. degassed	saturated at H_2O temp. degassed
coolant/fuel volume ratio	2;5	2;5	2;5

4. Progress to date

4.1 Experiments

The experiments with stainless steel of programme A (see table) have been terminated.

4.2 Essential results

Conclusion of these series of measurements is, that only very weak thermal interactions have been observed in the described experimental conditions.
Pressure excursions never exceeded 0,5 bars and the steel fragmen tation was very poor.

5. Next steps

- Interaction tests with molten UO_2 and H_2O for programme A.
- Preparation of experiment for programme B.

6. Relations with other projects

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7. Reference documents

   1975 JRC Safety Progress Report

8. Degree of availability

   free available

9. Budget

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10. Manpower

   6 man/year.

6. Relations with other projects

Probabilistic accidental transients analysis.

7. Reference documents

[1] Validation of probabilistic transient analysis by comparison with theoretical data of the Obrigheim PWR Working paper, Oct. 1976.
J. Amesz, G.F. Francocci.

Classification

2.2

Title 1	COUNTRY UNITED KINGDOM
FUEL-COOLANT INTERACTIONS (1)	SPONSOR UKAEA
	ORGANIZATION WINDSCALE (RDL)
Title 2	Project Leader DR H LAWTON
Initiated 1970 Completed :	Scientists:
Status : Last updating	

Description:

1. General Aim

To predict and thus contain the mechanical effects following core melt-down.

2. 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.

3. 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. Batelle has seen evidence of chemical reaction in the Al/H_2O system apparently causing reaction, and this mechanism should be borne in mind.

4. Project Status

The results of this series of test will be reviewed about the end of 1975.

5. 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 UNITED KINGDOM SPONSOR UKAEA ORGANIZATION CULHAM LABORATORY
Title 2	Project Leader DR T DULLFORCE
Initiated 1972 Completed : Status : Last updating 1976	Scientists:

Description:

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 studies. High-speed cine films (500 frames per sec) have been made and studied. Initially the system molten tin/distilled water has been studied; other materials are planned.

Reference Documents

D Buchanan, T A Dullforce, Nature 245, September 1973. Mechanism for Vapour Explosions.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	2.3	RS 154

Vorhaben/Project Title	Land/Country; Fördernde Institution/Sponsor; Auftragnehmer/Contractor
Kernschmelzen - Untersuchung der Wechselwirkung zwischen Kernschmelze und Reaktorbeton Core Meltdown - Investigation of the Interaction between Coremelt and Concrete	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik RB 33, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 2. 75	31. 12. 77	Dr. Peehs
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	31. 12. 77	1'691.787,-- DM

1. General Aim and 2. Particular Objectives

To provide a risk analysis for LWR-accidents of very low probability, the research project "Kernschmelzen" of the BMFT was started in the FRG in 1971. The objective of the related R and D-work is to study the consequences of a hypothetical core melting accident. This may occur if we hypothesize that the emergency core cooling system fails completely after a loss-of-coolant accident. Performing this analysis, one can define the chronological order in the sequence of a core melting accident

1) the heating up of the core until the core structure may fail; which starts at a certain water level in the core and ends with the failure of the grid plate,

2) the second phase is characterized by the evaporation of the water left in the lower plenum and it lasts after the dryout of the pressure vessel until a molten core debris is formed,

3) the third phase is concerned with the heating up of the pressure vessel after the melt was formed and

4) finally, after the pressure vessel failure, the molten Corium will interact with the concrete structure beneath the pressure vessel.

Very early it was recognized that the concrete structure is an excellent barrier against propagating molten Corium. Therefore, a detailed R and D-program was initiated to study experimentally the high temperature behaviour of concrete and the interaction of molten Corium with the concrete. The objectives of these activities are to provide detailed information to compute the propagation of the core melt in the foundation of a LWR and to establish energy and mass balances for the fourth phase of the hypothetical core-melt-accident.

3. Research Program

The research program is broken into the following subtasks:

- Compilation of literature on 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.

4. Experimental Facilities

Within relevant R + D-work the same experimental facilities are used as with the investigation of metallurgical and chemical reaction between Corium and RPV-material. To investigate the thermal shock behaviour of concrete a plasma torch within an electrical input up to 40 kW is available.

5. Progress to Date

The investigation of the thermal shock behaviour of a limestone aggregate concrete was investigated. Also the gas penetration of concrete as function of the temperature was investigated. The complete set of data from the investigations of the different properties of heated concrete was evaluated and compared to theoretical estimations.

6. Results

Basaltic concrete looses its water bound in the cement phase at 100 - 120 °C, 550 °C and 800 °C as shown by DTA- and DGT-analysis. Parallel to this water losses the cement phase between the aggregates shrinks as microscopic investigations clearly indicate. The dehydrated cement starts to sinter at 800 °C thus reinforcing the concrete structure. Investigating the elongation behaviour of the cement phase, cement plus sand and the complete concrete shows, that the thermal elongation is mainly determined by the aggregates. The density of concrete decreases meanly by the waterlosses and the $\alpha \rightarrow \beta$ -phase change of SiO_2. The melting point of basaltic concrete is about 1300 °C. The total specific melting enthalpy is 5225 J/cm^3 thus resulting in an erosion rate of 22 mm/min at a heat flux of 200 W/cm^2. The heat capacity of basaltic concrete increases from 0.96 J/gr · grad at room temperature to 1.88 J/gr · grad at 1100 °C. The $\bar{c}$ of 1.54 J/gr · grad fits well together with those values calculated from the total melting enthalpy. The thermal diffusivity varies between $5.8 \cdot 10^{-3}$ cm^2/s at 20 °C, $3,2 \cdot 10^{-3}$ cm^2/s at 500 °C and $4,5 \cdot 10^{-3}$ cm^2/s at 1100 °C. The thermal conductivity measurements resulted at 50 °C in $15.9 \cdot 10^{-3}$ J/cm s grad, having a minimum at 300 °C of $11,3 \cdot 10^{-3}$ J/cm s grad. It increases again to 1.96 J/cm s grad at 1100 °C. The basaltic concrete shows an excellent thermal shock resistance.

7. Next Steps

The final report will be written.

8. Relation with Other Projects

RS 183.

9. References

2. Technischer Fachbericht (Okt. 1977)

10. Degree of Availability

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.77 - 31.12.1977	2.3	PNS 4314 (4244)
Vorhaben/Project Title Konstitution und Reaktionsverhalten von LWR-Materialien beim Coreschmelzen Constitution and Reaction Behaviour of LWR Materials at Core Melting Conditions		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS) IMF I
Arbeitsbeginn/Initiated January 1974	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader H.Holleck, A.Skokan
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31.12.1977	Bewilligte Mittel/Funds

1. General aim

Theoretical and experimental investigations on chemical reactions between coremelt, fission products and concrete.

2. Particular objectives

Experimental examination of the chemical interactions in a complex system containing corium + fission products + concrete (chemical reactions and vaporization behaviour).

3. Research program

3.1 Melting experiments upon homogeneous powder samples (corium + fission products + concrete) differing in composition and in degree of oxidation.

3.2 Vaporization tests upon molten samples containing corium + fission products + concrete

4. Experimental facilities

Laboratory high temperature furnaces (tungsten resistance f., induction f., electric arc f.), metallography, ceramography, X-ray diffraction, microprobe analysis, chemical analysis.

5. Progress to date

Reg.3.1 Several series of tests have been performed variing the composition of corium (A,E,A+R,E+R), the degree of oxidation of corium, the concentration of fission product elements, the composition of concrete (silicate type and limestone aggregate), and the ratio of the components corium and concrete.

Reg. 3.2 Vaporization of corium melts of the lowest degree of oxidation has been examined and preliminary tests for the qualitative examination of the vaporizing species have been performed.

6. Results

Reg.3.1 - Corium + fission products: In the corium melt the partition and the reaction behaviour of the fission product elements is deduced in the first instance from the differing stability of their oxides. Beyond that the reactions among the components themselves at increasing degree of oxidation have an effect on the partition of the fission product elements to the oxide and the metallic melt.

- Corium + concrete: The oxide part of the corium melt is readily mixed up with molten concrete. Its solidification temperature is continuously decreased with increasing amounts of dissolved concrete. If the aggregate is of the silicate type, solidification of the oxide melt leads to a vitreous material even at cooling rates as low as 1 K/min. When limestone aggregate is used, several crystalline phases (UO_2, Ca_2UO_4, Ca_2ZrO_4 and a Ca-Cr-oxide) are observed in the solidified oxide melt.
- Corium + fission products + concrete: The partition of the fission product elements to the oxide and the metallic melt is the same as in the system without concrete. The concentration of the fission product elements in the vitreous silicate type material is homogeneous. In the solidified oxide melt of the limestone aggregate type Ce and Nd were found in the uranium containing phases, whereas Sr was found together with Zr.

Reg.3.2 The loss by vaporization of molten corium E at 2950 - 3000 K is lower than formerly accepted. It amounts to ∿ 2 wt.-% after 5 minutes, and ∿ 4 wt.% after 10 minutes. Its increase with rising temperature is approximately linear. The loss is caused essentially by vaporization of steel components.

7. Next steps

Reg.3.1 Examination of samples from large scale experiments and experimental tests to check the possibility of substituting UO_2 by inactive oxides in these experiments.

Reg.3.2 Further experiments in order to determine the species vaporizing from the melt.

8. Relation with other projects

This project is connected with the other research projects of the German reactor safety research program dealing with core meltdown accidents.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	2.3	RS 237
Vorhaben/Project Title Ingenieurstudie zur H_2-Entwicklung aus der mit Beton wechselwirkenden Kernschmelze Engineering Study on H_2-Formation during the Interaction between Core Melt and Concrete		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik RZR 2, Erlangen
Arbeitsbeginn/Initiated 1. 10. 1976	Arbeitsende/Completed 31. 10. 1977	Leiter des Vorhabens/Project Leader C. Goetzmann/H. Hassmann
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating 31. 12. 1977	Bewilligte Mittel/Funds 110.520,-- DM

1. General Aim

The scope of this study is to clarify whether or not the containment integrity is endangered because of the release of steam passing through the melt during heat-up of the concrete, and what influence this has on the conditions prevailing in the containment atmosphere after a core melt.

2. Particular Objectives

Current test results on core melting show that the core melt is not expected to remain in the RPV. After the RPV failure the core melt comes into direct contact with the concrete of the containment foundation, the water present in the concrete will evaporate due to thermal heat-up. Most of the steam will be released by the melt into the containment atmosphere. Through various mechanisms, hydrogen build-up may occur which - if sufficiently concentrated in the containment atmosphere - by possible combustion or explosion presents a hazard to containment integrity.

The aim of the present research task therefore comprises analyses of the pressure and temperature build-up in the containment, considering hydrogen combustion as an additional heat source a) for a conservative maximum case and b) based on release rates consistent with current experience.

This is a GfK/PNS and KWU joint venture.
The work outlined below refers to KWU activities only.

3. Research Program

3.1 Definition of the initial PWR conditions, compilation of pertinent assumptions

3.2 Maximum estimate of hydrogen release in a PWR

3.3 Estimate of the influence of the expected H_2/H_2O-release on the integrity of the PWR containment

3.4 Estimate of the effects to be expected from H_2-release on the BWR containment integrity

3.5 Discussion of results

4. Experimental Facilities, Methods

The analysis will mainly be based on the experience gained in the area of process analysis during hypothetical core melt accident. The computer program BILANZ, developed for R+D task RS 72 a, b, can be used to calculate, besides other parameters, the time history of pressure, temperature and the mass distribution in the containment atmosphere which are of high importance in connection with hydrogen build-up.

5. Progress to Date

A theoretical model was formulated, which calculated the time dependent oxidation of the metallic components of the molten core, the heat production and hydrogen formation. This model was integrated into the KAVERN I-code.

A theoretical model, which calculated the reaction processes in the containment atmosphere, was programmed and integrated into the containment code COCO.

6. Results

The metallic components of the molten core, including the molten iron from the rebar in the concrete, will be fully oxidized after about 1 day. In this time period a lot of hydrogen will be set free. Theoretically without H_2-combustion in the containment atmosphere the tolerable pressure in the containment will be doubled after 2 1/2 days. The ignition point of the steam-air-hydrogen mixture is already reached after about 4 hours from the beginning of the accident. The results indicate that for standard PWR there will be no overpressure failure of the containment caused by combustion of the hydrogen because no sufficient oxygen is available.

7. Next Steps

The work has been completed, the final report will be written.

8. Relation with Other Projects

RS 154, RS 183

9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	2.3	RS 288

Vorhaben/Project Title	Land/Country: FRG
Dampfentwicklung nach Fluten der Kernschmelze	Fördernde Institution/Sponsor: BMFT
Steam Evolution after Core Melt Flooding	Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik RZR 2, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 10. 77	31. 12. 78	C. Goetzmann/H. Hassmann
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 77	215.067,-- DM

1. General Aim

Under the scope of theoretical work performed in connection with R + D project Core Melting, KWU will prepare - among other pertinent projects - a study on "energy balances after hypothetical RPV failure". In addition to the development of an overall model in co-operation with the TU-Hannover and the performance of experimental work, a detailed analysis of the penetration of the concrete base by direct contact with core melt after RPV failure will be carried out.

2. Particular Objectives

Current results have revealed that, following reactor vessel failure, the bottom part of the detached concrete shield in the reactor cavity, which separates the melt from the sump, will be penetrated after approx. 4 to 5 hours. This is the earliest possible instance for consideration of the effects of the sump water flooding the core melt. This study will be a joint venture of the TU-Hannover and KWU, whereby TU-Hannover will consider the molten condition, i.e. the thermohydraulics, and KWU the solid phase as well as the integration of the models in the BILANZ code, and performance of the computer operations.

The following description refers to the KWU work only.

3. Research Program

The following key questions are to be clarified:

Will the detached lower concrete shield because of its weight slip down, thus impeding massive penetration of the water? Attention should be paid to the fact that the outside of the concrete shield passes through concrete ribs.

Would the rapid quenching of the melt from above result in the formation of a stable solid crust?

How much of the overall generated decay heat and how much stored heat is used in the evaporation of sumpt water; how much for melting of the foundation?

How high a pressure is reached in the containment? Would a depressurization occur upon penetration of the concrete foundation by the melt? If so, when?

4. Test Facilities

No test equipment is required for this program. The models needed for the test are already integrated into the general core melt BILANZ code.

5. Progress to Date

First of all a balance of forces and a geometrical model were prepared so that the slippage of the internal concrete shield can be predicted. With respect to geometry, it should be noted that the computer program KAVERN I, which was prepared under RS 183, will be used for the investigations.

6. Results

No special results exist up to now.

7. Next Steps

Details of the model are to be elaborated to the extent that influences such as changes in level of the surface of the molten pool, and the heat-up of the concrete shield due to thermal radiation from the pool surface, will be evaluated. After completion of above work, the programming and integration of the models into KAVERN I will be started.

8. Relation with Other Projects

Theoretical Work: RS 57, RS 183, RS 237
Experimental Work: RS 154, RS 160

9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
Jan. 1, 77 - Dec. 31,77	2.3	PNS 4323 (4246)

Vorhaben/Project Title	
Experiments on the Simulation of Large Core Melts (Preliminary Project) Experimente zur Simulation großer Kernschmelzen (Vorprojekt)	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KfK Projekt Nukleare Sicherheit, RBT/IT

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1976		D. Perinić
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1977	

1. General Aim

1.1 Clear quantification of the physical and chronological development of a core meltdown accident in the fourth phase.

1.2 Clarification under which circumstances a representative melt can be retained in the reactor concrete or in the dry concrete bed, in sand, etc.

1.3 Clarification at which time and under which circumstances serious impacts on the environment have to be anticipated from the core meltdown accident.

1.4 Verification of relevant computer codes and extrapolation to reactor dimensions.

2. Particular Objectives

2.1 Investigations into the concrete (melting bed) destruction influenced by

- the thermohydraulics of the melt,
- the thermal power induced,
- the release of vapor and gas,
- the oxidation of the metallic melt, the formation of ceramic melts, the solubility of core melts and melting bed components,
- the material condition of the melting bed,
- the crust and stratum formation,
- the molten pool depth.

2.2 Verification of computer codes, models and theories by combination of significant parameters.

2.3 Study of the long-term behavior of large simulated core melts.

2.4 If appropriate, investigation into the fission product release influenced by

- the combination of thermohydraulics (inclusive of gas bubbles) and the chemical reactions taking place within the melt,
- the amount of melt,
- the ratio of surface to volume of the melt.

2.5 If appropriate, investigations into the release, transport and behavior of aerosols.

3. Research Program

3.1 Summary of the state of knowledge of the physical and chronological development of a core meltdown accident in the fourth phase.

3.2 Development of a melting facility for experiments to be carried out with simulated core melts of 100 to 1000 kg.

3.2.1 Preliminary tests

3.2.1.1 Study of the phenomenology of the core melt-concrete reaction

3.2.1.2 Verification of the accuracy of model assumptions.

3.2.1.3 Development of the melting technology.

3.2.1.4 Development of the measuring technology.

3.2.2 Design and concept of the melting facility.

4. Experimental Facilities

4.1 SASCHA melting facility: induction type furnace for corium melts, previous capacity 0.5 kg at the maximum, extension to 5 kg.

4.2 Cupolar furnace (external): grey cast iron, 5000 kg at the maximum.

4.3 Thermite melting facility 1: 15 kg at the maximum.

4.4 Thermite melting facility 2: 600 kg at the maximum.

4.5 Thermite ignition in concrete: unlimited mass.

4.6 Induction type melting facility for steel/oxide melts (300 kg at the maximum): in the planning process.

5. Progress to Date

ad 3.2.1:

- Evaluation of tests with corium melts (0.5 kg at the maximum) in concrete crucibles.
- Evaluation of the experiments with grey cast iron melts (160 kg at the maximum) in concrete crucibles.
- Preparation, conduct and evaluation of experiments with thermite melts (300 kg at the maximum) in concrete crucibles.

ad 3.2.2:

- Calculation of costs incurred by the melting facility.
- Evaluation of the space requirements.
- Calculation of the pressure waves occurring during a hydrogen and water vapor explosion, respectively, in the safety containment.
- Calculation of the propagation of the melting front in concrete crucibles.
- Calculation of the electricity supply required.
- Design of the electric power supply.
- Conceptional design of the melting facility.

6. Results

ad 3.2.1:

- Separation of the metallic melt from the oxide melt. The metallic melt lies beneath the oxide melt. Hydrogen explosive flames. In flat melts toroidal circulation cell in the center downwards. The oxide melt generates viscous bubbles.

- Maximum melting front velocity measured in the concrete: 40 mm/min in the axial direction, 23.1 mm/min in the radial direction, 61.8 mm/min in the steel reinforcement. A great dependence of the melting front velocity on the molten pool temperature was found

- The melting cavern is pear shaped.

- The melting front is followed by a dehydrated concrete stratum of 25 to 70 mm thickness which, in its turn, is followed by a partly dehydrated stratum of 30 to 132 mm thickness.

- By use of a fiber glass reinforcement and a two-shell design concept of concrete crucibles a reliable crucible configuration was developed.

- Proven measurement and monitoring technique: thermocouples embedded in concrete, two-color pyrometer, total immersion thermometer, contact microphone, gas sampling, vapor detector, remotely operated cinematograph camera.

ad 3.2.2:

- For a corium E-melt of 1000 kg in concrete with a cylindrical mold of the molten pool a net power of about 3 MW has to be generated in the molten pool to retain the melt at the boiling temperatures of the metallic components. At 1 MW power the molten pool reaches a temperature of about 2000°C.

- The electric instrumentation for an induction type facility was dimensioned for a 1 and alternatively 3 MW net power. The supply of such powers seems to be feasible technically also at extremely unfavorable cross-sectional ratios (inductor/molten pool).

- A technical concept was designed. The melt is to be prepared in a crucible above the test crucible proper. The molten liquid corium is admitted into the test crucible by tapping the bottom.

7. Next Steps

- Preparing the final report on the preliminary project.
- Planning of step I of realizing large experiments with induction heated uranium free melts up to 300 kg.

8. Relation with Other Projects

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9. References

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	2.3	RS 183
Vorhaben/Project Title: Energiebilanzen nach hypothetischem RDB-Versagen unter Berücksichtigung der Betonzerstörung Energy Balances after Hypothetical RPV-Failure under Consideration of Concrete Decomposition		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik RZR 2, Erlangen
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 9. 75	31. 3. 78	H. Goetzmann / H. Hassmann
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	31. 12. 77	400.000,-- DM

1. General Aim

In continuation of the theoretical investigations of the energy balances within the pressure vessel and within the containment, the progression of the melt in the concrete structures will be studied.

2. Particular Objectives

A computer code will be developed to describe the destruction of the concrete. Additionally, the energy balances and the pressure increase within the containment will be studied, considering the energy and mass transport to the containment atmosphere.

3. Research Program

3.1 Problem related theoretical investigations:

- Study of the existing knowledge of the available destruction models
- Definition and formulation of the heat transport model.

3.2 Energy balances for the RPV surrounding:

- Definition of the region in which contact with the molten core can occur after hypothetical core melting
- Setting up the energy balances
- Consideration of the conditions which have to be fulfilled in order to keep the molten core as long as possible within the containment.

3.3 Energy balance for the containment after a hypothetical RPV destruction:

- Calculation of the energy and mass transport to the containment wall
- Energy balance and calculation of the pressure increase in the containment.

3.4 Sensitivity study regarding the parameters, which influence the accident course.

4. Experimental Facilities

No experimental facility necessary.

5. Progress to Date

The pressure in the containment was calculated regarding the mass- and energy transport during core heat-up, residual water evaporation after core structure decay, heat-up of the RPV and penetration through the concrete wall.

The program KAVERN I has been improved; the development of an integral code was started which calculates the hypothetical core meltdown in all phases.

6. Results

The meltdown calculations have shown that the penetration through the concrete wall is the most important part of the meltdown process. The result was that the concrete bottom plate of 5.75 m will be penetrated after about 4.5 days.

The containment pressure is lower than 5 - 6 bar when only the gaseous products are considered, which are emitted from the heat-up of the concrete. Higher pressures arise, when the molten core contacts the bottom water. After 1.5 days 5 - 6 bar will be reached.

7. Next Steps

The code development will be continued.

8. Relation with Other Projects

RS 154, RS 166.

9. References

1. Technischer Fachbericht RS 133 (Mai 1977)
2. Technischer Fachbericht RS 183 (Aug. 1977)

10. Degree of Availability

The reports are company confidential.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
Nov.1 - Dec.31, 1977	2.3	RS 293
Vorhaben/Project Title: Theoretical Analysis of the Heat Effects in the Core Melt-Concrete Interaction Theoretische Untersuchungen des Wärmeübergangs bei der Wechselwirkung zwischen Kernschmelze und Beton		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Battelle-Institut E.V., Frankfurt am Main Metals and Composites Division
Arbeitsbeginn/Initiated: Nov. 1, 1977	Arbeitsende/Completed: Oct. 31, 1978	Leiter des Vorhabens/Project Leader: Dr. W. Baukal
Stand der Arbeiten/Status: Continuing	Berichtsdatum/Last Updating: Dec. 31, 1977	Bewilligte Mittel/Funds: 221.424,-- DM

1. General Aim

It is assumed that during the 4th phase of the hypothetical core melt accident " Core Melt-Concrete Interaction " the core melt that has broken through the wall of the reactor pressure vessel penetrates into the concrete of the foundation. The resultant heat effects essentially determine the further progress of the accident and therefore have to be analyzed experimentally and theoretically. The objective of project RS 293 is the theoretical analysis.

2. Particular Objectives

The heat effects are to be determined as " Integral Melting Enthalpy " by breaking down the total reaction into single reaction steps and theoretically analyzing the contribution of each step.

3. Research Program

3.1 Breaking down the total reaction " core melt-concrete interaction " into single reaction steps

3.2 Theoretical determination of the ΔH of each reaction step

3.3 Investigation of the oxidation-reduction behavior in the core melt-concrete system

4. Experimental Facilities, Computer Codes

5. Progress to Date

Ad 3.1 The initial, final and boundary conditions of the analysis (reaction steps 1 to 8 according to Battelle's proposal)were defined.

Ad 3.2 Taking into account the initial, final and boundary conditions, the single reaction steps 1 to 8 were estimated with regard to their expected contribution to the integral melting enthalpy.

Ad 3.3 Those reaction steps were selected whose oxidation - reduction behavior is expected to have the essential influence on the total reaction.

6. Results

Ad 3.1
- Core: A + R
- Concrete: Bn 25o
- Water content: a) 5 mass %
  b) according to a concrete age of about 2o years
- Initial state: concrete 20°C , core melt 1400°C
- Final state: quasi-stationary equilibrium
- Times of analysis:
  a) residual amounts of zirconium present
  b) nickel in the metallic molten phase and melting reinforcing iron as source of additional metal
  c) all-oxide melt

Ad 3.2 The reaction steps 1, 2, 4, 5 and 6, i. e. dehydration and water desorption, water vaporization, reaction between core melt and water vapor, decomposition into single oxides, and melting of the single oxides, are expected to give the main contributions to the integral melting enthalpy.

- Calcite-containing aggregates will be left out of consideration in this context (see reaction step 3).

- The ΔH value being contributed by the dissolution of the molten oxides in the oxidic phase of the core melt (reaction step 7) will be estimated with regard to its significance by comparing it with the contributions of the other reaction steps.

- The heat of vaporization resulting from the vaporization of specific oxides from the core melt (reaction step 8) will not be added to the integral melting enthalpy.

Ad 3.3 The analysis will be restricted to the interaction reactions (oxidation of the different metals by water vapor); if necessary, the Richardson diagram will be revised.

7. Next Steps

Literature study and analysis of relevant publications. In addition to the original proposal of Battelle, the following items will be considered:

- Average water content of concrete at an age of 2o years

- State of the art of the cement chemistry, with consideration of recent research at German universities

- Literature study on the measurement of water-vapor partial pressure over concrete

8. Relation with other Projects

The investigations are being coordinated with project RS 154.

9. References

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1o. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	2.3	RS 295
Vorhaben/Project Title: Wechselwirkung der Kernschmelze mit erweitertem Fundamentbereich Interaction of Core Melt with Extended Foundation Region		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik RB 3, Erlangen
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 12. 77	30. 9. 79	Dr. Peehs
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	31. 12. 77	675.000,-- DM

1. General Aim

To provide a risk analysis for LWR-accidents of very low probability, the research project "Kernschmelzen" of the BMFT was started in the FRG in 1971. The objective of the related R and D-work is to study the consequences of a hypothetical core melting accident. This may occur if we hypothesize that the emergency core cooling system fails completely after a loss-of-coolant accident. Performing this analysis, one can define the chronological order in the sequence of a core melting accident:

1) the heating up of the core until the core structure may fail; which starts at a certain water level in the core and ends with the failure of the grid plate,

2) the second phase is characterized by the evaporation of the water left in the lower plenum and it lasts after the dryout of the pressure vessel until a molten core debris is formed,

3) the third phase is concerned with the heating up of the pressure vessel after the melt was formed and

4) in the fourth phase after the pressure vessel failure, the molten Corium will interact with the concrete structure beneath the pressure vessel,

5) finally after the penetration of the concrete foundation

of the reactor the molten Corium plus the molten Concrete starts to interact with the materials just beneath the concrete structure.

2. Particular Objectives

Subsequent to penetration of the bottom concrete layer the hot core melt will directly contact the geological strata found underneath the concrete reactor foundation. However, it is not realistic to assume that, when the melt reaches these strata, and instantaneous release of the core melt fission product inventory occurs. It should rather be assumed that the extended foundation similar to the concrete, will be heated up by the core melt, causing a series of melting processes, which again leads to a crust formation between the melt and the geological strata.
This crust should now prevent in some extent the propagation on fission product into the geological strata beneath the concrete foundation and into the ground water.
It is the objective of the R+D-work to give an experimental base for theoretical modelling of the 5th phase of a hypothetical core melt accident.

3. Research Program

3.1 Definition of the status of molten corium after penetration of the reactor foundation.

3.2 Compilation of the status of the extended foundation region interacting with the molten corium.

3.3 Literature study on the performance of typical natural aggregate material and geological strata.

3.4 Determination of experimental boundary conditions.

3.5 Testing of the available experimental techniques to cover the experimental requirements.

3.6 Test of the pure aggregate material

3.7 Investigation of core melt/mineral aggregate interaction

3.8 Test evaluation in order to prepare a core melt/extended foundation interaction model

3.9 Data verification by means of RS 183 computer program in order to provide an estimate of the extended foundation penetration.

4. Experimental Facilities

Experimental work will be performed with the equipment already installed for RS 74 a and RS 154, i.e. melt quantities of 1 to 2 kgs will mainly be used. The max. melt quantity, however, will be increased up to 5 kgs, if possible.

5. Progress to Date

As was the case with all previous relevant core melt tests, the melts (approx. 5 kgs) will be heated up by an electric arc device.

Construction of the melting device was completed and its assembly initiated.

6. Results

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7. Next Steps

After completion of preliminary tests for checking the melting device, a detailed test program will be prepared.

8. Relation with Other Projects

RS 154, RS 183

9. References

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10. Degree of Availability ---

	Classification 2.3
Title 1 CONTROL OF MOLTEN CORE DEBRIS (1)	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION CULHAM LABORATORY
Title 2	Project Leader
Initiated 1972 Completed : Status : Last updating	Scientists:

Description:

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 programme 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 Reineke, 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.

Classification

2.3

Title 1	COUNTRY UNITED KINGDOM
CONTROL OF MOLTEN CORE DEBRIS (2)	SPONSOR UKAEA
	ORGANIZATION AERE HARWELL
Title 2	Project Leader R G BELLAMY
Initiated 1972 Completed :	Scientists:
Status : Last updating	

Description:

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

Two experimental rigs using weak acids and ohmic heating have been operated. 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. A third larger rig constructed to enable Rayleigh numbers appropriate to molten UO2 to be attained met with difficulties due to attack on the heaters by the acid but high Rayleigh numbers were achieved with the second rig by using higher power densities. A rig using low melting point lead alloy eutectics heated by an array of immersion heaters has been constructed.

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.

The basic heat transfer experiments have been completed and have given a much better understanding of the way in which melted out fuel caught in horizontal trays would lose heat to the surrounding coolant. A paper on the work will be presented at an international conference on Turbulent Bouyant Convection to be held in Yugoslavia in August 1976.

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Classification

Title 1	COUNTRY
	SPONSOR
	ORGANIZATION
Title 2	Project Leader
Initiated Completed :	Scientists:
Status : Last updating	

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 it 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.

3. EXTERNAL INFLUENCES

122-1 -01/4170-10
222-1 -01

3

Titre Identification et caractérisation des agressions liées aux phénomènes naturels extrêmes.	**Pays** FRANCE
	Organisme directeur CEA/ DSN
Titre (anglais) Identification and description of extreme natural events.	**Organisme exécuteur** CEA/DSN/SESRS
	Responsable DSN/SESRS/FONTENAY

Date de démarrage	1/01/77	**Etat actuel**	En cours	**Scientifiques**
Date prévue d'achèvement	1/01/82	**Dernière mise à jour**	12/77	

1 - OBJECTIF GENERAL
Identification des phénomènes naturels extrêmes susceptibles d'affecter la sûreté des installations nucléaires, détermination de leurs caractéristiques et de leur probabilité d'occurence.

2 - OBJECTIFS PARTICULIERS
2.1
Etablissement de dossiers d'information concernant les principaux cours d'eau (bassins d'alimentation, ouvrages d'aménagement des eaux). et les principales régions côtières (fonds marins, reliefs, ...).

2.2
Etablissement de fichiers concernant les évènements extrêmes historiques dans les domaines de :
- l' hydrologie continentale (inondations, étiages...)
- l'hydrologie marine(raz-de-marée, ondes de tempête, marées exceptionnelles ...)
- météorologie (tempêtes, tornades, précipitations...)
- géologie (glissement de terrain, effondrements, liquéfaction des sols...)
- Traitement statistique des données obtenues.

2.3
- Mise en évidence des relations causales existant entre les différents phénomènes naturels (ou artificiels), (raz-de-marée d'origine sismique, inondations dues aux précipitations, seiches, mascarets, ruptures de barrages d'origine sismique ou hydrologique ...).

2.4
- Définitionsdes évènements naturels extrêmes à prendre comme référence pour la sûreté (caractéristiques associées à une probabilité ou phénomène "enveloppe").

.../...

4 - ETAT DE L'ETUDE :

4.1 - Avancement à ce jour

- Le recueil d'informations concernant l'hydrologie continentale est en voie d'achèvement.
- Le recueil d'informations concernant l'hydrologie marine et la météorologie est commencé.

4-2 - Résultats essentiels

Depuis Septembre 1977, les informations recueillies en hydrologie continentale sont utilisées pour les analyses de sûreté.

5 - PROCHAINES ETAPES

A court et moyen terme (1978)

- Recueil des informations générales en hydrologie continentale et marine.
- Commencement des fichiers concernant les évènements extrêmes historiques dans les domaines hydrologiques, météorologiques et géologiques.
- Caractérisation des phénomènes complexes (raz-de-marée, seiche ...).
- Définition des évènements de référence dans le domaine de l'hydrologie continentale.

A long terme (1979 - 1980)

- Définition des évènements de référence en hydrologie marine et en météorologie.
- Définition des évènements de référence en géologie.

6 - RELATION AVEC D'AUTRES ETUDES

- 121-1-02 : Carte sismotectonique de la France.
- 121-1-05 : Collecte..... de mesures sur les mouvements en zone épicentrale et d'informations sur les dégâts correspondants.

122-2 -01/4170-10 222-2 -01		3
Titre Identification et caractérisation des agressions externes liées aux activités humaines. Détermination de leur probabilité d'occurrence		Pays FRANCE Organisme directeur CEA/DSN
Titre (anglais) Identification and description of man - induced events - Determination of their probability of occurrance		Organisme exécuteur CEA/DSN/SESRS Responsable DSN/SESRS/FONTENAY
Date de démarrage 1/01/1975	Etat actuel En cours	Scientifiques
Date prévue d'achèvement 31/12/82	Dernière mise à jour 12/77	

1 - OBJECTIF GENERAL
Identification des sources d'agression potentielle issues des activités humaines et susceptibles d'affecter la sûreté des installations nucléaires.
Détermination des caractéristiques et des probabilités des phénomènes extrêmes susceptibles d'être engendrés par ces sources.

2 - OBJECTIFS PARTICULIERS
2.1
Etablissement de dossiers d'information sur les industries, les réseaux de transports de fluides, les stockages et les voies de communications concernés par des produits dangereux (inflammables, explosifs, toxiques, corrosifs...).
2.2
Etablissement de fichiers concernant des accidents caractéristiques de ces différentes activités. Analyse des causes et des effets relatifs à ces accidents.
2.3
Détermination des caractéristiques et des probabilités des évènements potentiels menaçant la sûreté des installations nucléaires (onde de choc, projection de missile, chaleur, fumée et poussières, dégagement de gaz inflammable ou explosif, inondation ou perte d'eau, subsidence ...).

- Activités humaines étudiées
- Industries chimiques diverses, pétrochimie, ...
- Stockage de produits dangereux (corrosifs, toxiques, explosifs, inflammables ...)
- Transports de fluides dangereux par canalisation (notamment gazoducs, oléoducs, oxyducs ...)
- Industries minières, carrières ...
- Aménagement des cours d'eau (notamment retenue des eaux)
- Activités militaires.(stockage de munitions et poudrières, champs de tir ...)
- Circulation aérienne, maritime, fluviale, routière, par voie ferrée.

4 - ETAT DE L'ETUDE

4.1 - Avancement à ce jour

- La constitution de dossiers d'information sur les stockages pétroliers et les réseaux de gazoducs et d'oléoducs est en voie d'achèvement.
- Des données partielles concernant les chutes d'avions militaires (armée de l'air) sont en exploitation.
- Des données essentielles sur le trafic de l'aviation commerciale ont été acquises à l'issue de contacts avec les organismes régissant la navigation aérienne (U.N.A., D.R.A.C., A.P., I.T.A.) et leur exploitation est commencée.
- La constitution d'un fichier concernant les grands sinistres (explosions et incendies) survenus à des installations industrielles et à des moyens de transports est en cours.

4.2 - Résultats essentiels

- Les dossiers d'informations sur les stockages, les réseaux de transports et le trafic aérien sont déjà utilisés pour contrôler, préciser ou compléter les indications contenues dans les rapports de sûreté.
- Des études spécifiques de dispersion de nuages toxiques ou explosifs ont été réalisés.

5 - PROCHAINES ETAPES

- A court terme (1978)

. Achèvement des collectes d'information en cours (produits pétroliers, navigation aérienne).

. Constitution de dossiers d'information concernant :
  - les transports dangereux par voies fluviale, maritime, routière et ferrée
  - les industries et les stockages de produits dangereux autres que pétroliers
  - l'aviation générale et militaire (aéronavale et armée de terre).

. Exploitation du fichier "grands sinistres" pour l'établissement de scénarios d'accidents et la détermination de distances de sécurité.

- A long terme (1979)

. Détermination des paramètres caractéristiques et des probabilités des évènements extrêmes, dus à l'environnement industriel et aux voies de communication.

. Mise au point de méthodologies simples et standardisées utilisant ces éléments et susceptibles de servir de base à une future réglementation.

6 - RELATION AVEC D'AUTRES ETUDES

. 122-2-02 : Agressions d'origine externe sur les installations nucléaires explosions chimiques non confinées dûes à un environnement industriel ou aux voies de communication.

. 122-2-03 : Formation et dispersion de nappes de gaz dérivantes, explosives ou toxiques suite à une fuite massive sur un transport ou un stockage de produits chimiques.

. 120-1-01 : Etude des caractéristiques démographiques des sites sous l'angle de la sûreté, établissement de critères de classement et de sélection.

7 - DOCUMENTS DE REFERENCE disponibles :

- "Etudes récentes concernant les principales agressions d'origine externe et recommandations de sûreté adaptées à la situation européenne", A. DOURY, A. BARBREAU, R. GERARD - Rapport DSN R 83, Octobre 1975;

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Title 1 (Original language)	Classification
Statistical analysis of randome signals	II - 3 - 4 - 8 IO - I4
Title 2 (English)	Country ITALY Sponsor / Organisation CNEN
Date initiated 1966 Date completed in progress Last updating April I977	Project Leader A. Federico

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1977	3.1	RS - 170

Vorhaben/Project Title	
Seismische Kriterien zur Standortauswahl kerntechnischer Anlagen in der BRD Seismic Risk Maps for Nuclear Power Plants in the Federal Republic of Germany	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: BGR SZGRF

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.1976	31.12.1977	Dr. Harjes
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
completed	10.1.1978	98.000,-- DM

1. General Aim

The project undertakes a seismic regionalization of the Federal Republic of Germany. For this purpose, the observed earthquakes (historical descriptions as well as instrumental recordings) will be mapped with different parameters. These maps facilitate the selection of sites for nuclear power plants with respect to seismic risk.

2. Particular Objectives

A complete and uniform data base is an essential condition for estimating the occurrence rate of dangerous earthquakes. This data base is an earthquake catalogue with all available relevant information in computer-readable format.

Earthquakes are prominent indicators of tectonic earth movements in space and time. Therefore, recurrence rates and maximum events have to be investigated individually for each seismic region. A combination of all these regional activities together with a statistical model of earthquake occurrence gives a realistic estimate of seismic risk at a specified site.

3. Research Program

3.1. Completion of the earthquake catalogue.

3.2. Determination of average focal depths and representative absorption coefficients for each seismic region.

3.3. Computation of macroseismic maps and intensity distributions.

3.4. Assignment of intensity to horizontal acceleration.

3.5. Computation of magnitude (intensity)-frequency relations.

4. Experimental Facilities, Computer Codes

Several computer programs have been developed to evaluate the earthquake catalogue and to present the results:

4.1. Maps of earthquake epicentres for the F.R.G. with geographical details.

4.2. Contour maps of

a) maximum intensities

b) number of earthquake intensities I_k

c) maximum horizontal accelerations

4.3. Magnitude (intensity)-frequency curves for geologically different regions

4.4. Data base management system (sort, merge, search routines)

5. Progress to Date

The collection of earthquake data is finished and a catalogue was distributed to all German Seismological observatories for critical review. This catalogue contains about 800 events during the time-period 1000-1974 in a computer-readable format. Maps of earthquake epicentres for different time-periods have been plotted with geographical details.

To facilitate detailed investigations, each earthquake is attached to a seismic region. So it is very simple to estimate average focal depths as characteristic values for each seismic region if there are no exact measurements.

6. Results

To illustrate the effect of earthquake activity, theoretical intensity distributions of all seismic events were computed and summed within grid points of 5 km distance. Then contours of maximum intensity were plotted. Adequate relations between intensity and horizontal acceleration were established to map this very important parameter for engineering purposes. These maps facilitate to estimate the seismic risk for a specified site and they show the regional and timely variation of seismic actitivity.

The spatial distribution allows to define seismic provinces whereas recurrence rates can be derived from the time sequence. This was done by evaluating intensity-frequency curves for different seismic provinces.

Combined with maximum magnitudes this concludes the seismic risk analysis as it stands now.

142-1 -01/4113-01		3.1
Titre Analyse parasismique d'une centrale nucléaire. Bâtiment réacteur PWR 900. Méthode de calcul.		Pays FRANCE Organisme directeur CEA/DgCS
Titre (anglais) Seismic analysis of a nuclear power plant. PWR 900 reactor building.		Organisme exécuteur CEA/DEMT Responsable DEMT - Saclay
Date de démarrage 1/1/75	Etat actuel terminée	Scientifiques (DEMT)
Date prévue d'achèvement 31/12/77	Dernière mise à jour 1/12/77	

1 - Objectif général :

Le but de cette étude est la mise au point d'un programme de calcul permettant d'évaluer la réponse d'une tranche de centrale PWR 900 à une excitation sismique donnée, caractéristique du site de la centrale. Le programme devra être transposable d'un site à un autre, même dans le cas de légères modifications de réalisation.

2 - Objectifs particuliers :

1) Modélisation des structures.

2) Analyse modale des constituants.

3) Méthodologie de prise en compte des couplages.

4) Evaluation des spectres de planchers.

.../

3 - Installations expérimentales et programme

4 - Etat de l'étude :

1) Avancement à ce jour :

ETUDE D'UNE CENTRALE PWR 900. L'étude de comportement et de la tenue au séisme d'une enceinte de confinement du type contrat programme a été effectuée.

2) Résultats essentiels :

Cette étude s'est attachée en particulier à préciser l'évolution, en fonction de la raideur du sol de fondation, des modes et fréquences de vibration de l'ensemble sol-structure et de leur amortissement (Rapport en préparation).

6 - Relations avec d'autres études

7 - Documents de référence

Rapport en préparation.

142-1 -02/4113-01 152-1 -02/4113-01			3.1
Titre Analyse parasismique d'une centrale nucléaire, interaction sol-fondation.			Pays FRANCE Organisme directeur CEA/DgCS
Titre (anglais) Seismic analysis of a nuclear power plant soil-structure interaction.			Organisme exécuteur CEA/DEMT-Saclay Responsable DEMT - Saclay
Date de démarrage	1975	Etat actuel en cours	Scientifiques
Date prévue d'achèvement	1978	Dernière mise à jour 12/77	

1 - Objectif général :

Recherche d'une méthode pour établir la relation entre le signal sismique en champ libre et les mouvements d'un radier rigide sous la forme d'une fonction force-déplacement.

2 - Objectifs particuliers :

1) Relation force-déplacement pour un seul radier.

2) Application au cas de radiers multiples.

3 - Installations expérimentales et programme

.../

4 - Etat de l'étude :

1) Avancement à ce jour :

Dans l'hypothèse d'une centrale posée sur un sol semi-infini homogène élastique, on peut calculer analytiquement, en fonction de la fréquence, le déplacement d'un point quelconque de la surface sous l'effet d'une force exercée en un autre point.
Basé sur ce principe, un programme est en cours de mise au point pour déterminer l'impédance de sol (c'est-à-dire la relation complexe force-déplacement en fonction de la fréquence) correspondant aux divers mouvements d'un radier rigide.

2) Résultats essentiels :

Examen de l'interaction sol-structure dans le cas d'un bâtiment réacteur CP 1.

5 - Prochaines étapes :

Application du même programme au calcul de l'interaction :

- entre deux radiers voisins de forme quelconque.
- entre plusieurs radiers voisins de forme quelconque.

6 - Relations avec d'autres études

7 - Documents de référence :

Rapport à paraître.

142-1 -03/4113-01 152-1 -03		3.1 * 10
Titre Tenue de structures - types sous excitation sismique. Essais sur table vibrante.		Pays FRANCE Organisme directeur CEA /DgCS
Titre (anglais) Behaviour of typical structures under seismic excitation. Shake table tests.		Organisme exécuteur CEA/DEMT - Saclay Responsable (DEMT-Saclay)
Date de démarrage 1/75	Etat actuel en cours	Scientifiques
Date prévue d'achèvement 12/78	Dernière mise à jour 12/77	

1 - Objectif général :

L'objet principal de cette étude est d'approfondir la connaissance des limites de résistance et du processus de ruine en régime dynamique des éléments de structures utilisées dans la construction des centrales nucléaires. Subsidiairement, les essais sur des éléments représentatifs de construction traditionnelle devront permettre de fonder une meilleure corrélation entre les intensités macrosismiques et les paramètres mécaniques utilisés en ingénierie.

2 - Objectifs particuliers :

1) Essais de structures traditionnelles.
2) Essais d'éléments en béton armé ordinaire - Flexion : poteaux.
3) Essais d'éléments en béton armé ordinaire - Cisaillement : voiles.
4) Essais d'éléments en béton armé composites (en L, etc.)
5) Essais d'éléments de béton précontraint.

.../

3 - Installations expérimentales et programme :

Table vibrante VESUVE (DEMT).

4 - Etat de l'étude :

1) Avancement à ce jour :

1 - Campagne d'essais exécutée sur murs non contreventés, chargés.

a) Briques et mortier classés " anciens ".

b) Briques et mortier classés " modernes ".

2 - Campagne d'essais sur poteaux de béton.

2) Résultats essentiels :

1 - Murs.

Connaissance des fréquences propres et de leur variation au cours de la dégradation du mur. Amortissements en essais de lâcher et variation des amortissements.

2 - Poteaux en béton armé.

Comportement dynamique des poteaux en flexion. Connaissance de la longueur effective (encastrements) et comparaison avec l'approche par sections équivalentes du béton fissuré.

5 - Prochaines étapes :

Voiles en béton armé travaillant dans leur plan principal.
Comportement dynamique en cisaillement.
Calculs préliminaires : estimation des efforts nécessaires pour des effets significatifs. Etude de l'effet du rapport hauteur-longueur.
Essais statiques.
Essais dynamiques.

7 - Documents de référence : rapports internes non disponibles .

121-1 -01/4113-01		3.1
Titre Signaux sismiques synthétiques.		Pays FRANCE Organisme directeur CEA
Titre (anglais) Synthetic seismic signal studies.		Organisme exécuteur CEA/DEMT Responsable M. LIVOLANT DEMT - Saclay
Date de démarrage 1/1/75	Etat actuel en cours	Scientifiques F. JEANPIERRE - DEMT
Date prévue d'achèvement 31/12/78	Dernière mise à jour 1/12/77	

1 - Objectif général :

Constructions de signaux temporels synthétiques vraisemblables, avec des spectres réguliers.

2 - Objectifs particuliers :

1) Analyse détaillée d'enregistrements sismiques disponibles.
2) Mise au point de spectres réguliers.
3) Calculs de validation.

3 - Installations expérimentales et programme

.../

4 - Etat de l'étude :

1) Avancement à ce jour :

Une analyse détaillée des accélérogrammes de séismes réels a été faite ; le but de cette étude est de dégager les paramètres caractéristiques de ces accélérogrammes (durée, évolution du contenu spectral au cours du temps, etc.) de façon à construire des séismes synthétiques vraisemblables.

2) Résultats essentiels :

L'étude est terminée actuellement pour le séisme de SAN FRANCISCO.
Un séisme synthétique de caractéristiques semblables mais avec des spectres réguliers a été mis au point. Une série de spectres-types a été mise au point à partir de séismes américains.

5 - Prochaines étapes :

Validation sur enregistrements de caractéristiques européennes.

6 - Relations avec d'autres études

7 - Documents de référence :

- Rapport en préparation.

121-1 -02/4172-10		3.1
Titre Carte sismotectonique de la France.		Pays FRANCE Organisme directeur SCSIN-CEA/DSN-EDF-BRGM.
Titre (anglais) The seismotectonic map of France.		Organisme exécuteur BRGM - CEA/LDG. Responsable Comité présidé par M. GOGUEL - BRGM
Date de démarrage 01/76	Etat actuel en cours	Scientifiques MM. VOGT (BRGM) MASSINON (LDG) BARBREAU (DSN) FAURE (DSN)
Date prévue d'achèvement 12/78	Dernière mise à jour 11/77	

1 - Objectif général :

Elaboration d'un document exhaustif rassemblant les données géologiques, géophysiques et sismiques afin de disposer d'une synthèse des connaissances sismotectoniques concernant la France. Ce document est essentiel à l'évaluation des risques sismiques des sites nucléaires et à la définition des séismes de référence.

2 - Objectifs particuliers :

Permettre la meilleure évaluation possible de la sismicité des sites par :

1) Collation des données de géologie structurale, de sismicité historique et de sismicité instrumentale.

2) Synthèse pour définir les provinces sismotectoniques et les accidents sismogènes.

3) Accumulation des éléments d'évaluation des séismes de référence (localisation des épicentres, intensités, profondeur...).

.../

3 - Installations expérimentales et programme :

Pour la sismicité instrumentale :

- réseau de détection sismique du Laboratoire de Détection Géophysique.
- réseau de l'Association Française de Sismologie Expérimentale.
- données des stations DSN (Cadarache).

4 - Etat de l'étude :

1) Avancement à ce jour :

- Collation des données réalisée à environ 80% dans tous les domaines.
- Synthèse amorcée au niveau de chaque région.
- Possibilité d'utilisation des divers fichiers de données.

2) Résultats essentiels :

- Publication d'une esquisse sismotectonique régionale (Provence).
- Première utilisation des données rassemblées pour les analyses de site.

5 - Prochaines étapes :

Fin de la collation des données.
Réalisation de la synthèse.

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121-1 -03/4172-10		3.1.
Titre Méthodologie pour le calcul des spectres des séismes de référence des sites à partir des paramètres physiques.		Pays : FRANCE Organisme directeur CEA/DgCS/DSN
Titre (anglais) Methodology for the calculation of reference earthquake spectra of vibratory ground motion for sites using physical parameters.		Organisme exécuteur CEA/DSN Responsable DSN/SESRS - FONTENAY
Date de démarrage : 01/01/1976	Etat actuel : en cours	Scientifiques H. FERRIEUX G. MOHAMMADIOUN
Date prévue d'achèvement : 31/12/1979	Dernière mise à jour : 15/11/1977	

1- Objectif général:

Détermination des mouvements sismiques de référence pour un site d'installation nucléaire.

2- Objectifs particuliers:

- Calcul des spectres de référence du site en fonction des paramètres physiques (magnitude, distance focale, loi d'atténuation des ondes), à l'aide d'une analyse statistique prenant en compte des enregistrements réalisés dans des conditions analogues: . modèles sismotectoniques semblables,
  . géologie comparable du site.
- Etude des mécanismes au foyer et de leur influence sur le mouvement en zone proche.
- Etude des lois de transmission locales des différentes ondes.

3- Installations expérimentales et programme:

- Voir fiche 121-1 05 pour l'obtention des enregistrements en zone proche.

Détermination des coefficients d'atténuation en fonction de la distance à partir des enregistrements obtenus sur un profil grâce à des sources artificielles ou à des séismes naturels.

Etude expérimentale des fonctions de transfert locales.

Etude du mécanisme au foyer et des lois de transmission des ondes à l'aide des données du FRIOUL.

4- Etat de l'étude:

1. Avancement à ce jour:

. Une relation statistique liant les spectres à la magnitude et à la distance focale a été déterminée.

. Analyse des données de KARNIK concernant les séismes européens pratiquement terminée.

. Etude des relations entre l'intensité à l'épicentre, la profondeur du foyer et la magnitude. En cours .

2. Résultats essentiels:

L'analyse statistique montre les coefficients d'atténuation différents de ceux de la Californie et voisins de ceux de l'Utah.

L'influence de la magnitude sur la forme spectrale apparaît prépondérante dans cette étude.

On observe l'influence de la nature du sol au point d'enregistrement sur la durée du signal.

5- Prochaines étapes:

Amélioration des méthodes de modélisation des spectres par introduction de nouvelles données dans la détermination des coefficients utilisés: étude de l'influence des paramètres à la source sur le mouvement sismique (chute de contrainte, vitesse de rupture, longueur de faille, modèle de dislocation).

6- Relation avec d'autres études:

"Etudes des phénomènes sismiques en zones proches". Ces études apportent des informations importantes au calcul des spectres synthétiques correspondant aux enregistrements de séismes à courte distance.

7-Documents de référence:

"Les études sismologiques effectuées au CEA dans le domaine de la sûreté des sites nucléaires", A. BARBREAU, H. FERRIEUX, B. MOHAMMADIOUN - AIEA, Vienne 1975.

"Etudes sismologiques effectuées en vue de la protection des installations nucléaires", Communication présentée à la réunion des spécialistes sur la conception antisismique des installations nucléaires organisée par le CSIN et l'OCDE PARIS 1-3/12/1975.

"Etude des répliques du séisme du 6 mai 1976 au FRIOUL", A. BARBREAU, B. MOHAMMADIOUN, H. FERRIEUX, G. MOHAMMADIOUN - OCDE - ROME 11/13 octobre 1977.

8-Degré de disponibilité

Disponibles

121-1 -04/4172-10		3.1
Titre Surveillance de la sismicité des sites nucléaires et de l'activité des failles.		Pays FRANCE Organisme directeur CEA/DgCS/DSN
Titre (anglais) Instrumental monitoring of seismicity and fault activity surrounding nuclear sites.		Organisme exécuteur CEA/DSN Responsable DSN/SESRS - FONTENAY
Date de démarrage : 01/01.1976	Etat actuel : Etude en cours	Scientifiques
Date prévue d'achèvement: 31/12/1981	Dernière mise à jour : 15/11/1977	

1- Objectif général:

L'objectif de cette étude est la recherche d'une meilleure connaissance de la sismicité aux alentours d'un site nucléaire en vue de l'évaluation du risque sismique.

2- Objectifs particuliers:

1. Etude de l'activité des failles au voisinage d'un site nucléaire par la surveillance de l'activité sismique.

2. Détermination descaractéristiques spécifiques du site en ce qui concerne la transmission des ondes sismiques (enregistrement sur la roche dure, sur les alluvions, etc...).

3. Prévision des mouvements de référence à prendre en compte pour le calcul ou pour la vérification du dimensionnement.

3- Installations expérimentales et programme:

- Observatoire de Cadarache : Equipé de différents types d'appareils de mesure (accéléromètres, capteurs de vitesse, capteurs de déplacement) et disposant d'une dynamique étendue en amplitude et en fréquence.

- Réseau de surveillance de Pierrelatte comprenant deux stations de mesure, dont une comporte des capteurs au fond d'un forage de 80 mètres.

4- Etat de l'étude:

1. Avancement à ce jour:

. Surveillance de l'activité sismique de la région de Cadarache.

. Surveillance de l'activité sismique de la région de Pierrelatte.

2. Résultats essentiels:

Station de CADARACHE: Mise en évidence des activités des failles de Jouques et de Beaumont de Pertuis près de Cadarache et de la région de Gardanne.

Enregistrement de séismes alpins et des séismes européens importants. Ex. FRIOUL.

Toutes ces données sont communiquées aux organismes intéressés:

. Centre Sismologique Européo-Méditerranéen (CSEM) FRANCE.
. National Earthquake Information Service (USA).
. International Seismological Centre (ISC) Newbury GB.

Station de PIERRELATTE: Enregistrements de petits séismes locaux montrant une activité de failles (région d'Aiguebelle).

Détermination de fonctions de transfert locales et de lois d'atténuation .

Spectres de référence adaptés au site du Tricastin à l'aide des enregistrements effectués.

5- Prochaines étapes:

Poursuite de la surveillance des deux sites précités.

Surveillance éventuelle d'autres sites à sismicité importante.

6- Relation avec d'autres études:

Carte sismotectonique de la France:
- Prévisions des spectres de référence.

. Utilisation des données dans l'élaboration de la carte sismotectonique.

7- Documents de référence:

Internes

121-1 - 05/4172-10		3.1
Titre Collecte de mesures sur les mouvements en zone épicentrale et d'informations sur les dégâts correspondants.		Pays FRANCE
		Organisme directeur CEA/ DSN
Titre (anglais) Collection of recorded data concerning motion in the near field and of information about corresponding damage.		Organisme exécuteur CEA/DSN
		Responsable DSN/SESRS -FONTENAY
Date de démarrage : 01/01/1976	Etat actuel : en cours	Scientifiques
Date prévue d'achèvement: 31/12/1981	Dernière mise à jour : 11/1977	

1- Objectif général:

Connaissance des caractéristiques des mouvements sismiques en zones proches et de leurs effets en vue de la protection des installations nucléaires contre les séismes.

2- Objectifs particuliers:

1. Rassemblement des données instrumentales mondiales concernant les mouvements forts (établissement d'une sismothèque - fin 1978).

2. Enregistrement de séismes en zone proche à l'aide des dispositifs DSN.

3. Analyse de ces données en fonction des caractéristiques des séismes (magnitude, mécanisme au foyer) et des particularités des sites d'enregistrement (1980). Etude de la répartition spectrale, du niveau d'intensité sismique en corrélation avec les caractéristiques du séisme (magnitude, distance focale et atténuation), de la relation entre l'intensité macrosismique et les paramètres physiques, (accélération, vitesse, déplacement).

3- Installations expérimentales et programme:

Stations sismologiques légères destinées à l'enregistrement des séismes et de leurs répliques dans la zone épicentrale.

- Installation d'appareillage dans des zones à forte sismicité.

- Etude de répliques après un tremblement de terre important.

- Réalisation d'une expérimentation sur un site à partir de sources sismiques artificielles de forte puissance afin d'étudier le comportement du sol et des matériaux dans le domaine non linéaire.

4- Etat de l'étude:

1. Avancement à ce jour:

- Les enregistrements des nombreuses répliques (plusieurs centaines) dans la région du FRIOUL ont été exploités. Les résultats ont fait l'objet d'une communication à l'OCDE (Réunion sur la conception antisismique des installations nucléaires, ROME, octobre 1977).

- Les stations de Saint Paul sur l'Ubaye et de Bagnères de Bigorre installées au cours de l'année 1977, ont enregistré des petits séismes locaux.

- Un contact a été établi avec l'United States Geological Survey pour obtenir les copies d'enregistrements de mouvements forts que l'USGS détient(enregistrements américains, japonais, russes etc.).

- Des enregistrements des séismes du FRIOUL et de ROUMANIE obtenus par le réseau Yougoslave ont été fournis.

- Une synthèse des résultats des expérimentations effectuées au Tricastin en 1974 et 1975, déjà exploités partiellement, fait l'objet d'un rapport en cours de rédaction (1er semestre 1978).

2. Résultats essentiels:

L'étude du FRIOUL a permis:

- de quantifier la variation du spectre en fonction de la magnitude et de la distance focale (les spectres deviennent riches en basses fréquences quand la magnitude et la distance augmentent). L'influence sur la forme spectrale de la magnitude est mieux marquée dans le cas du FRIOUL que dans le cas des séismes de Californie.

- l'influence de la nature du sol au lieu d'enregistrement sur la durée du mouvement sismique,

- un rapport entre les mouvements verticaux et horizontaux, toujours inférieur à 1,

- des lois d'atténuation,en fonction de la fréquence,spécifiques du site.

Ces résultats ont permis d'élaborer une analyse statistique liant le paramètre du mouvement (pics d'accélération, de vitesse, de déplacement) à la magnit et à la distance focale.

5- Prochaines étapes:

- Poursuite de l'établissement de la sismothèque à partir de données étrangères.

- Poursuite des enregistrements des séismes dans les zones proches: Saint Paul sur l'Ubaye, Bagnères de Bigorre, Cadarache.

- Analyse des données.

6- Relations avec d'autres études:

Amélioration de la prévision des spectres en zones proches. Cette étude est étroitement liée avec l'étude "Prévision des spectres de référence" à laquelle elle apporte un certain nombre de données utiles pour la définition des paramètres en zones proches.

7- Document de référence: disponible

Etude des répliques du séisme du 6 mai 1976 au FRIOUL par A. BARBREAU, B. MOHAMMADIOUN, H. FERRIEUX, G. MOHAMMADIOUN - OCDE - ROME 11/13 octobre 1977.

Title 1 (Original language)	Classification
Rete di rilevamento sismico	3.1
Title 2 (English)	**Country** ITALY
Seismic monitoring network	Sponsor ENEL Organisation ENEL
Date initiated 1973 Date completed - Last updating April 1977	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 168 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

The seismic network has been completed. The first time it operated was on the occasion of the earthquake occurred in Friuli on May 1976.
Furthermore a computer program has been developed which enables to obtain the seismic spectra (acceleration, velocity and displacement) and their envelopes.
The first results of data processing have been published by ENEL-CNEN Commission for the study of problems connected with the construction of nuclear power plants.

5. Next steps

Further recording will be necessary to obtain the definition of reference earthquake.

02

5 02

Title 1 (Original language)	Classification
Rete di rilevamento sismico	3.1

6. Relation to other projects

   Joint Commission CNEN/ENEL to study the seismicity of Italian territory for future nuclear power plants.

7. Reference documents

   Reports on the 1976 Friuli earthquake are available.

Title 1 (Original language)	Classification
Ricerche di sismotettonica	3.1
Title 2 (English) Seismotectonic researches	Country ITALY Sponsor Organisation CNEN
Date initiated January 1975 Date completed In progress Last updating April 1977	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.

   Studies on Friuli seismotectonic features have been performed.

Related projects

3.1 (other programs: CNEN, ENEL).

Title 1 (Original language)	Classification
Sviluppo di strumentazione e misure sismiche per la valutazione dei siti	3.I
Title 2 (English)	**Country** ITALY
Seismic instruments development and seismic measurements for site evaluation	Sponsor / Organisation } CNEN
Date initiated May 1974 (present phase) Date completed In progress Last updating March 1977	Project Leader R. CERVELLATI

General aim

Seismic instruments development and seismic measurements, in order to have a characterization of sites from the seismological point of view.

Particular objectives

Development of a digital accelerometer; operation of accelerometers; setting up and operation of seismometric equipments; analysis of the response of seismometric instrumentation.

Experimental facilities

A live network of accelerometers. An electronic shop for maintenance and calibration.

A mobile seismic laboratory.

Project status

I) 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 during strong earthquakes shocks. The "time-histories" will be employed in the characterization of the design earthquake.

2) Seismometric equipments have been set up and are operated, in order to obtain a contribution to characterization of sites from the seismological point of view (determination of the earthquake mechanisms, hypocenters, etc.).

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Title 1 (Original language)	Classification
Sviluppo di strumentazione e misure sismiche per la valutazione dei siti	3.I

3) A mobile seismic laboratory has been set up. It is presently in operation in Friuli (Northern Italy), since the May 6th, 1976 catastrophic earthquake.

4) A prototype digital accelerometer is developed in cooperation with Pisa University.

Next steps

Introduction of timing coded signal into the accelerometer records.

Direct transfer of seismic data into a scientific computer.

Relation to other projects

1) Other CNEN programs (3.I).

2) The research is performed in cooperation with ENEL and Istituto Nazionale di Geofisica. In particular a Joint Commission CNEN-ENEL has been established to study the seismicity of Italian territory for future nuclear power plants.

Reference documents

Reports on the 1976 Friuli earthquake are available.

Degree of availability

Open.

Personnel involved

6 men-year.

Title 1 (Original language)	Classification
Ricerca sulla liquefazione dei terreni	3.1
Title 2 (English)	**Country** ITALY
Research on sands liquefaction	Sponsor / Organisation: CNEN
Date initiated: May 1975 Date completed: In progress Last updating: April 1977	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.

Title 1 (Original language)	Classification
Studio sulla possibilità di previsione di terremoti con metodi idrogeochimici	3.1
Title 2 (English)	**Country** ITALY
Study on the possibility of predicting earthquakes by hydrogeochemical methods	Sponsor / Organisation: CNEN
Date initiated: January 1975 Date completed: In progress Last updating: April 1977	Project Leader M. Dall'Aglio

Description

It has been demonstrated 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.

TITLE 1 (original language) Progettazione sismica di componenti, sistemi e strutture nucleari	Classification 3.1
TITLE 2 (english) Seismic design for nuclear components, systems and structures	Country: ITALY Sponsor: Organisation: AGIP Nucleare S.p.A.
Date initiated May 1975 Date completed December 1975 Last updating June 1976	Project Leader P. Grillo

Description :

Design methods of components, systems and structures for nuclear plants based on their tridimensional analysis with time-history and design spectra.
In particular a detailed study of soil-structure interactions has been made (Authors: P. Grillo, G. Pochini).

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.

Title 1 (Original language)	Classification
Studi di ingegneria del sito	$\boxed{3:1}$ - 3.5
Title 2 (English)	**Country** ITALY
Studies of site engineering	**Sponsor** CNEN **Organisation** CNEN
Date initiated November 1974 (present phase) **Date completed** in progress **Last updating** April 1977	**Project Leader** S. Polinari

General aim

Studies on parameters occurring in the evaluation of sites for nuclear plants.

Particular objectives, Program

The program is organized into the following tasks:
- analysis of earthquakes strong motion records
- dynamic response analysis of soil
- analysis of soil-structure interactions
- experimental and theoretical determination of the vibration characteristics of nuclear plant structures
- development of codes for above analysis
- statistical analysis and studies of exceptional meteorological events
- stochastic analysis of seismic data.

Project status

1) Development of procedures for macroseismic and microseismic records processing.
2) Studies on seismicity of Friuli (Northern Italy) and other Italian regions (focal mechanism, earthquake spectra, seismic moment, linear dimension of fault, dislocation and stress drop, etc.).
3) Development of techniques for shear waves measurements.

Next steps

Besides development of above items: probabilistic treatment of the 1976 Friuli earthquake.

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Title 1 (Original language)	Classification
Studi di ingegneria del sito	3.1 - 3.5

Relation to other projects

3.1 (CNEN, ENEL programs).

Reference documents

Reports on the 1976 Friuli earthquake and other publications are available.

Additional information

The research is performed in cooperation with ENEL and Istituto Nazionale di Geofisica; in particular a Joint Commission CNEN-ENEL has been established to study the seismicity of Italian territory for future nuclear power plants.

Degree of availability

Open.

Personnel involved

12 men-year

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. bis 31.12. 1977	3.2	RS 165 und RS 149

Vorhaben/Project Title	Land/Country
Grenztragfähigkeit von Stahlbetonplatten bei hohen Belastungsgeschwindigkeiten (z.B. Flugzeugabsturz) und: Untersuchung der Widerstandsfähigkeit von Betonstrukturen gegen Flugzeugabsturz Ultimate Bearing Capacity of Reinforced Concrete Plates under Time-Dependent Loads (e.g. Aircraft Crash) and: Investigation of the Resistance of Concrete Structures to Crashing Aircrafts	FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Hochtief AG, Frankfurt Abt. KT I und Bundesamt f. Wehrt. u. Besch. Fachbereich KG IV 7

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.7. 1975 und 1.1o. 1974	31.12. 1979	Riech (Koordination)
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	December 1977	RS 165: 1639 TDM RS 149: 254o TDM

1. General aim

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. 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.....

The high load peak values within short periods require the knowledge of the kinetic ultimate bearing loads for a safe and economic design of the plates and shells being used, i.e. the best utilization of all safety reserves.

2. Particular objectives

Investigation of impact load/time characteristics during the impact of deformable missiles
Investigation of the kinetic bearing behaviour of reinforced concrete plates

3. Research Program

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 stress state in the neighbourhood of the loading area.

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.

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.

4. Experimental facilities

-

5. Progress to date

5.1 Pilot plant

The remaining works in connection with the erection of the plant having been completed, the plant has been ready for testing purposes since early March.

5.2 Measuring series I

The tests performed during measuring series I aim at obtaining information on the behaviour of deformable missiles when being subjected to an impact onto quasi-rigid reinforced concrete structures. Special emphasis was attributed to a qualitative check of theoretical calculation models which were performed in order to record such impacts and to a numerical determination of the free parameters contained in these models by means of adapting the results of calculations to the measurements.

The velocity of the impact and the distribution of the longitudinal rigidity of the missile are considered to be the essential variable factors of measuring series I. In addition, the determination of the impact load/time characteristics is expected to be the substantial result of measuring series I.

Within the scope of measuring series I, a total of four tests was run during the first six months of 1977.

From the very beginning, the low number of tests only allowed for a variation of the impact velocity as an input factor, whereas the variation of the distribution of the rigidity was deferred.

Basic test data:

Test I/1:

Missile type 2/7 (total length = 6 m; weight = 1020 kg; material St37; outside diameter = 600 mm, shell thickness 5 mm in the front and 10 mm in the rear area; in the interior of the middle third, a concentric tube ist mounted as an additional reinforcement, velocity of impact: 239,3 m/s

Test I/2:
Missile type 11 (total length = 6 m; weight = 973 kg; material St37; outside diameter = 600 mm, shell thickness 7 mm in the front (2,5 m long) and 10 mm in the rear area (3,3 m long); no interior additional reinforcement,
velocity of impact: 241,4 m/s.

Test I/3:
Missile type 11; velocity of impact: 197,2 m/s.

Test I/4:
Missile type 11; velocity of impact: 268,2 m/2.

The same target structure was used for each of the four tests as subsequent to each shot, only minor surface destruction occurred at the impact place of the missile, these places being repaired after each test. The complete remaining structure of the rigid target body did not reveal any cracks or other defects. By means of this, in addition to the savings of cost, an advantage in the testing procedure was achieved.

Due to this measure, the testing equipment may be considered as being almost equivalent for each of the four tests, so that apart from the scheduled variation of the parameters "velocity of impact" and "distribution of rigidity" no falsifying effects due to alterations of the testing equipment are to be expected.

5.3 Measuring series II:
During the third quarter of 1977, three reinforced concrete plates were tested by a total of four tests. These tests aimed at achieving first findings with regard to the influence of the bending and shear reinforcement on the load-bearing behaviour of a reinforced concrete plate when being subjected to a short-term concentrated load. For each of the four tests the process of loading lasted for approximately 25 to 30 ms.

For an effective span of 5,40 x 5,40 m, the dimensions of the three used test plates amounted to 6,50 x 6,00 x 0,70 m each. As a model, a missile of type 11 was used, the impact load/time characteristics of which may be determined from the tests of measuring series I.

Basic test data:

Test II/1:
The bending reinforcement of the first test plate was determined on the basis of the presently valid I.f.Bt-Instructions for the Design of Components for Loading Case "Aircraft Crash".

Contrary to this, the shear reinforcement was reduced to approximately one third of the amount required by way of calculation.

The concrete corresponded to strength class Bn 250, however, as revealed by subsequent checks, it ranged in the lower limit of this class.

The missile reached an impact velocity of 247,6 m/s (nominal value: 240 m/s).

On the front side of the target plate, the penetration depth of the missile amounted to approximately 25 cm. On the back, beyond the outermost reinforcement layer, a large part of the concrete cover had spalled. The rear reinforcement had been pushed backwards for a maximum of 20 cm, however, without having been destroyed. The concrete towards the interior of the plate between the front and back reinforcement layers was cohered by the rear reinforcement. The plate was not perforated by the impact.

Test II/2:
Compared to the first test plate, the bending reinforcement of the reinforced concrete plate for the second test was reduced by about 40 %. The shear reinforcement, however, was maintained. The velocity of the impact was reduced to such an extent that the stress imposed onto the bending reinforcement exactly corresponded to that of test II/1.

With this, compared to the first test plate and related to the actually imposed load, the shear reinforcement had been enlarged by about 30 %.

The measured velocity of the impact amounted to v_o = 172,2 m/s. During this test, the stress imposed onto the plate was so that the missile could penetrate the concrete at the front of the plate by only 1 to 2 cm, entailing only individually occurring radial cracks of very trivial width at the back of the plate.

Test II/3:
During this test, the reinforced concrete plate having been preloaded but apparently hardly affected by the second test was again fired at. The primary purpose of this test was to check whether the instant of the penetration could be determined by means of telemetric measurements.

The velocity of the impact was chosen so high that perforation of the plate could be positively expected. The velocity of the impact was measured to be 217,9 m/s. The evaluation of the measurement data has not yet been terminated.

Test II/4:
In order to restrict the influence of the shear reinforcement on the bearing behaviour, compared to the first slab the reinforcement of the third test plate was modified as follows:
- the shear reinforcement was enlarged by about 100 %
- the bending reinforcement was reduced by about 40 %.

Due to intensified checks during the mixing procedure, the scheduled concrete class was achieved.

In order to gain comparative data, the loading imposed onto the plate corresponded to that of test II/1. Again, the missile (type 11) reached an impact velocity of 247,6 m/s.

Contrary to test II/1, the missile penetrated the front of the plate by approximately 4 cm only. The back of the plate revealed radial cracks, spalling of the concrete, however, did not occur.

6. Results

6.1 Preface

The evaluation of the performed tests has not yet been completed because of the fact that up to now only part of the measurement records could have been transmitted into the form required for evaluation purposes.

In addition, at present it cannot yet be stated whether the acceleration measurements taken at the missile by means of telemetry may be utilized. If these measurements prove to be unexact , some other measuring method must be applied and tested for recording the distance- resp. velocity/time characteristics of the rear end of the missile. Shots taken by a high-speed camera might be suited for this purpose, however, referring to the presently available shots, the velocity of the sequence of shots should be increased and the end of the impact period should be made perceptible.

6.2 Measuring series I

Based on the partially effected evaluation of the tests, at present it may be stated

- that the Riera-Model was qualitatively confirmed for the mathematical treatment of the impact procedure of deformable missiles, the actual deformation procedure, however, being considerably more complicated than described by the Riera-Model;

- that for the interesting speed range, the performed tests delivered sufficient information concerning the influence of the impact velocity on the impact load/time characteristics in connection with a constant distribution of the rigidity, the consideration of this aspect for theoretical calculation models thus appearing to be adequately experimentally proven;

- that both tests performed within the scope of measuring series I were not appropriate to clearly settle the question to what extent a distribution of the rigidity graded over the length of the missile influences the impact load/time characteristics and the question how the mathematical consideration of this influence shall be effected for the calculation model. A doubtless confirmation of this theory is not yet possible.

In order to fairly secure the entire set of problems involved with the theoretical consideration of the deformation behaviour by means of experiments, some additional tests exceeding the number of tests stipulated in RS 149/165 seem to be necessary.

6.3 Measuring series II

From the qualitative analysis of the fracture pattern of the first test plate (II/1), a local shear failure within the area of the load application was identified to be the reason for the failure. Apparently the plate was only minorily subjected to bending stresses. At the moment it cannot yet be decided whether the more favourable result of test II/4 compared to that of test II/1 was originated by the enlargement of the cross-sectional area of the stirrups or by the improved quality of the concrete.

7. Next steps

7.1 Evaluation of measuring data:

The systematic numerical evaluation of the measuring data shall be set about in the course of the first quarter of 1978.

7.2 Improvement of measuring methods for recording the behaviour of the missile

7.2.1 The reason for the occurance of jammings during the telemetrically transmitted measurements of the acceleration at the missile shall be analyzed and eliminated as far as possible. In addition, attempts shall be made to eliminate the jammings from the measuring records. From the thus corrected data, the derived functions (velocity-, distance-time characteristics) shall again be determined for the rear end of the missile. These functions may be checked by means of other test records (remaining length of the missile, determination of the velocity by evaluating the shots of the high-speed camera).

7.2.2 The shooting technique of the high-speed camera shall be improved. It is aimed at achieving an increased number of shots taken during the impact procedure and at recording the final impact stage.

7.3 Test procedure

Supplimentary tests within the scope of measuring series II are prepared.

7.4 Revision of the research programme

The research programme shall be revised under consideration of the results of the first tests of measuring series II, including the given possibilities of supplementary tests on a smaller model scale.

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.77 - 30.4.77	3.2	RS 226

Vorhaben/Project Title	Land/Country
Geschoßbelastung von armiertem Beton, Rechenprogramm und Berechnungen Impact Loading of Reinforced Concrete, Computer Code and Calculation	FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Fa. Kerntechnik Entwicklung Dynamik 6458 Rodenbach b. Hanau

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.9.1976	30.4.77	Dipl.-Ing. B. Nowotny
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
completed	April 1977	DM 80.000,--

1. General Aim

The aim is a contribution to the theoretical explanation of impact problems at nuclear power plants, as they arise e.g. in the case of airplane crash.

2. Particular Objectives

The purpose of this research project is to evaluate a mathematical model for reinforced concrete loaded by hard missiles and to check it against experiments. Using the mathematical model experimental results should be extrapolated later, especially to the effects of an airplane crash on a nuclear power plant.

3. Research Programm

3.1 Development of a dynamic anisotropic concrete model

3.2 Comparison of calculations and impact tests

4. Experimental Facilities, Computer Codes

(3.1) The mathematical model is formulated as special subroutines for the computer code PISCES-2DL /1/. This code is available by everybody at Control Data Corporation.

The material model includes the following phenomena and capabilities:

- anisotropic cracking and spalling of the concrete due to tensile stresses,
- destroying of the concrete due to shear stresses,
- "yielding" of the destroyed concrete,
- crushing of the concrete (recordering of grains to more compacted form), where a permanent volume change takes place,
- elastic-plastic behaviour of the steel reinforcement, independently in the three directions of the reinforcement,
- sliding of the steel in the concrete,
- breaking of the steel reinforcements at the ultimate tensile strain,
- two dimensional (axial symmetric), fully dynamical.

5. Progress to date

(3.2) Four missile tests performed by the Ernst Mach Institute have been calculated and compared with measurements.

6. Results

Test No.	149	130	161	153
missile size [mm]	40 dia x 80	40 dia x 40	40 dia x 40	40 dia x 40
missile speed [m/s]	226	146	164	291
concrete scale	1 : 1	1 : 1	1 : 2	1 : 2
calculated maximum missile displacement into concrete [mm]	60	15	16,5-19*)	38
measured crater depth [mm]	59	26	31	44
calculated crater depth [mm]	60-65	25-30	25-30	-

*) parameter variation

In the calculation /2/ the maximum displacement of the missile into the concrete can be casily delivered, the resulting crater depth can be estimated from the calculation using criteria like distribution of velocity and destruction of the concrete particles. In the tests /3/ only the depth and the shape of the crater has been measured but not the dynamic motion of the missile. The comparison shows a good agreement. The calculation and the comparison also show, that the crater depth tends to be bigger than the missile displacement due to destruction and thrusting aside of the concrete in front of the missile, when the speed is rather small.

7. Next steps

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8. Relation with other Projects

-

9. References

/1/ PISCES 2 DL, Manual A, B, C, D
Physics International Company
2700 Merced, San Leandro, Calif.

/2/ Geschoßbelastung von armiertem Beton, Entwicklung eines anisotropen Materialmodells für armierten Beton und Vergleich von Berechnungen mit Geschoßversuchen.
KED-Bericht, Juni 1977

/3/ Helga Langheim,
Impactuntersuchungen an armierten Betonplatten, Teil 1, Bericht Nr. E 9/76, Ernst-Mach-Institut, Freiburg, Juni 1976, RS 102 - 7

10. Degree of Availability of the Reparts

-

TNO-IBBC	CLASSIFICATION : 3.2.3.3.7.1
TITLE : Responsieberekeningen voor reactorgebouw	COUNTRY: THE NETHERLANDS
	SPONSOR : Ministry of Social Affairs; Ministry of Public works (DEF); TNO-IBBC
	ORGANIZATION : TNO-IBBC
TITLE (ENGLISH LANGUAGE): Dynamic response of reactor structures (building and containment)	PROJECTLEADER : Kusters
INITIATED : June 1974 — LAST UPDATING : June 1978	SCIENTISTS : Kusters, de Groot, de Witte
STATUS : Progressing — COMPLETED : 1979	

General aim

Development and application of calculational tools for the evaluation of structural response of reactor buildings under dynamic loading conditions.

Particular objectives

The calculations are directed to the evaluation of the effects of
(1) pressure/blast waves of gas explosion in the vicinity of the site
(2) impact of striking aircraft on the reactor (containment) buildings
(3) investigation of the load-carrying capacity of the buildings in the vicinity of the reactor
(4) investigation of the load-carrying capacity of flat plate panels of reinforced concrete, subjected to local dynamic loads

Experimental facilities and program

None.

Project status

A computational method has been developed and implemented in the MARC-computer code to describe the non-linear behaviour of reinforced concrete with a finite element technique. This method has been applied to a typical reactor building to analyse the structural response due to (1) blast wave, (2) impact of large commercial aircraft.
The new implemented Buyukozturk criterion including cracking in the Marc-code has been tested, in comparing with the results obtained with the DIANA-computer code.

Next steps

To close the investigation of the load-carrying capacity of flat plate panels.
Investigation of the load-carrying capacity of the building in the vicinity of the reactor.
To implement the developed computational methods in the computer code of DIANA of TNO-IBBC, to get available the improved numerical description of the inelastic behaviour of concrete in DIANA.

Relation to other projects

None.

Reference documents

1. "Mathematical description of the non-linear behaviour of reinforced concrete" by Ir. H. Geertsema, June 1976, TNO-IWECO, report nr. 11261/2.
2. "Responsieberekeningen aan een reactorgebouw voor belastingen t.g.v. een drukgolf en een neerstortend vliegtuig" ("Structural response analysis of a reactor building subjected to a blast wave and aircraft impact") by Ir. H. Geertsema, September 1976, TNO-IWECO, report nr. 11261/3.
3. "Onderzoek naar een mogelijke verbetering van het materiaalgedrag voor de berekening van gewapende betonkonstrukties", by Ir. G.M.A. Kusters and Ir. A.K. de Groot, February 1978, TNO-IBBC, report nr. B-78-60/64.7.0128.

Degree of availability

Through: Ministry of Social Affairs
TNO-IBBC

Budget

Hfl. 378.000.- (1975, 1976, 1977)

Approx. Hfl. 350.000.- (1978 + 1979).

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
July 1.- Dec. 31.77	3.3	RS 265
Vorhaben/Project Title Untersuchung über die Einwirkung schädlicher Stoffe auf Kernkraftwerke Analysis on the Effects of Dangerous Materials on Nuclear Power Plants		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Battelle - Institut e.V. Frankfurt am Main
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
July 1. 1977	Sept. 30. 1978	Dr. H.J. Nikodem
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
continuing	Dec. 31, 1977	485.470,-- DM

1. General Aim

Identification of possible effects and their description, and consequences of the exposure of components of critical systems and personnel of nuclear power stations to dangerous materials; proposals for measures to reduce the risk.

2. Particular Objectives

2.1 Identification of possible points of attack by dangerous materials.

2.2 Description of mechanisms from exposure to dangerous materials, determination of critical exposures.

2.3 Listing and evaluation of potential dangerous materials and their sources.

2.4 Selection and description of critical events.

2.5 Quantitative evaluation of critical events.

2.6 Catalogue of proposed measures to reduce the risk.

3. Research Program

3.1 Analysis of the reference plant Biblis B.

3.2 Determination and description of the effects of potentially dangerous materials.

3.3 Compilation of groups of dangerous materials.

3.4 Transport behaviour of dangerous materials.

3.5 Determination and description of external sources of dangerous materials.

3.6 Compilation of dangerous materials, their sources and chains of events which lead to effects on sensitive elements.

3.7 Compilation of system failures in nuclear power plants which may occure by attacks of dangerous materials.

3.8 Identification of critical chains of events.

3.9 Quantification of external sources of dangerous materials.

3.10 Quantification of critical events.

3.11 Compilation of critical dangerous materials, sources and their risk potential

3.12 Catalogue of propos ed protective measures, estimates of the expected safety gais.

4. Experimental Facilities, Computer Codes

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5. Progress to Date

to 3.1 Compilation and description of exposed components and control systems, inspection of reference plant.

to 3.2 Work initiated: semiconductors, electrical components, synthetic materials.

to 3.3 Work initiated: tabulation of potentially dangerous materials.

6. Results

none

7. Next Steps

Completion of the Project.

8. Relation with Other Projects

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9. References

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10. Degree of Availability of the Reports

-

122-2 -02/4175-10		3-3
Titre Agressions d'origine externe sur les installations nucléaires: explosions chimiques non confinées dues à un environnement industriel ou aux voies de communication.		Pays FRANCE Organisme directeur CEA/DSN EdF/SEPTEN
Titre (anglais) External impacts on nuclear plants: unconfined chemical explosions due to industrial environment or to communication routes.		Organisme exécuteur CEA/CESTA ENSMA Responsable (CEA/DSN/FAR) (EdF)
Date de démarrage: 01/01/1976	Etat actuel : en cours	Scientifiques
Date prévue d'achèvement: 31/12/1978	Dernière mise à jour :02.01.1978	

1. Objectif général

Protection des installations nucléaires contre des agressions d'origine externe: cas particulier des explosions chimiques de masses gazeuses dérivantes libérées par un accident dans un environnement industriel ou au niveau de voies de communication.

2. Objectifs particuliers

2.1 - Recherche de lois d'échelle pour les caractéristiques de l'onde de pression aérienne engendrée par la détonation (au sens strict) d'un mélange air-hydrocarbure.

Conditions pour une initiation en détonation.

2.2 - Recherche de lois d'échelle pour les caractéristiques des secousses telluriques induites par la détonation en surface (au sens strict) d'un mélange air-hydrocarbure.

2.3 - Recherche de l'effet de divers paramètres sur la cinétique d'une explosion: caractéristiques du mélange,géométrie, intensité et localisation de l'initiation, obstacles, confinement partiel, ..

Possibilités pour une déflagration d'évoluer vers une déflagration rapide ou une détonation.

2.4 - Recherche de modèles, utilisables dans les calculs de sûreté, représentant les caractéristiques de l'onde de pression aérienne dans les cas où il n'y a pas de détonation.

3. Installations expérimentales et programme

Les essais sont actuellement réalisés en deux endroits différents:

a) A l'Université de Poitiers, dans le laboratoire d'Energétique et de Détonique de M. le Professeur N. Manson.

Les effets de divers paramètres sur la cinétique de l'explosion (objectif 2-3) sont systématiquement analysés à petite échelle, les mélanges explosibles étant confinés à l'intérieur de bulles de savon hémisphériques de 20 cm de diamètre.

b) Au Centre d'Etudes Scientifiques et Techniques d'Aquitaine (CESTA).

Des essais de détonation de mélanges air-hydrocarbures sont effectués sur des volumes de mélange de plusieurs dizaines de m^3. Des tirs de 1 000 m^3 sont envisageables sur le terrain d'essais.

Le but poursuivi est l'obtention de données expérimentales pour les objectifs 2-1, 2-2 et (ultérieurement) 2-4.

Le programme est défini et réajusté périodiquement par un comité regroupant des experts du CEA, d'Electricité de France et de Gaz de France. Ce comité s'est adjoint par contrat le support scientifique des spécialistes de l'Université de Poitiers.

Programme actuel:

a) A l'Université de Poitiers.

Objectif 2-3. Il s'agit d'essais exploratoires destinés à rechercher les paramètres influant de manière importante sur la cinétique des déflagrations. Ces résultats devraient permettre d'orienter un programme d'essais à grande échelle.

Indépendamment du travail expérimental, les experts de l'Université de Poitiers assurent, à partir des essais au CESTA, la modélisation nécessaire à l'objectif 2-1, puis ultérieurement celle correspondant à l'objectif 2-4.

b) Au CESTA.

Obtention de données expérimentales pour les objectifs 2-1 et 2-2, ultérieurement pour l'objectif 2-4.

c) Au DSN.

Travail de modélisation pour l'objectif 2-2.

4. Etat de l'étude

4.1 - Avancement à ce jour.

a) Objectifs 2-1 et 2-2.

Les essais sont relatifs à des tirs de ballons sphériques en latex ou en mylar de volumes s'échelonnant de 1 à 54 m^3 de mélange gazeux explosible. L'allumage se fait par explosif au centre des ballons.

18 essais ont porté sur des mélanges air-acétylène initiés par un détonateur de 1 à 1,5 g (détonateur Briska "P" ou fulminate de mercure); 12 ont conduit à une détonation autonome dans le mélange gazeux.

9 essais ont porté sur des mélanges air-éthylène, le détonateur étant renforcé par 10 g de plastic; 4 ont conduit à une détonation autonome de gaz.

5 essais ont été réalisés sur des mélanges air-propane, dont 2 ont conduit à une détonation autonome (renfort de 500 g de plastic).

L'interprétation des essais de détonation air-acétylène et air-éthylène est terminée quant à l'objectif 2-1.

b) Objectif 2-3

Les travaux en laboratoire n'ont pu commencer qu'en octobre 1976 (venue de M. Huneau). Le montage expérimental est terminé et diverses expériences d'orientation ont été réalisées (allumage de faible énergie par étincelle, mélanges oxygène-propane de rayons variables, mélanges concentriques de richesses variées).

c) Objectif 2-4

Etude bibliographique seulement.

4.2 - Résultats essentiels.

a) Objectif 2-1

Une modélisation et une formule ont été proposées en ce qui concerne la surpression de crête de l'onde lancée dans l'environnement.

b) Objectif 2-2

L'instrumentation des essais en accéléromètres enterrés ayant été tardive, les données expérimentales sont encore insuffisantes pour tirer des conclusions.

c) Objectif 2-3

Une dissymétrie de la propagation de la flamme en géométrie hémisphérique a été observée pouvant provoquer des effets directionnels dans le champ de pression.

La célérité moyenne de la flamme augmente très rapidement avec le rayon initial de la charge jusqu'à atteindre 110 m/s pour un rayon de 10 cm (rappelons qu'il s'agit d'un mélange propane-oxygène).

5. Prochaines étapes

a) Objectif 2-1

Comparaison des résultats de la formule proposée avec d'autres modèles

Mise au point de la mesure en un point donné de la surpression en fonction du temps (seule la surpression de crête a été obtenue jusqu'à présent avec précision). Reprise de quelques essais de détonation.

Recherche d'une modélisation pour les durées de phase positive et les impulsions.

Recherche des conditions d'initiation en détonation pour des mélanges gazeux moins réactifs.

b) Objectif 2-2

Poursuite des mesures à l'occasion des essais de détonation.

Tentative de modélisation.

c) Objectif 2-3

Poursuite des essais en laboratoire. Définition et réalisation d'essai de déflagration à grande échelle (sphères, hémisphères, boyaux).

d) Objectif 2-4

Dépouillement des essais à grande échelle prévus en c) et recherche de modélisations représentatives (en s'aidant de la bibliographie).

6. Relations avec d'autres études

- Formation et dispersion de nappes de gaz dérivantes explosibles ou toxiques suite d'une fuite massive sur un transport ou un stockage de produits chimiques.

- Identification et caractérisation des agressions externes dues aux activités humaines . Détermination de leur probabilité d'occurrence.

7. Documents de référence - non disponibles

122-2 -03/4171-10 222-2 -03		3.3
Titre Formation et dispersion de nappes de gaz dérivantes explosives ou toxiques suite à une fuite massive sur un transport ou un stockage de produits chimiques.		Pays France Organisme directeur CEA/DSN
Titre (anglais) Formation and dispersion of drifting explosive or toxic clouds as a conséquence of a mass leakage incident on transport or facilities for the storage of chemicals products.		Organisme exécuteur CEA/DSN/SESRS Responsable DSN/SESRS/FAR
Date de démarrage 01.01.78	Etat actuel Lancement	Scientifiques
Date prévue d'achèvement 31.12.82	Dernière mise à jour 02.01.78	

1 - Objectif Général.

Gaz ou aérosols toxiques : détermination en fonction du temps du champ de concentrations dans l'air, au niveau du sol. Dépôts sur le sol ou sur l'eau.

Gaz explosibles (hydrocarbures essentiellement) : détermination en fonction du temps des régions où l'hydrocarbure se trouve mélangé à l'air dans une fourchette donnée de concentrations; calcul de la masse correspondante d'hydrocarbure.

2 - Objectifs particuliers

2.1. Détermination du terme source

- Cas d'une rupture de gazoduc.
  En particulier, effet de la vitesse initiale du jet gazeux
- Cas d'une rupture de cuve de gaz liquifié.
  Effet des caractéristiques du stockage (sous pression, réfrigéré)
  Calcul du débit d'évaporation en cas d'épandage et d'évaporation simultanés sur le sol ou sur l'eau (surface limitée ou non).
- Cas d'une rupture de canalisation de transport sous pression de liquides (ammoniac.....) : formation d'aérosols.

2.2. Dispersion atmosphérique.

Modélisation des phases initiales, où le gaz ou l'aérosol ne sont pas des polluants minoritaires.

- Cas des gaz lourds, des gaz légers, des gaz de densité variable (évolution en température), des aérosols liquides avec vaporisation éventuelle des goutelettes, des aérosols solides.

  Raccordement avec les modèles classiques de dispersion atmosphérique de polluants minotaires.

4 - Etat de l'étude : lancement

5 - Prochaines étapes.

Compilation bibliographique.

6 - Relation avec d'autres études.

Agressions d'origine externe sur les installations nucléaires : explosions chimiques non confinées dues à un environnement industriel ou aux voies de communication.

Etude des transferts atmosphériques.

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TNO-IBBC		CLASSIFICATION : 3.2.3.3.7.1
TITLE : Responsieberekeningen voor reactorgebouw		COUNTRY: THE NETHERLANDS
		SPONSOR : Ministry of Social Affairs; Ministry of Public works (DEF); TNO-IBBC
		ORGANIZATION : TNO-IBBC
TITLE (ENGLISH LANGUAGE): Dynamic response of reactor structures (building and containment)		PROJECTLEADER : Kusters
INITIATED : June 1974	LAST UPDATING : June 1978	SCIENTISTS : Kusters, de Groot, de Witte
STATUS : Progressing	COMPLETED : 1979	

PROJECT TITLE : Preliminary Research on Tornado Effects 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) Studio preliminare sulle trombe d'aria e sui problemi che ne derivano per gli impianti nucleari.	Classification 3.5
TITLE 2 (english) Preliminary study on tornadoes and their effects on nuclear power plants.	Country: ITALY Sponsor: CNEN Organisation: University of Pisa.
Date initiated 23/3/74 Date completed 1976 Last updating June 1976	Project Leader MARINI Marino

A study is in course on tornadoes, referring to their main physical characteristics on the basis of data of experiences on tornadoes 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 have been examinated, with particular reference to the nuclear plants.

One has studied the behaviour and the effects of some tornadoes in Italy making an attempt for a first analysis of the experimental data.

The problem of a classification of the tornadoes by area and intensity on the basis of the provoqued damages has been examinated.

Title 1 (Original language)	Classification
Studi di ingegneria del sito	$\boxed{3:1}$ - 3.5
Title 2 (English)	**Country** ITALY
Studies of site engineering	**Sponsor** CNEN **Organisation** CNEN
Date initiated November 1974 (present phase) Date completed in progress Last updating April 1977	**Project Leader** S. Polinari

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	4	RS 153

Vorhaben/Project Title	
Untersuchung von Betriebstransienten bei Versagen des Schnellabschaltsystems (ATWS-Studie) Investigation of Operation Transients after Failure of the Shutdown System (ATWS-Study)	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 11, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 11. 74	30. 9. 77	F. Winkler
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	31. 12. 1977	739.000,-- DM

1. 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.

2. Particular Objectives

If necessary, the design conditions for a second scram system can be specified on the basis of the results of this investigation. For this purpose some pre-investigations shall be carried out.

3. Research Program

3.1 PWR

The consequences on the transient behaviour of a failure of the shut down system will be investigated, corresponding to the RSK guide lines. The analysis will be 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 is assumed that only the shut down system failed and all the other systems are operating well.

3.2 BWR

The behaviour of a typical BWR power plant will be 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 %/s) speed, and the collective electromechanical insertion of the control rods (insertion time: 120 seconds) are considered.

4. Test Facilities

No test facilities necessary.

5. Progress to Date

5.1 PWR

Development of a 1-dimensional reactor-dynamic code was initiated. This program, called KINBOX, is based on the multi-dimensional code IQSBOX. Although the latter can also be used as an 1-dimensional code, it requires - compared to three-dimensional calculations - relatively high computational costs. Since in connection with various ATWS accidents spatial effects in axial direction are most severe, it was thought advisable to prepare a special 1-D code. For this purpose the neutronic IQSBOX module was completely revised and all necessary instructions only applying to higher dimensional calculations were eliminated.

5.2 BWR

A fault tree was established for failures which may lead to ATWS.

The statistics of an unscheduled scram were prepared and evaluated for both US and German nuclear power plants.

In addition, a parameter study was performed on main heat sink failure and a check-up made of the piping and component stresses during ATWS transients.

6. Results

6.1 PWR

The neutronic mudule of the 1-dimensional KINBOX code worked about twice as fast as the IQSBOX on similar 1-D problems.

6.2 BWR

According to operational experiences gained from the German plants Grundremmingen-A, Lingen and Würgassen, the failure probability leading to ATWS, is lower by approx. one order of magnitude than is usually assumed by safety analysis.

7. Next Steps

The final report will be written.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

146-1 -01/4151-20 4

Titre Développement des moyens de calcul nécessaires à l'étude des transitoires anormaux sur les réacteurs PWR. Etude de transitoires anormaux .		Pays FRANCE Organisme directeur CEA/DSN
Titre (anglais) Development of computer codes necessary to study abnormal transient conditions on PWR. Abnormal transients .tudies		Organisme exécuteur CEA/DSN-SETSSR Responsable DSN/FAR
Date de démarrage 01/01/74	Etat actuel en cours	Scientifiques
Date prévue d'achèvement 31/12/77	Dernière mise à jour 1/78	

1 - <u>Objectif général</u> :

Développer un code de calcul permettant de contrôler l'étude des transitoires accidentels de classe 2, 3 et 4 (hors A.P.D.R.)* figurant dans les rapports de sûreté des centrales PWR.

* - APDR: accident de perte de réfrigérant .

2 - <u>Objectifs particuliers</u> :

Etude des transitoires de fonctionnement des centrales à eau pressurisée.
Contrôle des transitoires accidentels conduisant à des surpressions primaires (examen des situations).
Etude de l'influence des régulations sur le comportement transitoire de la centrale.
Recherche des points délicats des représentations mathématiques des phénomènes physiques.
Etude des A.T.W.S.

.../

4- Etat de l'étude :

1) Avancement à ce jour :

- Des modèles ponctuels décrivent le comportement des différents éléments constitutifs du circuit primaire y compris certains systèmes de régulation et de protection.
Un modèle de générateur de vapeur axial permettant de calculer entre autre l'évolution du niveau d'eau dans le générateur de vapeur et un modèle de pressuriseur fonctionnant indifféremment en simple phase (vapeur ou liquide) et en double phase sont intégrés à la version actuelle.

- Calcul de la répartition des débits dans les lignes vapeur lors d'une rupture de tuyauterie vapeur.

- Calcul des débits de décharge en phase liquide en vapeur aux soupapes du pressuriseur par les méthodes de MOODY, FAUSNE et du modèle homogène équilibré.

- Couplage d'un modèle de générateur de vapeur axial décrivant la réduction de l'échange primaire secondaire lors du dénoyage des épingles.
Le module utilisé lors de l'étude des A.T.W.S. décrit en particulier la dynamique de l'assèchement du générateur de vapeur en s'appuyant sur un jeu de corrélations d'échange variables tout au long du transitoire.

2) Résultats essentiels :

- Etude de transitoires accidentels de classe 2 et 3 concernant la centrale de Fessenheim.
Le calcul des accidents suivants :
  - Retrait incontrôlé des grappes de réglage en puissance.
  - Perte totale de la charge électrique.
  - Augmentation excessive de la charge.
  - Perte totale de débit primaire, révèle un bon accord avec les calculs présentés par le constructeur dans le rapport provisoire de sûreté.

- Etude de transitoires de dimensionnement du circuit primaire.
Dans ce domaine l'étude de la surpression consécutive à une perte de la charge électrique a été effectuée sur la centrale de Fessenheim.

- Etudes des A.T.W.S. :
  - Etude de la perte de l'eau alimentaire avec déclenchement de la turbine. Cette étude a montré que, dans la mesure où l'on adopte des hypothèses communes en ce qui concerne le modèle de dégradation de l'échange de chaleur aux générateurs de vapeur, les résultats sont comparables à ceux obtenus par EdF et FRAMATOME.

  - Afin de se prononcer sur la validité de ces résultats de calcul obtenus à l'aide d'un générateur ponctuel, des études, s'appuyant sur un modèle axial muni en particulier d'un jeu de corrélations d'échange variables au cours du transitoire, sont en cours.

.../

- Etude de la perte de l'eau alimentaire sans déclenchement de la turbine. Ce jeu d'hypothèses conduit à un pic de pression primaire beaucoup plus élevé que le cas précédent.

- Etude de l'ouverture intempestive d'une soupape du pressuriseur. Le code permet entre autre d'étudier le début de l'accident pendant lequel le D.N.B. passe par un minimum.

5- Prochaines étapes :

- Etude d'un modèle de corrélation du niveau d'eau dans les générateurs de vapeur.
- Elaboration d'un modèle coeur tenant compte d'un mélange imparfait de l'eau des différentes branches et d'une répartition dissymétrique des températures dans le coeur.
- Traitement de la double phase.

6- Relation avec d'autres études :

Etude des transitoires anormaux.

Etude des A.T.W.S.

7- Documents de référence : rapports internes non disponibles

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 Scientists M. LARMINAUX
Initiated 1973	Completed 1975	
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.

Title 1 (Original language)	Classification
Dynamic studies for safety analysis	4 - 8
Title 2 (English)	Country ITALY Sponsor / Organisation } CNEN
Date initiated I962 Date completed in progress Last updating April I977	Project Leader M. Di Bartolomeo

I. - General aim - Concern the development of analog and hybrid models for dynamic analysis of nuclear power plants (both water and LMFBR reactors)

2. - Particular objectives - To study the behaviour of nuclear power plants and their components in abnormal conditions. To analyze all the possible accidents and the related safety problems.

3. - Experimental facilities - 2 analog computers - EAI-PACE-23I/R
   I hybrid computer - EAI-8945
   I hybrid computer - EAI-PACER-700

4. - Project status - These studies are made according to the CNEN need.

5. - Reference documents:

- I) P. Giordano - A. Mathis - G. Melucci Dynamics and control studies for a steam-generating pressure-tube reactor. Doc.CNEN RT/ING(65)I3 - Sept. I965
- 2) A. Mathis - B. Musso - E. Turrini Accident analysis for a fast source reactor. Energia Nucleare - Vol. I5 - n. 7 Luglio I968.
- 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 I967.

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Title 1 (Original language)	Classification
Dynamic studies for safety analysis	4 - 8

- 4) A. Mathis
  The use of hybrid computers in the Italian CNEN nuclear program.Conferen ce 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 Meet g on Computer Control
  Loen (Norway) - May 29th - June 2nd, 1972

6. - Degree of availability - Besides the equipments mentioned there is also a know-how in dynamic model implementations.

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Title 1 (Original language) Statistical analysis of randome signals	Classification I - 3 - 4 - 8 IO - I4
Title 2 (English)	Country ITALY Sponsor / Organisation CNEN
Date initiated 1966 Date completed in progress Last updating April I977	Project Leader A. Federico

Classification 4.1

Title 1 ANDYCAP: Tre dimensional dynamisk model af kogendevands reaktor kerne.	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
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

565

Classification: 4.1

Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: NORHAV - Three-Dimensional Transient Calculation Program for the PWR Core (ANTI)	Organization: Risø National Laboratory
Initiated date: 1977 Completed date: Status: in progress.	Scientists: Anne Margrethe Larsen

1. General aim

Development of a three-dimensional computer program for the calculation of transients in the PWR core.

2. Particular objectives

The computer program should be able to describe PWR transients where the spatial power distribution is important, covering the range from operational transients to design basis accidents (Rod ejection, ATWS). The neutronics part is a three-dimensional nodal theory program, and the hydraulic model is a transient subchannel program which was originally intended for blowdown calculations. The program is planned to deal only with the reactor core, so the boundary conditions at the core inlet and outlet will have to be specified.

3. Experimental facilities

4. Project status

The programming is in progress. The steady state part is running (in the debugging phase).

5. Next steps

Programming of the dynamics. Documentation.

6. Relation with other projects

The nodal theory routines are the same as in the ANDYCAP program (ANDYCAP: 3-D dynamical model of a BWR-core, classification 4.1).

The hydraulic model is the TINA program (NORHAV-P(B)WR blow-down computer program, classification 1.1).

7. Reference documents

8. Degree of availability

Available on exchange basis when completed.

567

Classification 4.1

Title 1 Kontrolstavs udskydning i en kogendevands-reaktor	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 Rod ejection transients in BWR	Project leader: P. Skjerk Christensen
Initiated: 1974 Completed: 1976 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. Several transients have been analysed

2. Essential results.

5. Next steps

6. Relation with other projects

7. Reference documents

B. Thorlaksen, Analysis of Control Rod Ejection Accidents in Large Boiling Water Reactors (Risø Report 344).

8. Degree of availability

The thesis is freely available.

Classification 4.1, 4.2, 4.3

Title 1	COUNTRY Denmark
PWR-stations dynamik model	SPONSOR Risø National Laboratory
	ORGANIZATION Risø National Laboratory
Title 2 PWR: A PWR power plant dynamics model	Project leader:
Initiated: 1972 Completed: 1974 Status: in use, being improved 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 finished and the two parts have been coupled together. Simple transients have been run.

   A more detailed version with two primary loops, a turbine and a feedwater system has been programmed for simulation by means of a digital simulation system.

2. Essential results:

5. Next steps

6. Relation with other projects

7. Reference documents

Risö report no. 318.

8. Degree of availability

Classification 4.1, 4.2, 4.3

Title 1 BWR-stations dynamik model	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 Development of a Dynamic Model of a BWR Nuclear Power Plant.	Project leader: P. Skjerk Christensen
Initiated: 1973 Completed: 1976 Status: in use Last updating: currently	Scientists: E. Nonbøl

1. General aim

The purpose of the project is to develop a dynamic model of a nuclear power plant based on a BWR reactor which simulates various transients occurring during normal operating conditions.

2. Particular objectives

The model includes a boiling water reactor, high- and low pressure turbines, moisture separator, reheater, condenser, feedwater heaters and feedwater pump. It is one-dimensional except for the nuclear part of the reactor which is based on point kinetics equations. A great deal of attention has been devoted to the model of the turbine and the feedwater heaters.

3. Experimental facilities

4. Project status

1. Progress to date: The model is finished. Several transients have been run. The kinetic model has been improved by introducing a one-dimensional part.
2. Essential results.

5. Next steps

572

6. Relation with other project

7. Reference documents

Risø Report No. 335, Risø Report No. 336.

8. Degree of availability

Title 1 (Original language)	Classification
Rampe di potenza	4.1
Title 2 (English) Overpower tests	**Country** ITALY **Sponsor** CNEN **Organisation** CNEN
Date initiated March 1974 **Date completed** February 1977 **Last updating** February 1977	**Project Leader** C. Lepscky

Description:

1. General aim

   Fuel cladding interaction and mechanical properties of irradiated cladding after rupture determined by power excursion.

2. Particular objectives

   Investigation 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 examinations at IFA Kjeller

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 as well destructive tests.
   Overpower test on 5 rods already performed including post-irradiation analysis.

5. Next steps

   None

6. Relation with other projects

   Interamp programme at Studsvik on standard irradiated rods, at different burn-up levels.

7. Degree of availability: to a limited extent

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Title 1 (Original language): Ricerca sui transitori di potenza del reattore RTS-1 senza circolazione forzata.	Classification 4.1
Title 2 (English): Research on the power transients of RTS-1 reactor without forced water cooling.	Country : ITALY Sponsor : C.A.M.E.N. Organisation : C.A.M.E.N.
Date initiated : 1974 Date completed : In progress Last updating : April 1977	Project Leader: F. D'AMONE (CAMEN) A.M. SPANO (CAMEN)

Description :

It has been investigated power transients of RTS-1 reactor without forced water cooling. Calculations have been carried out by the NCOMEUR code. The selfcompensated reactivity, without any action of external safety sistems and without any damage to the installation, has been found low related to the operation requirements and to the associated risks.

Reference documents :

- NCOMEUR. Report CNEN RT/PROT(71)31.

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:** THE 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 :October 1974 — **LAST UPDATING** : May 1978	**SCIENTISTS :** Bruens
STATUS :in progress — **COMPLETED** : End 1978	

General aim

Development of a calculational tool which 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.

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.

Experimental facilities and programma: -

Project Status

The following simulation programs are finished:
- hybrid reactor core model describing the neutron-kinetics and the thermal behaviour
- natural circulation steam generator computer modules describing the thermal/ hydraulic behaviour.
- coupling of reactor core model with steam generator model.
- digital model of the pressurizer.

Next steps

- digital models of the turbine, reheater and generator
- digital model of the preheater.
- coupling of the different computer programs.

Relations with other projects: -

Reference documents: -

Degree of availability

through Ministry of Social Affairs.

Budget: -

Personnel: 40 manmonths

4. POWER TRANSIENTS

Classification 4.1, 4.2, 4.3

Title 1 PWR-stations dynamik model	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 PWR: A PWR power plant dynamics model	Project leader:
Initiated: 1972 Completed: 1974 Status: in use, being improved Last updating: Currently	Scientists: P. la Cour Christensen P. Skjerk Christensen

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Classification 4.1, 4.2, 4.3

Title 1 BWR-stations dynamik model	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 Development of a Dynamic Model of a BWR Nuclear Power Plant.	Project leader: P. Skjerk Christensen
Initiated: 1973 Completed: 1976 Status: in use Last updating: currently	Scientists: E. Nonbøl

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:** THE 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 : October 1974 — **LAST UPDATING** : May 1978	**SCIENTISTS :** Bruens
STATUS : in progress — **COMPLETED** : End 1978	

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	4.3	RS 178

Vorhaben/Project Title	Land/Country
3D-Transientenprogramm - Modifizierung eines 3D-Transientenprogramms für den SWR 3D-Transient Program - Modification of a 3D-Transient Program for BWR	FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 121, Frankfurt

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 9. 75	31. 8. 77	Dr. Lockau
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	31. 12. 1977	475.000,-- DM

1. General Aim and 2. 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 these conservative factors can be corrected, without loosing safety margin. Therefore a 3 dimensional program will be developed, which gives more realistic details for the calculation of unsymmetric incidents in the core.

3. Research Program

The central activity concentrates on the development of a thermohydraulic 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.

4. Experimental Facilities

No experimental facilities necessary.

5. Progress to Date

The following experimental calculations were performed by IQSBWR:

a) pump failure NPP-Test 137 (new core)
b) preheater trip NPP test 134

In addition, the program was considerably improved as far as data processing is concerned.

6. Results

6.1 NPP Test 137 with IQSBWR

The 1- and 3-dimensional verification calculations on the failure of both recirculation water pumps (NPP Test 137) by IQSBWR showed good agreement between the two calculations and the measurement of the integral reactor power history over the pure flow rate change region (time range <10 sec). The subsequent subcooling change (-increase), however, does not quite result in the expected power increase given by the two IQSBWR-models. As was expected, the 3-dimensional model, however, provides a better basis for consideration of subcooling change than the 1-dimensional. The relatively long box height, 30.5 cm of the 3-dimensional and 18.3 cm of the 1-dimensional model will possibly lead to a somewhat unsatisfactory treatment of the subcooling transients.

The radial power distribution shows good agreement between the steady-state reactor simulator RS 3D and IQSBWR.

6.2 NPP Test 134 by IQSBWR

The verification of the preheater trip showed good agreement with the measurement results. It could be shown that even with a coarse mesh grid arrangement of $Z \doteq 30.5$ cm per node, the model is sufficiently exact for the thermohydraulics. The verification also showed, that only the use of the H_2O reflector as radial boundary condition renders satisfactory results. The simulation with constant Albedo-coefficients leads to incorrect results.

6.3 Improvements of the Computer Programs

By improving the iteration control, the I/O time can be halved.

It could be shown by a systematic parameter study that the parameter NSMAX (= number of flux iterations/power iteration) strongly influences the convergence. The optimum seems to lie between NSMAX = 3-5.

The program was considerably improved with respect to problem size and convergence behaviour. The present revision 1 can full-dynamically process 1/8 core KKK with 110 thermohydraulic channels and will converge in approx. 70 iterations down to 10^{-5}.

The implementation of IQSBWR on the CD 7600 used by the weather-service also allows for the treatment of considerably larger problems than it was the case up to now. Thus problems up to quarter-core size (approx. 200 channels) can be calculated.

7. Next Steps

The final report will be written.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Classification 4.1, 4.2, 4.3

Title 1 BWR-stations dynamik model	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 Development of a Dynamic Model of a BWR Nuclear Power Plant.	Project leader: P. Skjerk Christensen
Initiated: 1973 Completed: 1976 Status: in use Last updating: currently	Scientists: E. Nonbøl

Classification 4.1, 4.2, 4.3	
Title 1 PWR-stations dynamik model	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 PWR: A PWR power plant dynamics model	Project leader:
Initiated: 1972 Completed: 1974 Status: in use, being improved Last updating: Currently	Scientists: P. la Cour Christensen P. Skjerk Christensen

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: THE 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 : October 1974 / LAST UPDATING : May 1978	SCIENTISTS : Bruens
STATUS : in progress / COMPLETED : End 1978	

5. BEHAVIOUR, TRANSPORT AND RELEASE OF RADIOACTIVE SUBSTANCES

Classification 5

Title 1 Dosisbelastninger i A-kraftværker	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 Radiation Doses in Nuclear Power Plants	Project leader: Kurt Lauridsen
Initiated: February 1974 Completed: Status: Progressing Last updating:	Scientists: Kurt Lauridsen

1. General Aim. To study the radiation doses received by power plant personnel and the factors influencing the size of these doses.

2. Particular objectives. Development of a mathematical model for the transport of radioactive material in the coolant circuit of a BWR-plant, and calculation of the radiation fields outside the components of the coolant circuit.

3. Experimental facilities None

4. Project status

4.1. Progress to date. Three computer codes have been developed, FICOPI, INAPI, and SHIELD, all in FORTRAN.

FICOPI calculates the inventories of radioactive fission and corrosion products in the components of the coolant circuit as a function of power history.

INAPI calculates the inventories of radioactive nuclides created by activation of the coolant itself.

SHIELD is a simple shielding code based on point-kernel technique.

4.2. Essential results

5. Next steps. Further testing of the inventory calculations still has to be done in order to verify the models. A more sophisticated shielding code is being considered.

6. Relation with other projects. No formal relations to other projects are established, but interfaces exist with studies performed at Risø on the subjects: Systems delineation and power plant operation.

7. Reference documents Kurt Lauridsen, Development of a Model for the Assessment of Radiation Fields Around Nuclear Power Plant Components, Risø Report No. 353 (1977), 106 pp.

8. Availability The project information is freely available.

Berichtszeitraum/Period 1. 1. 77 - 31. 12. 77	Klassifikation/Classification 5.1	Kennzeichen/Project Number RS 285
Vorhaben/Project Title Untersuchung der Jod-Exhalation aus UO_2 unter stationären und transienten Bedingungen Investigation of the Iodine Release from UO_2 Under Steady-state and Transient Conditions		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik RB 33, Erlangen
Arbeitsbeginn/Initiated 1. 12. 77	Arbeitsende/Completed 31. 8. 80	Leiter des Vorhabens/Project Leader Dr. Peehs
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 77	Bewilligte Mittel/Funds 639.580,-- DM

1. General Aim

In LWR's, the amount of fission-Cesium generated during burn up is approx. 10 times higher than that of fission Iodine. Since Cs and Iodine form a more stable Iodide than the comparable Zirconium-Iodides, it is necessary to evaluate the related chemical equilibria based on the actual Cs/I-release to evaluate the effect of Iodine on the loss of ductility of Zr-cladding.

After experimental data on the Iodine and Cesium release from fuel are obtained, it will be possible to determine the effect of Iodine-caused loss of ductility on the fuel rod performance during transient and especially LOCA conditions. By using the computed Cs/I-inventory of the fuel, the measured release characteristics the results from SSCC of Zry.

2. Particular Objectives

The fission Cesium and Iodine integrated in the UO_2 lattice can only migrate to a free surface (outer and inner pellet surface and/or fracture) by diffusion.

It is the objective of this program to investigate on the I/Cs release characteristics under the following conditions:

- I/Cs release under steady-state conditions (isochronous-isothermal heat-up)

- I/Cs release under transient temperature conditions (transient heat-up).

In conjunction with

- tests performed at PNS/GfK on MCA burst performance,
- tests performed at KWU on unirradiated cladding tubes,
- PNS/GfK results obtained from cladding tubes doped with varying amounts of Iodine, and also
- in-pile tests to be performed under PNS 4237 scope,

the above test results will describe the cladding tube behaviour from fuel rods in different burn up stays under LOCA conditions.

3. <u>Research Program</u>

The Research Program is broken into the following items:

3.1 Thermodynamic analysis of the Systems U-O-I-Cs-Zr to estimate the degree of stability of possible I- and Cs compounds under LOCA conditions using basic data from the literature.

3.2 Providing measurement techniques in order to determine the Iodine migration, under the aspect of handling hot fuel.

3.3 Determination of I- and I/Cs penetration rates of UO_2 with different pore structure of temperatures of 650 - 1100 ^oK.

3.4 Selection and characterization of spent UO_2-archive pellets for the investigation of the Cs/I-release.

3.5 Determination of the Iodine migration from UO_2 samples with burnup as test parameter.

3.6 Characterization of UO_2 fractures for various UO_2 qualities under as-supplied and spent condition.

3.7 Determination of I-migration from UO_2 fractures with burnup as parameter.

4. Experimental Facilities

Under the scope of the above program, a test equipment is planned to be installed for determination of the I/Cs migration in irradiated UO_2.

5. Progress to Date

Construction work on the above test equipment was started.

6. Results

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7. Next Steps

Construction work will be continued. Thermodynamic analysis of the U-O-I-Cs-Zr Systems will be initiated.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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		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/TEGACH
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 circuits 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 SENA.

IV - PROJECT STATUS

4.1 - Progress to date

- début des essais sur SEPAI,

- campagne de mesures sur SENA.

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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.

149-1 -02		5-1
Titre Etude théorique de la contamination du circuit primaire des réacteurs à eau pressurisée.		Pays FRANCE Organisme directeur CEA/DSN
Titre (anglais) Theoretical study of the contamination of PWR primary circuit.		Organisme exécuteur CEA/DRE/SERMA Responsable /SERMA DSN/SESRS/FAF
Date de démarrage 1/01/1975	Etat actuel En cours	Scientifiques
Date prévue d'achèvement 31/12/1980	Dernière mise à jour 31/12/1977	

1 - OBJECTIF GENERAL

Modéliser les phénomènes conduisant à la contamination du circuit primaire. Elaborer les codes de calcul correspondants en vue de pouvoir effectuer des calculs prévisionnels.

2 - OBJECTIFS PARTICULIERS

a) Amélioration de la connaissance des coefficients de relâchement des produits de fission hors des assemblages de combustible (oxyde et jeu oxyde-gaine). Paramétrisation de ces coefficients en fonction de la température, du taux de combustion. Etude de l'effet des régimes transitoires.

b) Réalisation de calculs de référence sur la contamination du circuit primaire des réacteurs de puissance (900 MWe, 1 300 MWe).

c) Amélioration des calculs de production de tritium dans le fluide caloporteur.

d) Etude de l'effet de l'irradiation neutronique sur la migration du tritium.

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3 - INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

Cette étude utilise les résultats des expériences dans la boucle BOUFFON (CEN/G) ; les codes de calcul mis au point seront qualifiés à partir des programmes de mesure de suivi de la contamination dans les réacteurs en fontionnement, notamment FESSENHEIM.

4 - ETAT DE L'ETUDE

1. Avancement à ce jour

Mise au point d'un modèle semi-empirique permettant l'interprétation des expériences BOUFFON. Un modèle de diffusion des produits de fission avec piégeage est en cours d'étude et de test en vue d'affiner la représentation des phénomènes de migration.

2. Principaux résultats

. Détermination des coefficients de sortie des produits de fission hors du combustible dans l'expérience Cyrano.

. Variation des paramètres gouvernant le relâchement en fonction de la température.

. Confrontation avec les mesures d'activité du circuit primaire à TIHANGE.

. Etude bibliographique des rendements de fission ternaire du tritium.

. Interprétation du bilan des rejets de tritium à TIHANGE - Evaluation du taux de fuite par diffusion à travers les gaines.

5 - PROCHAINES ETAPES

. Interprétation des expériences BOUFFON (régime stationnaire et cyclage de puissance), en vue de qualifier le modèle de diffusion des produits de fission avec piégeage.

. Introduction du modèle dans le code de calcul PROFIP 3 qui traite la contamination du circuit primaire. Test sur les résultats du suivi FESSENHEIM.

. Etude critique des sections efficaces de production de tritium sur le bore et calcul fin du taux de production dans le fluide caloporteur.

6 - RELATION AVEC D'AUTRES ETUDES

- Expérience de fonctionnement PWR : synthèse des campagnes de mesures concernant les transferts de radioactivité. fiche 149-1-03
- Synthèse des programmes d'étude sur la contamination des circuits primaires de PWR. fiche 149-1-01
- Ruptures de gaine et dissémination des produits de fission dans les réacteurs à eau - Programme BOUFFON. 411 2-C

7 - DOCUMENTS DE REFERENCE : rapports internes

149-1 -03		5-1
Titre Expérience de fonctionnement PWR : Synthèse des campagnes de mesures concernant les transferts de radioactivité.		Pays FRANCE Organisme directeur CEA/ DSN
Titre (anglais) Operating experience in PWR		Organisme exécuteur CEA-DSN/SESRS Responsable DSN/SESRS/FAR
Date de démarrage 1977	Etat actuel en cours	Scientifiques
Date prévue d'achèvement 1982	Dernière mise à jour novembre 1977	

1 - Objectif général

Le but essentiel de cette étude est une meilleure connaissance de l'origine, la nature, l'importance et la répartition de l'activité dans les circuits primaires des réacteurs et dans les enceintes qui les entourent. La détermination des mécanismes de production et de transferts des isotopes radioactifs dans les différents milieux se fait par mesure in situ. Les résultats doivent eux-mêmes être utilisés pour qualifier des codes de calcul représentatifs des phénomènes.

2 - OBJECTIFS PARTICULIERS

Détermination des sources radioactives en fonctionnement normal ou en cas d'incident qui sont à l'origine :

- de la contamination des locaux et enceintes
- des rejets gazeux vers l'environnement
- des effluents liquides destinés à être traités
- des rejets liquides après traitement
- de la production de déchets solides.

Toutes ces informations constituent un support indispensable à l'analyse de sûreté. L'interprétation des résultats devrait permettre en particulier d'évaluer :

- le débit de fuite primaire dans le bâtiment réacteur
- le débit de fuite primaire-secondaire
- les performances des circuits d'épuration

.../

3 - Installations expérimentales et programme :

Néant

4 - Etat de l'étude

1) Avancement à ce jour

Les mesures effectuées à la CNA ont permis de mettre au point les techniques expérimentales.
Une première campagne de mesures a été effectuée sur Fessenheim I par les équipes expérimentales du DSN/SESTR et du DRE/SEN.
Sur Tihange le DRE/SEN suit l'évolution de l'activité de l'eau primaire (produits de fission et produits de corrosion).

2) Résultats essentiels

- Les résultats des premières mesures sur Fessenheim I, qui doivent être publiés prochainement après accord de l'exploitant, ont mis en évidence :

  . le faible niveau d'activité de l'eau primaire. L'activité des produits de fission dans l'eau primaire a permis d'évaluer la contamination initiale des gaines.

  . la présence de tritium gaz dans l'enceinte réacteur et dans l'air rejeté à la cheminée.

- Les mesures sur Tihange ont mis en évidence des ruptures de gaine ; l'activité primaire reste stable en fonctionnement à puissance constante. Au cours de la baisse de charge et du refroidissement du réacteur, on observe deux pics de remontée d'activité pour l'iode 131.

5 - Prochaines étapes

Interprétation des résultats obtenus à Fessenheim.
Lancement d'expériences analogues à Bugey 2.

6 - Relations avec d'autres études

L'étude de fonctionnement PWR et les études menées sur boucles sont complémentaires pour comprendre les mécanismes qui régissent le transport des produits de fission et des produits de corrosion. La présente étude est donc directement liée aux études faites sur boucles (expériences BOUFFON pour les PF, boucles HPZ, BIHAN, CIRENE pour les produits de corrosion).

7 - Documents de référence : rapports internes non disponibles .

149-I -05		5-I
Titre DEPOT-JET : étude du dépôt des produits de fission issus d'une rupture de gaine sur les structures du circuit primaire d'un réacteur à eau ordinaire.		Pays FRANCE Organisme directeur CEA/DgCS et EdF/SEPTEN
Titre (anglais) DEPOT-JET : research on deposition of fission products issued from a cladding defect on the primary circuit structures in a pressurized water reactor.		Organisme exécuteur CEA/DMECN/DMG (GRENOBLE) Responsable DMG - Grenoble
Date de démarrage 1/1/77	Etat actuel en cours	Scientifiques
Date prévue d'achèvement 31/12/81	Dernière mise à jour 19/12/77	

1 - Objectif général :

Etudier le taux de dépôt des produits de fission issus d'une rupture de gaine sur les structures du circuit primaire, la nature de ces dépôts et leurs combinaisons éventuelles avec les produits de corrosion du circuit primaire.

2 - Objectifs particuliers :

1 - Mise au point d'une boucle représentative des conditions thermohydrauliques rencontrées dans un réacteur.

2 - Simulation des principales zones de dépôt (structures) observées dans un réacteur.

3 - Etude de l'influence des additifs présents dans l'eau du circuit primaire (bore et lithine notamment) sur la formation des dépôts.

4 - Etude de l'influence de la présence de produits de corrosion et des composés formés avec ces derniers.

.../

3 - Installations expérimentales et programme :

Ces essais sont réalisés dans un dispositif BOUFFON-JET pourvu d'un thermosiphon et accéléré par un jet de vapeur, installé dans le réacteur SILOE à Grenoble.

4 - Etat de l'étude :

1) Avancement à ce jour

La mise au point du dispositif est en cours.

2) Résultats essentiels

Sans objet dans cette phase préliminaire du programme.

5 - Prochaines étapes :

Le premier essai est prévu au deuxième semestre de 1978.

6 - Relation avec d'autres études :

Cette étude doit être utilisée pour l'analyse de sûreté des réacteurs à eau ordinaire sous pression (évaluation des conséquences des ruptures de gaine sur la contamination permanente du circuit primaire). Elle est, d'autre part, en relation avec les programmes EDITH, CYFON, CRUCIFON et FLASH.

7 - Documents de référence :

Aucun document n'a encore été publié sur cette partie du programme.

149-1 - 07		5.1
Titre CONTAMINATION DES CIRCUITS PRIMAIRES - SUIVI DE FESSENHEIM		Pays : FRANCE
		Organisme directeur CEA/D.S.N.
Titre (anglais) PRIMARY CIRCUIT CONTAMINATION : ACTIVITY AND RADIATION FIELD SURVEY AT THE FESSENHEIM REACTOR .		Organisme exécuteur DRE/SEN - CEN/Cad.
		Responsable - id -
Date de démarrage : 1977	Etat actuel : En cours cf. Note Technique SEN/LFR/77-39	Scientifiques :
Date prévue d'achèvement : 6° cycle de FESSENHEIM	Dernière mise à jour 1/78	

1 - OBJECTIF GENERAL -

. Meilleure compréhension des comportements des produits de corrosion et de fission devant aboutir à :

- d'une part à une réduction de l'exposition du personnel sur les réacteurs (sur les réacteurs actuels en élaborant des consignes d'exploitation et sur les réacteurs futurs - modification et amélioration -).

- d'autre part d'ajuster les calculs prévisionnels et tendre vers plus de réalisme en ce qui concerne les sources de rejet en situation normale ou accidentelle.

2 - OBJECTIFS PARTICULIERS

1) - Campagnes de mesures sur la centrale

2) - Qualification et amélioration des codes de calcul PACTOLE pour les P.C. , PROFIP pour les P.F.

3 - PROGRAMME -

. Suivi des produits de corrosion et de fission dans l'eau du réacteur.

Au moins trois campagnes de mesures sont prévues par cycle de combustible (plus éventuellement des campagnes exceptionnelles en cas de ruptures de gaines Ces mesures portent sur la détermination d'activité de l'eau et des filtres et des teneurs en éléments.

. Interprétation des résultats en provenance de la centrale (mesure de routine) et des campagnes décrites ci-dessus.

. Mesures de l'activité déposée sur les tuyauteries primaires, les tubes G.V. et autres portions du circuit (BAN, etc).

4 - ETAT DE L'ETUDE -

1) - Une campagne réalisée (en début de cycle) du 2 au 11.9.77.

Les résultats de cette campagne sont donnés dans la Note Technique SEN/LPR/77-39 de Novembre 1977.

2) - Cette campagne a permis de montrer qu'actuellement l'activité des P.F. n'est due qu'à la contamination des gaines (pas de rupture de gaine).

5 - PROCHAINE ETAPE -

En Février 1978 des prélèvements sont prévus sur la ligne de charge pour déterminer l'influence des stellites et du RCV.

Deux autres campagnes de mesures sont prévues, à mi-cycle et en fin de cycle (ou à une autre date si l'évolution de l'activité P.F. indique l'apparition de ruptures de gaines).

6 - <u>RELATIONS AVEC D'AUTRES ETUDES</u> -

Voir fiches 149-1-02, 03, 05, 06 .

7 - <u>DOCUMENTS DE REFERENCE</u> : rapports internes non disponibles .

140-1 -02		5 -1
Titre EDITH : évaluation du taux de relâchement des produits de fission à partir d'une rupture de gaine dans un réacteur à eau ordinaire en fonction de la puissance linéaire.		Pays FRANCE Organisme directeur CEA/DgCS et EDF
Titre (anglais) EDITH : assessment of the fission products release rate from a cladding defect in a pressurized water reactor in terms of the linear heat rating		Organisme exécuteur CEA/DMECN/DMG Grenoble Responsable DMG - Grenoble
Date de démarrage 1/1/76	Etat actuel en cours	Scientifiques
Date prévue d'achèvement 31/12/78	Dernière mise à jour 12/77	

1 - Objectif général :

Etablir une relation entre le taux de dégagement des produits de fission hors d'un crayon présentant une rupture de gaine de section donnée et située au droit d'un bouchon, et la puissance linéaire de ce crayon en fonctionnement normal, en début de vie et en régime stable (20 000 à 40 000 $W.m^{-1}$).

2 - Objectifs particuliers :

1 - Pour un état donné de la charge de combustible d'un réacteur, évaluer le degré de contamination du circuit primaire à attendre en fonction de la puissance.

2 - Etudier la dégradation d'un crayon présentant une rupture de gaine en milieu aqueux.

3 - Modéliser les phénomènes observés.

3 - Installations expérimentales et programme :

Ces essais sont réalisés dans le dispositif BOUFFON installé dans le réacteur SILOE à Grenoble. Il s'agit d'un bouilleur à thermosiphon en circuit fermé comportant un système de prélèvement pour effectuer hors pile les mesures de contamination.

.../

4 - Etat de l'étude :

1 - Avancement à ce jour :

1) La phase expérimentale du programme, comprenant l'irradiation des expériences J 51 et EDITH 1 et leur examen après irradiation, est terminée.

2) La synthèse et la modélisation des résultats obtenus sont en cours.

2 - Résultats essentiels :

Les principales conclusions tirées de ces essais sont les suivantes :

1) Le dégagement des produits de fission hors du crayon est d'autant plus important que la puissance est élevée.

2) Le taux de dégagement des halogènes est environ dix fois moins élevé que celui des gaz rares.

3) A basse puissance linéaire (20 000 $W.m^{-1}$), le comportement des halogènes et des gaz de fission se distingue de celui observé à puissance linéaire plus élevée. Les modes de transport hors du crayon sont probablement différents.

5 - Prochaines étapes :

Terminer la synthèse et la modélisation de ces résultats.

6 - Relation avec d'autres études :

Cette étude doit être utilisée pour l'analyse de sureté des réacteurs à eau sous pression (contamination du circuit primaire). Elle est, d'autre part, en relation avec les programmes CYFON, CRUCIFON, JET et FLASH.

7 - Documents de référence :

Rapports internes non disponibles

140-1 -03		5 -1
Titre CYFON : évaluation du taux de relâchement des produits de fission à partir d'une rupture de gaine dans un réacteur à eau ordinaire sous l'influence des cyclages de puissance.		Pays FRANCE Organisme directeur CEA/Dg.CS et EDF
Titre (anglais) CYFON : assessment of the fission products release rate from a cladding defect in a pressurized water reactor under power cycling operation.		Organisme exécuteur CEA/DMECN/DMG/Grenoble Responsable DMG - Grenoble
Date de démarrage 1/1/76	Etat actuel En cours	Scientifiques
Date prévue d'achèvement 31/12/79	Dernière mise à jour : 7/12/77	

1 - Objectif général :

Déterminer le taux de dégagement des produits de fission hors d'un crayon présentant une rupture de gaine de section donnée et située au droit d'un bouchon dans des conditions de cyclage représentatives du suivi du réseau (cyclages de type 15/7).

2 - Objectifs particuliers :

1) Pour un état donné de la charge de combustible d'un réacteur, évaluer le degré de contamination du circuit primaire à prévoir dans l'hypothèse d'un fonctionnement en suivi de réseau.

2) Etudier la dégradation d'un crayon présentant une rupture de gaine dans les mêmes conditions.

3) Modéliser les phénomènes observés.

3 - Installations expérimentales et programme :

Ces essais sont réalisés dans le dispositif BOUFFON installé dans le réacteur SILOE à Grenoble. Il s'agit d'un bouilleur à thermosiphon en circuit fermé comportant un système de prélèvement pour effectuer, hors pile, les mesures de contamination.

.../

4 - Etat de l'étude :

1 - Avancement à ce jour :

1) L'irradiation des deux expériences initialement prévues (CYFON 1 et CYFON 2) est terminée, ainsi que leurs examens après irradiation et le dépouillement de ces derniers.

2) La synthèse et la modélisation des résultats obtenus sont en cours.

2 - Résultats essentiels :

1) Les cyclages périodiques du type 15/7 provoquent des bouffées de produits de fission lors des variations de puissance. Ces bouffées se produisent à la baisse de puissance pour des cyclages entre 20 000 et 40 000 Wm^{-1} et à la montée pour des cyclages plus profonds (12 000 à 40 000 Wm^{-1}).

2) Les bouffées principales contiennent des gaz rares et des halogènes. les bouffées secondaires surtout des gaz rares.

3) Les activités des produits de fission ainsi dégagés représentent une part importante de la contamination du circuit primaire.

5 - Prochaines étapes :

1) Reprise, en 1978, d'une expérience analogue à CYFON 2, CYFON 3, pour préciser l'effet propre de la variation de puissance.

2) Irradiation d'un crayon long (type CAP) pour préciser l'effet de la géométrie.

3) Synthèse et modélisation des résultats obtenus.

6 - Relation avec d'autres études :

Cette étude doit être utilisée pour l'analyse de sûreté des réacteurs à eau ordinaire sous pression (contamination du circuit primaire) pour le cas où un fonctionnement suivi du réseau serait envisagé. Elle est, d'autre part, en relation avec les programmes EDITH, CRUCIFON, JET et FLASH.

7 - Documents de référence :

Rapports internes non disponibles .

140-1 -04		5 -1
Titre CRUCIFON : évaluation du taux de relâchement des produits de fission hors d'une rupture de gaine déclenchée en fonctionnement normal dans un réacteur à eau sous pression		Pays FRANCE Organisme directeur CEA/DgCS et EdF
Titre (anglais) CRUCIFON : assessment of the fission products release rate from a cladding defect triggered under normal operation in a pressurized water reactor.		Organisme exécuteur CEA/DMECN/DMG ~~GRENOBLE~~ Responsable DMG - Grenoble
Date de démarrage 1/1/77	Etat actuel en cours	Scientifiques
Date prévue d'achèvement 31/12/81	Dernière mise à jour 16/12/77	

1 - Objectif général :

Déterminer le taux de dégagement des produits de fission hors de crayons préirradiés à différents taux de combustion représentatifs du début de vie (2000 MWJ/T) et la fin de vie (au-delà de 20000 MWJ/T) subissant une rupture de gaine en fonctionnement normal.

2 - Objectifs particuliers :

1 - Déterminer l'influence du taux de combustion.

2 - Déterminer l'influence du niveau de puissance linéaire en palier avant l'essai (25000 et 40000 $W.m^{-1}$).

3 - Déterminer l'influence de paramètres géométriques (jeu résiduel, longueur du crayon).

.../

3 - Installations expérimentales et programme :

Ces essais sont réalisés dans le dispositif BOUFFON installé dans le réacteur SILOE à Grenoble. Il s'agit d'un bouilleur à thermosiphon en circuit fermé comportant un système de prélèvement pour effectuer, hors pile, les mesures de contamination.

4 - Etat de l'étude :

1) Avancement à ce jour

Deux essais ont déjà été réalisés : CRUCIFON 1 et CRUCIFON 2.

2) Résultats essentiels

Quoique aucune synthèse n'ait encore été faite, dans l'attente des irradiations suivantes, on constate une décroissance du taux d'émission des isotopes à vie courte (inférieure à 3mn) après l'apparition de la rupture. Ceci suggère un rebouchage partiel de cette dernière par foisonnement de l'oxyde sous-jacent.

5 - Prochaines étapes :

Deux autres essais sont prévus en 1978 : CRUCIFON 3 et CRUCIFON 4. Ultérieurement, les deux essais à fort taux de combustion et utilisant des crayons longs (type CAP) seront réalisés.

6 - Relation avec d'autres études :

Cette étude doit être utilisée pour l'analyse de sûreté des réacteurs à eau ordinaire sous pression (évaluation de la contamination du circuit primaire par les produits de fission, en fonction du taux de rupture de gaine). Elle est, d'autre part, en relation avec les programmes EDITH, CYFON, DEPOT-JET et FLASH.

7 - Documents de référence :

Aucun document n'a encore été publié sur cette partie du programme.

143-1 -12/4112-01		5.1
Titre Investigation et développement des méthodes d'identification et de localisation des défauts de gaine pour les réacteurs à eau.		Pays FRANCE Organisme directeur CEA /DgCS
Titre (anglais) Investigation and development of identification and localisation method of clad failure for PWR.		Organisme exécuteur CEA/SES-SAI Saclay Responsable SAI - Saclay
Date de démarrage 01/1/76	Etat actuel en cours	Scientifiques
Date prévue d'achèvement 01/12/79	Dernière mise à jour 12/77	

1 - Objectif général :

Recherche d'une méthode qui permette de distinguer une augmentation d'activité due à l'évolution d'une rupture ancienne de celle due à l'apparition d'une nouvelle rupture.

2 - Objectifs particuliers :

Pour avoir un moyen d'évaluer l'évolution probable d'une rupture il est nécessaire d'effectuer une analyse quantitative de la distribution des produits de fission émis en fonction du type de rupture (rupture mouillée, fissure, microfuite).
La méthode de localisation prévue, consiste à corréler les signaux délivrés par des détecteurs de produits de fission installés sur chaque boucle primaire du réacteur (Bugey).

.../

4 - Etat de l'étude :

- Programme de calcul de la concentration des émetteurs à neutrons différés en tenant compte des coefficients de mélange de l'eau dans les boucles. Une série de calculs a été faite avec les résultats partiels fournis par EDF, et a été complétée pour obtenir directement le nombre de neutrons émis par centimètre cube de caloporteur dans le bloc de détection. Ceci permettra d'exploiter directement les essais sur la maquette expérimentale installée sur le réacteur Bugey IV
- La réalisation des blocs détecteurs de Bugey IV est terminée mais leur mise en place a été retardée pour l'intégrer au calendrier prévu pour le reste de l'installation.
- Un appareil capable de suivre en continu seize voies gamma à partir d'une diode au germanium lithium a été mis au point (note technique à paraître) pour suivre les activités de produits de fission afin d'améliorer le système actuel qui mesure le rayonnement global.

5 - Prochaines étapes :

- Installation d'un appareillage expérimental sur le réacteur PWR de Bugey IV et suivi des résultats afin de vérifier la validité des modèles mathématiques mis au point pour la détection puis la localisation de la rupture de gaine.

- Mise au point sur la boucle Isabelle du nouveau système équipé permettant de suivre 16 raies Gamma

7 - Documents de référence : - rapports internes non disponibles .

149-1- 06/4165-10 159-1- 04	5.I X 10.4
Titre : TRANSFERT DE LA CONTAMINATION DANS LES REACTEURS EN SERVICE	Pays FRANCE Organisme directeur CEA/DSN
Titre (anglais) TRANSFERRING OF CONTAMINATION IN OPERATING REACTOR	Organisme exécuteur CEA/DSN/SESTR/CADARACHE Responsable DSN/SESTR

Date de démarrage 01/09/73	Etat actuel EN COURS	Scientifiques
Date prévue d'achèvement 1981	Dernière mise à jour 30/11/77	

1 - OBJECTIF GENERAL -

Obtenir des informations sur le transfert de la contamination par les produits de fission à travers les barrières successives existantes entre le combustible et l'environnement immédiat d'un réacteur de puissance en fonctionnement normal .

2 - OBJECTIFS PARTICULIERS

1- Etude du transfert de la contamination auprès des centrales nucléaires à eau légère : FESSENHEIM 1, BUGEY 2, et, de la centrale à neutrons rapides:PHENIX .

2- Mise au point de techniques de prélèvement et d'analyses permettant :

- la mesure des produits de fissions solides et gazeux présents dans l'eau ou la vapeur des circuits primaires des centrales à eau légère .
- la mesure des gaz rares de fission et d'activation : du tritium, des différentes formes d'iode présents dans l'air de l'enceinte des réacteurs ou dans l'argon de couverture du coeur dans le cas de Phénix .

3 - INSTALLATIONS EXPERIMENTALES ET PROGRAMMES -

Les installations expérimentales comprennent :

- Des dispositifs de prélèvement et d'analyse permettant d'effectuer des mesures de produits de fissions présents dans le circuit primaire, le circuit secondaire, la piscine de désactivation, les circuits d'épuration et l'air des enceintes .
- Des montages expérimentaux en laboratoire permettant la mise au point des techniques de mesure qui doivent être utilisées en particulier pour l'analyse de faibles concentrations d'éléments radioactifs .

.../...

4 - ETAT DE L'ETUDE -

- Avancement à ce jour

1. Filière à eau légère -

Centrale de CHOOZ Mesures du taux de fuite du circuit primaire vers l'enceinte du réacteur.
Mesure de l'efficacité des circuits d'épuration :

Evaluation des taux de fuite de la barrière primaire vers l'enceinte du réacteur pour les gaz de fission, le tritium et les iodes.

Détermination de l'efficacité des déminéraliseurs et de l'évaporateur de la station de traitement des effluents.

Centrale de FESSENHEIM 1 :

Elaboration d'un programme de mesures.

Campagnes systématiques effectuées depuis le démarrage (mai 1977).

Mesure du taux de contamination du circuit primaire due à la pollution initiale des gaines des éléments combustibles.

2. Filière rapide (Phénix) -

- Bilan du transfert de la contamination dans l'environnement et bilan des effluents liquides.
- Bilan des effluents depuis le démarrage jusqu'au 30 Juin 1976.
- Bilan de l'activité en tritium dans les circuits primaires, secondaires et l'air des bâtiments.

5 - PROCHAINES ETAPES -

- Mesures systématiques à Phénix et à Fessenheim 1 et à Bugey 2.
- Interprétation des résultats à la qualification des codes de calcul, en particulier le code ALICE.

6 - RELATION AVEC D'AUTRES ETUDES -

Migration de la contamination dans les piles à eau ordinaire en situation accidentelle, fiche 148-1-03

Contamination circuit primaire réacteurs rapides, 159-1-0

Code de calcul ALICE, fiche 148-2-01

Expérience de fonctionnement PWR, fiche 149-1-03

7 - DOCUMENTS DE REFERENCE : rapports internes non disponibles

623

141-3 -09/4112-60 151-3 -04	12.3 * 5.1.3

Titre Utilisation de la neutronographie pour l'inspection en service des réacteurs nucléaires et le contrôle non destructif des composants du coeur.	**Pays** FRANCE **Organisme directeur** CEA
Titre (anglais) Use of neutro radiography for in-service inspection of nuclear reactor and for non destructive testing of core components.	**Organisme exécuteur** CEA/DSN/SEESNC VALDUC **Responsable** DSN/SEESNC/Valduc

Date de démarrage 01/01/73	**Etat actuel** étude en cours	**Scientifiques**
Date prévue d'achèvement 31/12/78	**Dernière mise à jour** 15/12/77	

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1977	5.2	PNS 4315 (4243)

Vorhaben/Project Title	
Experiments on Determination and Limitation of Fission and Activation Product Release During Core Meltdown Versuche zur Erfassung und Freisetzung von Spalt- und Aktivierungsprodukten beim Kernschmelzen	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Kernforschungszentrum Karlsruhe, Projekt Nukleare Sicherheit, IRCH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1973	1978	Dr. H. Albrecht
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	Dec. 1977	

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 melt, including also concrete; characterization of the physical and chemical behavior of the released products; development of techniques for reducing the release.

3. Research Program

3.1 Experiments with 30 g of inactive Corium to investigate the melt behavior during induction heating in a ThO_2 crucible and to measure the release-fractions of the main components of the melt (Fe, Cr, Mn, Ni, Zr, U) as a function of temperature, atmosphere, and pressure.

3.2 Experiments with 30 g of Corium containing activated steel and Zircaloy for preparation of tests with active Fissium (see 3.3) and to measure release fractions of those elements which can not be analyzed precisely in the tests with inactive Corium (e.g. Sn, Zr).

3.3 Release experiments with masses of 30 g - 3 kg Corium containing slightly active Fissium with simulated burn-up in the range of 10.000 - 50.000 MWd/t; same parameters as under 3.1.

3.4 Release experiments with Fissium-Corium and additions of CaO, SiO_2, concrete, and other materials.

4. Experimental Facilities, Computer Codes

- Melting furnace SASCHA
- transport and collection system for the released products
- facility for production of slightly active Fissium (FIFA)
- computer code GAMMA9 for evaluation of γ-spectra with respect to activation analysis.

5. Progress to Date

5.1 After completing the experiments according to 3.1, preparations were made for conducting on-line measurements of released radioactive fission and activation products. For this purpose, a Ge(Li)-detector was installed at the rear side of the primary filter. Gamma ray spectra are then collected for time intervals of 50-100 sec during the release experiments and the activity of each spectrum is related to a corresponding temperature interval of the melt. In addition, the radioactivity of the melt, of the deposits on the walls of the transport system and on the filters is analyzed after each experiment.
First tests according to 3.2 and 3.3 were conducted.

5.2 In some additional experiments with inactive corium, the size distribution and elemental composition of the aerosol particles was investigated. A fraction of the effluent stream was directed to a spiral duct centrifuge which deposits the particles on a removable band as a function of particle mass. The shape and elemental composition of these particles were analyzed by electron scanning microscopy and X-ray microanalysis.

6. Results

6.1 The release fractions of the elements Cs, Se, Te, Cd, Mo, and Sb as a function of temperature is shown in Fig. 4315-1. Theses curves were gained from 2 melting tests with 16 g steel, 3 g Zircaloy and 10 g fissium in air (1.5 bar). More than 70 % of the released activity was found on the membrane filters in a distance of about 1 m from the melting crucible. In previous experiments, the elements Fe, Cr, Co, Zr, and U were found to deposit predominately on the glass vessel directly above the crucible. That means that in the course of a core melt accident only a small fraction of the released low volatile species will be transported over longer distances from the melt - an aspect which is especially important for Pu and the other transuranium elements.

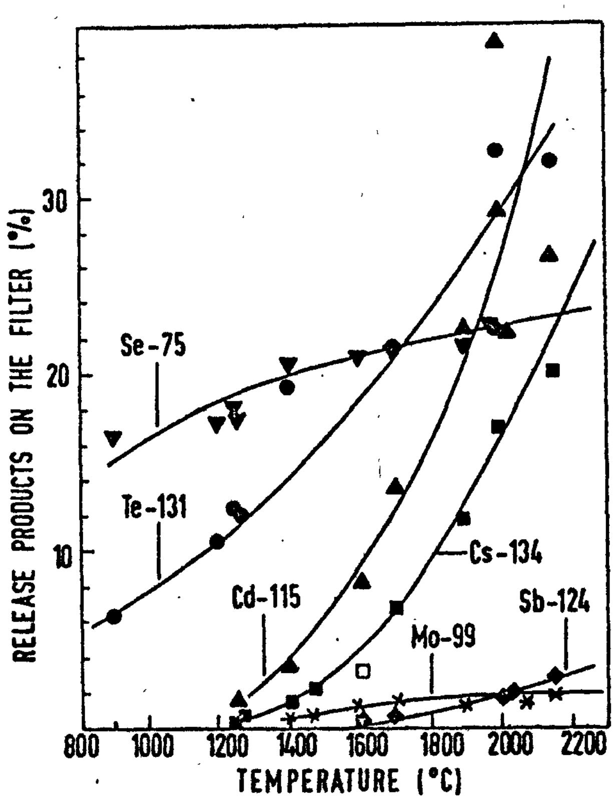

Fig. 4315-1: Fractional release of the fissium components as a function of temperature

6.2 The size distribution of the aerosols was found to be trimodal with maxima of 0.17 µm, 0.30 µm and 0.73 µm. The structure of the particles of the first and third maximum was compact, and their shape was nearly spherical. The particles of the middle maximum, however, had a complicated structure. They were generated by a coagulation process following the primary condensation.
The X-ray microanalysis gave the qualitative result that, in spite of the different geometric forms, the particles of all three maxima contained all main elements of the corium melt: Fe, Cr, Mn, Ni, and U.

7. Next steps

The most important work to be done in the near future will be
- additional release experiments with at broader spectrum of

fission products and variation of the parameters atmosphere, heating velocity, maximum temperature
- experiments with larger melt masses including also concrete.

8. Relation with other Projects

PNS 4316 (4243): Development and Opration of Facilities for Investigation of Fission Product Release during LWR Core Meltdown Accident

9. References

Report KFK-2435 (1977) p. 90 (in English)
p. 400 (in German)
Report KFK-2500 (1977) p. (in English)
p. (in German)

H. Albrecht, V. Matschoß, H. Wild, M.F. Osborne:
Experimental Determination of Fission and Activation Product Release During Core Meltdown, ANS Thermal Reactor Safety Meeting, Sun Valley, Idaho, 31.7.-4.8.77

10. Degree of Availability of the Reports

Unrestricted distribution

Berichtszeitraum/Period Jan. 1, 77 - Dec. 31,77	Klassifikation/Classification 5.2	Kennzeichen/Project Number PNS 4316(4243)
Vorhaben/Project Title Development and Operation of Facilities for the Investigation of Fission Product Release during LWR Core Meltdown Accident Aufbau und Betrieb von Anlagen zur Untersuchung der Spaltproduktfreisetzung beim LWR-Kernschmelzunfall		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor KfK Projekt Nukleare Sicherheit, RBT /IT
Arbeitsbeginn/Initiated 1972	Arbeitsende/Completed 1978	Leiter des Vorhabens/Project Leader D. Perinić
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1977	Bewilligte Mittel/Funds

1. General Aim

Investigation into the release and transport of radioactive materials under different core meltdown conditions.

2. Particular Objectives

Development of technical means for performing tests allowing to record and limit the release of fission and activation products during core meltdowns.

3. Research Program

3.1 Development of a melting facility for generating simulated core melts (CORIUM).

3.2 Development of a facility for generating nuclear fuel material with a simulated burnup (FISSIUM).

3.3 Development of a measuring technique for monitoring the molten pool temperatures.

3.4 Development of melting crucible configurations allowing to retain corium melts.

3.5 Operation of the experimental facilities.

4. Experimental Facilities

4.1 Melting facility for low activity specimens (SASCHA). It is heated by direct coupling of the melt in the induction field of the inductor. Working frequency 50 kHz, terminal power 250 kW, molten mass 5.0 kg. The facility is presently converted.

4.2 Vacuum melting furnace, resistance heated, max. 3000°C, 1 l volume.

4.3 Induction facility 500 kHz, 12 kW.

4.4 Facility for the fabrication of nuclear fuel with simulated burnup (FIFA). Nominal throughput per month 700 g of fissium pellets.

5. Progress to Date

ad 3.1

- Planning, fabrication and mounting of components used in the modification of the SASCHA facility.
- Measurements on the optimization of the oscillator circuit.
- Installation of supply lines for the cooling water, current and compressed air.
- Licensing procedure under the Atomic Energy Act.

ad 3.2

- Completion and commissioning of the FIFA facility.
- Sinter test for process optimization.
- Licensing procedure under the Atomic Energy Act.

ad 3.3

- Use of a light beam oscillograph for high speed recording of temperatures and balancing signals from the optical pyrometer.

ad 3.5

- Operation of the SASCHA facility.
- Inactive trial operation and active operation of the FIFA facility.

6. Results

ad 3.1

- The installation of the components of the facility has been completed. Preparations have been made for final acceptance of the facility.
- The authorization under the Atomic Energy Act for active operation of the facility and for the storage of radioactive melts has been received.

ad 3.2

- In a glovebox the device was completed for preparing (grinding, weighing) of fission product elements to be irradiated in FR2.
- The first active fissium pellets were fabricated. This means that the development and trial operation of the FIFA facility have been terminated.

ad 3.3

- To measure the surface temperatures of the corium melts a method was applied which can be used in an oxidizing and inert atmosphere up to temperatures of 3000°C. The influence exerted by errors in various test parameters was investigated. By use of the light beam oscillograph the true signal can be determined also in case of extreme fluctuations in brightness. The work has been completed.

ad 3.4

- The development of the melting crucible configuration has been completed for melts up to 0.5 kg. Crucibles are used which consist of two shells with a filler in between.

ad 3.5

- 35 tests were performed in the SASCHA facility without any faulty operations.
- Following an inactive trial operation the first two charges of fuel pellets with the active fission products added were fabricated at the FIFA facility.

7. Next Steps

ad 3.1

- Acceptance, trial operation of the SASCHA facility.

ad. 3.2

- Operation of the FIFA facility.

8. Relation with Other Projects

PNS 4315 (4243): Experiments on Determination and Limitation of Fission and Activation Product Release during Core Meltdown

9. References

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10. Degree of Availability of the Reports

-

146-2 -01		5.2
Titre Dénoyage des éléments combustibles irradiés au cours de leur stockage dans les centrales PWR : Détermination du scénario accidentel représentatif.		**Pays** FRANCE **Organisme directeur** CEA/DSN
Titre (anglais) Dry out of irradiated fuel elements in storage pool of PWR : determination of representative sequences.		**Organisme exécuteur** DSN/SETSSR/FONTENAY **Responsable** SETSSR
Date de démarrage 1/2/78	**Etat actuel** en cours	**Scientifiques**
Date prévue d'achèvement 12/80	**Dernière mise à jour** 1.2.78	

1 - Objectif général :

Détermination d'un scénario d'accident type entraînant le dénoyage des éléments combustibles irradiés stockés dans la piscine de désactivation afin de fournir les hypothèses réalistes les plus probables pour l'étude de l'évolution thermique des éléments avant et après dénoyage. Les résultats de cette étude seront utilisés pour l'évaluation des conséquences radiologiques de l'accident type sur les personnes physiques de la centrale et de l'environnement.

2 - Objectifs particuliers :

2-1) Détermination des divers scénarios d'accidents entraînant le dénoyage avec ou sans rupture des éléments combustibles. Analyse probabiliste de ces différents scénarios.

2-2) Choix d'un ou plusieurs types d'accidents réalistes.

.../

2-3) Etude en fonction du temps

- de la puissance résiduelle des éléments combustibles.
- de la composition pondérale en éléments radioactifs et de leur radioactivité associée.

2-4) Etude des conditions de refroidissement naturel en fonction

- du type de ruine de la piscine (disposition relative des différentes brèches).
- de la disposition et du nombre des éléments combustibles présents dans la piscine.

3 - Installations expérimentales et programmes :

Néant.

4 - Etat de l'étude :

Non démarrée. Les points 2-3 et 2-4 seront abordés en priorité.

143-1 -01		5-2
Titre FLASH : évaluation du taux d'émission des produits de fission au cours d'un accident de dépressurisation dans un réacteur à eau sous pression.		Pays FRANCE Organisme directeur CEA/DgCS et EDF/SEPTEN
Titre (anglais) FLASH : assessment of the fission products release rate during a loss of coolant accident in a pressurized water reactor.		Organisme exécuteur CEA/DMECN/DMG Grenoble Responsable DMG/Grenoble
Date de démarrage 1/1/77	Etat actuel en cours	Scientifiques
Date prévue d'achèvement 31/12/81	Dernière mise à jour 16/12/77	

1 - Objectif général :

Déterminer le taux de relâchement des produits de fission hors d'un crayon subissant une rupture de gaine lors d'un accident de dépressurisation et pour une température de gaine comprise entre 1000 et 1200 °C.

2 - Objectifs particuliers :

1 - Etudier le relâchement par lavage ou effet thermique des produits de fission condensables initialement piégés par dépôt dans le jeu.

2 - Etudier le relâchement éventuel des produits de fission gazeux.

3 - Etudier, si possible, le taux d'immobilisation des produits de fission sous forme de composés.

3 - Installations expérimentales et programme :

Ces essais sont réalisés dans un dispositif BOUFFON axial installé dans le réacteur SILOE à Grenoble.

.../

4 - Etat de l'étude :

1 - Avancement à ce jour :

L'étude neutronique et thermique du dispositif est terminée. Le dispositif sera opérationnel en 1978.

5 - Prochaines étapes :

La première irradiation est prévue au deuxième semestre de 1978.

6 - Relation avec d'autres études :

Cette étude doit être utilisée pour l'analyse de sureté des réacteurs à eau ordinaire sous pression (évaluation du terme source de contamination lors d'un accident de dépressurisation). Elle est, d'autre part, en relation avec les programmes EDITH, CYFON, CRUCIFON et DEPOT - JET.

N.V. KEMA	CLASSIFICATION : 5.2
TITLE : Berekening van hoeveelheden radioactiviteit vrijkomend bij een ernstig reactorongeval.	COUNTRY: THE NETHERLANDS
	SPONSOR : KEMA ORGANIZATION : KEMA
TITLE (ENGLISH LANGUAGE): Calculation of the quantities of radioactivity released as a result of a serious reactor accident	PROJECTLEADER :
INITIATED : - / LAST UPDATING : 1978	SCIENTISTS : K.P. Termaat
STATUS : - / COMPLETED : 1977	

General aim

In analysing the risk of a nuclear power plant for the surrounding population one has, among 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.

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.

Experimental facilities

Not applicable.

Project status

Necessary calculations have been performed. An improvement in applied input data to the programme CORRAL could be usefull.

Next steps

Not applicable.

Relation to other projects

Equivalent calculations are performed in the USA as a contribution to the Rasmussen-study (WASH-1400).

Reference documents

See 2 and 6.

Degree of availability

Internal report.

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	5.3	RS 209

Vorhaben/Project Title	
Aktivierte Korrosionsprodukte in LWR-Kreisläufen Activated Corrosion Products in LWR Loops	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 53, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 1. 76	31. 12. 78	Dr. Neeb
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 77	977.450,-- DM

1. <u>General Aim</u>

Development of realistic contamination models for LWR primary coolant circuits with the aim to reduce the activity level of the circuits and loops, i.e. to reduce irradiation exposure of the control and maintenance personnel in nuclear power plants.

2. <u>Particular Objectives</u>

Improvement and completion of our knowledge of the sources, formation mechanisms as well as the transportation and deposition behaviour of those radionuclides which are primarily responsible for contamination of circuits and systems of LWRs, i.e. for the local dose rates in the plant as well as for the activity inventory of radioactive wastes.

3. <u>Research Program</u>

3.1 Compilation and evaluation of operating data for PWRs and BWRs.

3.2 Data balancing in order to identify radionuclide sources.

3.3 Evaluation of the actual Co-contents in the construction materials.

3.4 Improvement of analytical methods.

3.5 Specific PWR tests.

3.5.1 Variation of operation parameters.

3.5.2 Exchange behaviour of deposit- and protective layers.

3.6 Specific BWR tests.

3.6.1 Compilation of loop surfaces of various materials and their metal erosion rates.

3.6.2 Contamination influence of high-Co-Containing materials in a neutron field.

3.7 List of PWR and BWR contamination models.

4. Experimental Facilities

The necessary test facilities for radio-chemistry, the analysis and measurement techniques, and the coolant chemistry and hot cell techniques are available. All measurements necessary will be performed in KWU laboratories.

Provision of various analysis samples (system- and fuel assembly deposits), the collection of several data points are closely tied in with reactor refuelling shutdowns.

5. Progress to Date

The Co-60 activity concentration in the coolant of different PWR's was examined in order to find a correlation between the Co-content of the steam generator tubes and the primary water.

Some methods were tested to analyze the Co-content in the deposits and primary water.

The specific Fe-55 and Co-60-activity of primary water probes of the Biblis A power station was measured. In the PWR-nuclear power stations KKB, GKN and Biblis A and the BWR power station KKB deposit probes were removed, which were in contact with primary water or steam.

6. Results

The Co-60 activity analysis showed that there exists no direct correlation between the Co-content of the steam generator tubes and the Co-60-content of the primary coolant.

The contamination mechanism proved to be very complex.

Unexpectedly no direct connection was found between the Co-58 and Co-60 activity concentration of the primary coolant and the corresponding activities in the deposit layers of the loops and systems.

The activation analysis was the most qualified method to determine the Co-content in the ng-region. The Fe-55 content was measured by a modified fluid-szintillation method. Another method was the analysis of the K-Yray line of Fe-55 by a Si (Li)-detector.

When the power in the Biblis A station decreased from 100 % to 0 %, connected with a temperature drop from 295 °C to 200 °C the Fe-55 activity increased slowly but the age of radioactivity was not influenced markedly. The age increased however, when pressure and temperature decreased simultaneously. In the ional dissolved corrosion-products the Fe-55 activity age was nearly constant.

The KKB probes showed that the Fe-55 activity age was less than one day. This means that the contamination of the surfaces in the BWR-loop is caused mainly by nonactive iron.

7. Next Steps

The specific Ni-63 activities will be analyzed. The chemical and radiochemical measurements of deposit probes will be continued.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	5.3	RS 204
Vorhaben/Project Title Dosisabbau Dose Reduction		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 312, Erlangen
Arbeitsbeginn/Initiated 1. 4. 76	Arbeitsende/Completed 30. 9. 73	Leiter des Vorhabens/Project Leader Dr. Wille
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 77	Bewilligte Mittel/Funds 1'036.900,-- DM

1. General Aim

Improvement of currently used units and development of advanced techniques to handle gaseous and liquid radioactive waste, their optimization and the application of both lab and operating plant experience to technically mature systems.

2. Particular Objectives

2.1 Gaseous Activities

In order to reduce radiation exposure, the quantities of long-lived nuclides which are released by leakage and upon reactor shutdown will be reduced as much as possible by continuous extraction of radioactive gases from the primary coolant.

In addition, further improvement of the measurement and control of hydrogen and oxygen in the off-gas system will be pursued.

2.2 Water-Soluble Activities

By use of a Caesium-specific Ion exchange resin, an increased service life and minimization of Cs-build-up in the primary coolant, thereby leading to a reduction of personnel exposure, will be achieved. In addition, waste water evaporator distillates from NPP will be decontaminated with I.T. to achieve optimum waste water purification.

2.3 Decontamination

Decontamination methods will be further developed to the point of application in a NPP, so as to reduce exposure of personnel during repair work.

3. Research Program

3.1 Gas Treatment

Laboratory test and evaluation of a noble gas separating test facility and development of an overall applicable system, examination of oxygen- and hydrogen measuring apparatus and selection of measurement procedure.

3.2 Water-Soluble Active Wastes

Examinations of decontamination of primary coolant and waste water evaporator distillates with filters and Ion-exchangers; magnitude and origin of corrosion products in the primary coolant (primary circuit and emergency systems).

3.3 Decontamination of Units and Containers

Decontamination of large-scale tanks in NPPs by means of special methods; preparation and treatment of decontamination solutions.

4. Experimental Facilities

Tests will be performed at model test stands available either in Erlangen or in NPPs.

5. Progress to Date

5.1 Gas Treatment

All control loops of the test plant were successfully tested; minor modifications to several control loops were made. Carrier gas mixture interrupt tests yielded H_2-concentrations of 85 - 95 %. In parallel initial tests were run with Krypton at a concentration range of 30 - 50 vpm in H_2-N_2 carrier gas streams.

Further tests serve both for the evaluation of absorption capacities under various constant operating parameters as well as the selection of the most favourable regeneration method. Also, with the installation of additional control devices, further plant upgrading measures were made for the preservation of constant operating conditions (output, pressure).

Several loading and regeneration tests were simultaneously performed with Krypton, with Xenon or both Krypton and Xenon in the carrier gas. The carrier gas stream consisted of either a nitrogen-hydrogen mixture or pure nitrogen gas.

In addition, performance tests were run with the refrigeration unit at various working pressures.

5.2 Water Soluble Active Waste

Under the scope of the above program, water samples were taken from various depths of the spent fuel pit at different periods of time during 1977 Stade refuelling. The aim of this measurement series is to determine the corrosion product yield and/or discharge into the spent fuel pit during core unloading.

Ion exchange resins were taken from a Neckarwestheim mixed bed filter which has been used for 1 year.

Also, tests on Cs-retention were performed with Zirconium-phosphate.

5.3 Decontamination

The chemical one-step method, based on the use of inhibited citrate-/fluoride solution, was used for decontamination of Gundremmingen I safety valves.

The metallographic examination of the gap surfaces, which was to show the effect of the oxidation- and/or decontamination solution under PWR conditions, was completed.

In order to confirm that the chemical two-step decontamination method on austenitic material neither generates selective corrosion nor enhances already present IC defects, metallographic examinations were performed on decontaminated, predammaged feedwater sparger ring samples of different lengths which were taken from NPP Gundremmingen I. Thereby special attention was paid to the influence of cold work.

The chemical two-step method was tested on a BWR reactor component from NPP Gundremmingen I by the decontaimination of the secondary side S.G. manhole cover.

6. Results

6.1 Gas Treatment

During interrupt, H_2-concentrations of 85 - 95 % were achieved by the test plant, starting with approx. 5 - 15 % in the H_2-N_2 carrier gas stream.

The loading- and regeneration tests showed that 2 different capacities must be assumed for the adsorber bed design:

1. in-service capacity serving as a delay line
2. in-service capacity serving as an adsorber, i.e. the absorbed substance in the carrier gas stream cannot or can only to a small extent be flushed from the adsorber.

The values determined in the tests have to be tested in terms of capacities suitably related to reactor operation. Thereby it was noted, however, that -because of measurement problems- the initial inert gas concentration was higher by a factor of 10 than the gas concentrations to be expected in a NPP.

During commercial operation, final concentrations of 95 - 98 % H_2 could be achieved during H_2 - N_2 separation, starting with concentrations of 4 - 30 % H_2 in N_2.

In general, findings were obtained on the static working capacity of the activated carbon, i.e. on the minimum amounts of Krypton and Xenon, which remain on the carbon even after several days flushing by an inert gas-free gas stream. By heating up the adsorber in a vacuum these inert gas amounts could be recovered. Based on these results an estimate could be prepared both on the carbon amount necessary for nuclear power plant operation and the yield of regenerated material amounts.

6.2 Water Soluble Wastes

The Neckarwestheim test series on Cs-retention on Zirconium phosphate did not render sufficient capacity for Cs retention.

6.3 Decontamination

During decontamination of the Gundremmingen I safety valves, the dose rate was reduced from 30 (mR/h) to 1 (mR/h) after a 3 hours decontamination treatment.

The metallographic test performed on decontaminated feed water sparger ring samples showed that new selective corrosion IC-attack of max. 20 /um increased, nor was there any evidence of cold-work influence.

Metallographic examination of the gap surfaces did not reveal any relevant difference between the samples in as-supplied condition and the treated ones.

A decontamination factor (F_D) of 10 was reached with the secondary side S.G. manhole top contamination after a twice-repeated decontamination treatment (oxidation of two hours each and decontamination of 6 hours) and an initial dose rate of 850 mR/h.

7. Next Steps

The inactive inert gas test series are planned to be repeated using the motive gas method, i.e. the max. gas throughput. Subsequently the plant will be prepared for active gas operation and then integrated into the radiochemistry control area.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period 1. 1. 77 - 31. 12. 77	Klassifikation/Classification 5.3	Kennzeichen/Project Number RS 238
Vorhaben/Project Title Entwicklung eines Systems zur Absaugung von Anlagenteilen und Armaturen Development of a Suction System for Installations and Fittings		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION Reaktortechnik R 312, Frankfurt
Arbeitsbeginn/Initiated 1. 10. 76	Arbeitsende/Completed 31. 3. 78	Leiter des Vorhabens/Project Leader H. Queiser
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 77	Bewilligte Mittel/Funds 184.000,-- DM

1. General Aim

Previously, there were no requirements for retention of noble gases comming up with the vent-gases of the liquid-waste-tanks and the gland-leak-off-system.

Due to the extremely low release rates now demanded by the licensing authorities, however, noble gas retention systems must be developed and connected into activity paths previously thought to be negligible as far as activity is concerned.

2. Particular Objectives

Investigations will be made on activity-release, -paths and -concentration as well as on the development of systems for activity collection and retention.

In particular, noble gas evolution and -release from iodine-containing ion-exchange resins, e.g. reactor water clean-up-filter, will be investigated for a certain period of time after resin discharge.

3. Research Program

3.1 Experimental determination of the noble gas release form ion-exchange resins in nuclear power plants during transportation of suspended loaded resins to the waste tanks and under the operating conditions present in these tanks (temperature, pressure, movement, filling- and discharge rhythm, etc.).

3.2 Verification of the calculated noble gas concentration and sources by measurements performed in the components of the water clean-up-systems in nuclear power plants.

3.3 Determination of iodine and noble gas paths in the reactorwater and -steam circuit and the auxilliary systems.

3.4 Determination of the fluid motions in the water decontamination systems with respect to quantities and time sequence.

3.5 Investigations on the gas flow in the water clean-up systems.

3.6 Investigations on possible gas-flow-retention within water clean-up system containers.

3.7 Experimental investigation of noble gas adsorption observed on charcoal under the following conditions:

a) loading with steam-saturated gas flows at higher temperatures

b) Operation of charcoal columns during alternating loads (counter flow) as well as different steam saturation degrees of gas flow observed at R.T. and elevated temperature.

c) loading with steam saturated gas flow at batch-wise operation and higher temperatures.

3.8 Experimental measurements and investigations of leakages observed in glands and leak-off systems as well as possible reduction of air in leakages and steam contents in a nuclear power plant in operation.

Concept and optimization of systems with charcoal filter designed for:

a) collection of gland leakages

b) treatment of motive air quantities from water clean-up system containers with higher activity
c) exhausting from the H_2 concentration limitation system
d) degasification of the primary circuit after plant shut down
e) coolant filter cleaning by means of purging air.

4. Test Facilities

Tests will be performed by the "Bergbauforschung/Essen GmbH".

5. Progress to Date

Concepts have been developed on the plant leak-off systems, consisting of a gland leak-off system, tank leak-off system and penetration-leak-off system, as well as control and operating concepts.

In the "Bergbauforschung/Essen" Labs pressure losses on charcoal adsorber with varying gas-flow rates were observed on water-saturated carbon.

Furthermore, loading- and backflush cycles were performed on the optimization of back flush quantities and backflush times.

6. Results

Tests on loading and backflush of activated carbon which were performed by the "Bergbauforschung/Essen", have almost been completed. The data currently prepared on systems analysis concepts of the plant ventilation are satisfactory. The technical design of the charcoal column is such that accumulation of activity sources can be avoided to a large extent.

7. Next Steps

The measurement data obtained during the tests are presently being evaluated and documented.

The results obtained are planned to be utilized for further development of the plant concept.

Also, further measurements are planned to be performed in nuclear power plant Würgassen on the determination of activity paths.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

653

Berichtszeitraum/Period 1.7. - 31.12.1977	Klassifikation/Classification 5.3	Kennzeichen/Project Number PNS 4111
Vorhaben/Project Title Entwicklung von Störfall-Umluftfiltern für Reaktorsicherheitsbehälter Post Accident Recirculation Air Cleanup Filters for Fission Product Removal from the Containment Atmosphere		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum GmbH Projekt Nukleare Sicherheit LAF II
Arbeitsbeginn/Initiated 1971	Arbeitsende/Completed 12/77	Leiter des Vorhabens/Project Leader H.-G. Wilhelm
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating December 1977	Bewilligte Mittel/Funds

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. **Research Program**

3.1 Testing of iodine sorption materials for operation under steam atmosphere at elevated pressure and at high temperature.

3.2 Testing of single technical components e. g. moisture separators, heaters, prefilters under simulated accident conditions.

3.3 Irradiation tests.

4. **Experimental Facilities**

to 3.1 and 3.2 Lab scale rig and technical test rig for testing of original filter components (iodine and aerosols) under simulated accident conditions.

5. Progress to date

to 3.2 A new droplet separator was constructed and built because the first one had a too high pressure drop.
A new Aerosol generator was built and first experiences are positiv.

The parametric studies for the construction of postaccident recirulating filters were newly calculated regarding the data of release, that were laid down by the German reactor safety commission in 1976 for accidental fission product release, radiation load and pressure drop.

6. Results

to 3.2 The specialy constructed metal-fibre droplet separator showed high reasonable performance in the particle sice range above 1 μm. The efficiency is normaly greater than 99,0 % by using 4 flats of 22 μm fibres.

7. Next steps

to 3.2 The measurements on droplet separators will be continued. The measurements on Aerosol filters under MCA conditions will begin 78.

8. Relation with Other Projects

Results of PNS 4114, part 2, may contribute to the program described here.

9. References

KFK 2500, S. 101 - 105

10. Degree of Availibility of the Reports

Unclassified, reports are available without restriction.

Berichtszeitraum/Period 1.7. - 31.12.1977	Klassifikation/Classification 5.3	Kennzeichen/Project Number PNS 4.41.6
Vorhaben/Project Title Abluftfilterung an Reaktoren, (Alterung und Vergiftung von Jod-Sorptionsmaterialien) Off Gas Filtering in Reactor Stations (Ageing and Poisoning of Iodine Sorption Materials)		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum GmbH Projekt Nukleare Sicherheit LAF II
Arbeitsbeginn/Initiated 1973	Arbeitsende/Completed 1979	Leiter des Vorhabens/Project Leader J. Furrer
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1977	Bewilligte Mittel/Funds

1. General Aim

Improvement of iodine sorption filters.

2. Particular Objectives

Development of improved iodine sorption filters for extended operational periods.

3. Research Program

3.1. Gas chromatographic analysis of the influent and effluent of iodine sorption filters in nuclear power plants (nukes). Determination of the dependency of the removal efficiency for radioiodine on the nature and loading of poisoning components.

3.2. Development of an improved iodine sorption filter including in-place regeneration of the iodine sorption material.

4. Experimental Facilities

to 3.1. Gas chromatographic equipment. Test rigs in nukes for iodine sorption materials. Test apparatus for measurements of the removal efficiencies of sorption materials under simulated conditions of filter operation.

to 3.2. Test rig for whole filter units for a throughput of up to 2000 m^3/h.

5. Progress to Date

to 3.1. Movable test rigs for checking the MWS filter system were built and connected to the exhaust air of the PWR Biblis B. (MWS-filter = multiple way sorption filter. This is a filter construction for increased time of useful operation regarding the poisoning of the impregnated activated charcoal).

The research program contains the qualitative and quantitative determination of the distribution of the poisoning materials in the charcoal layers of the filter and of the removal efficiency for radioiodine as a function of the operating time.

For the measurement of the concentration of filter poisons in the exhaust air a mobile continuously measuring GC was calibrated.

A test unit for the determination of the loading of the individual zones of the MWS filter was designed and will be built. The method for sampling different zones of the sorption material was developed.

to 3.2. A 1 : 1 prototype MWS-filter for a volumetric flow rate up to 1200 m^3/h was tested.

6. Results

to 3.1. After two months of operation of the activated and impregnated charcoal in two layers each of 25 cm bed depth, in the exhaust air of the shut off room of the PWR Biblis B, a solvent content up to 6.3 $^w/_o$ was observed on the first charcoal layer, which included high boiling components such as dodecan. The second charcoal layer was loaded with solvents up to 4.7 $^w/_o$, no high boiling components were detected in this layer.

After desorption of the relatively low boiling solvents by heating the second charcoal layer, the radioiodine removal efficiency of this layer was nearly restored.

to 3.2. The 1 : 1 sized prototype MWS-filter showed the decontamination factors expected from the removal efficiency of the activated impregnated charcoal. No problems occurred considering the exchange of the charcoal fill under dry and wet conditions.

7. Next Steps

to 3.1. Continuing of the laboratory experiments for the periodic desorption of poisoning materials. Continuous measurement of the organic components in the exhaust air of a PWR nuke. Further investigations to optimise the operation of the MWS-filter.

to 3.2. Testing of the original MWS-filters for nukes.

8. Relation to other Projects

Results of PNS 4415 may contribute to the program described here.

9. References

PNS-Semi Annual Report (KFK 2500), p. 116 - 118.

10. Reports and publications are available without restriction.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.7. - 31.12.1977	5.3	RS 221 / PNS 4414
Vorhaben/Project Title Bestimmung der Jodkomponenten in der Abluft kerntechnischer Anlagen Determination of the Iodine Species in the Exhaust Air of Nuclear Installations		**Land/Country** FRG **Fördernde Institution/Sponsor** BMFT **Auftragnehmer/Contractor** Kernforschungszentrum GmbH Projekt Nukleare Sicherheit LAF II
Arbeitsbeginn/Initiated 1.12.1976	**Arbeitsende/Completed** 30.11.1978	**Leiter des Vorhabens/Project Leader** H. Deuber
Stand der Arbeiten/Status Continuing	**Berichtsdatum/Last Updating** December 1977	**Bewilligte Mittel/Funds** RS 300.000,-- DM

1. General Aim

Improvement of the assessment of the environmental impact of radioiodine released with the exhaust air of nuclear installations; improvement of the ventilation conception of nuclear installations.

2. Particular Objectives

Determination of the radioiodine species elemental iodine (I_2), particulate iodine, and organic iodine (CH_3I).
(Assumed ratio of the thyroid doses caused by the release of equal amounts of these species in the BRD 100 : 10 : 1.)

3. Research Program

3.1. Laboratory tests: Development of selective sorption materials for radioiodine species samplers.

3.2. In situ tests: Operation of radioiodine species samplers in the exhaust air and stack discharge.

4. Experimentel Facilities

to 3.1. Apparatuse for the generation of radioiodine species and the testing of sorption materials within a wide range of conditions.

to 3.2. Rigs for the operation of radioiodine species samplers in the exhaust air and stack discharge.

5. Progress to Date

to 3.1. Ascertainment of the removal efficiencies of several iodine sorption materials for $^{131}I_2$, $CH_3{}^{131}I$, $H^{131}IO$ (hypoiodous acid), and $C_6H_5{}^{131}I$ (iodine benzene).

to 3.2. Measurements were performed with radioiodine species samplers in the stack discharge of a BWR, which had been shut down because of an accident. Moreover, radioiodine species samplers were employed in the hood exhaust air and stack discharge of PWRs (PWR 2 and PWR 3).

6. Results

to 3.1. Removal efficiency of the I_2 sorption material DSM 11 for $^{131}I_2$ and $CH_3{}^{131}I$ > 99,9 % and < 0,5 % respectively in the relevant range of parameters (10-70°C, 20-80 % R.H.) at a residence time of 0,1 s; for $H^{131}IO$ < 1 % at at 40°C, 50 % r.H., and the same residence time. Of the sorption materials tested, IPH (from NES, USA) proved to be the most suitable for selectively removing $H^{131}IO$ under the test conditions. Removal efficiency of DSM 11, AC 6120, and LMS 13X-Ag for $C_6H_5{}^{131}I$ very low, of activated charcoal impregnated with KI or TEDA very high.

to 3.2. The percentage of particulate ^{131}I was always very small (mostly < 1 %). The average proportion of elemental ^{131}I was 30 % in the stack discharge of both BWR and PWR 2 (measurement time 8 and 37 weeks respectively) and 60 % in the stack discharge of PWR 3 (measurement time 11 weeks). The hood exhaust air of PWR 3 contained 80 % elemental ^{131}I on the average and furnished 50 % of the elemental ^{131}I of the stack discharge (measurement time 9 weeks).

7. Next Steps

to 3.1. Determination of the removal efficiencies of DSM 11 and IPH for $H^{131}IO$ in the relevant range of parameters. Development of a radioiodine species sampler applicable to the exhaust air of reprocessing plants.

to 3.2. The measurements with radioiodine species samplers will be continued in the mentioned and other flows of exhaust air until a measurement time of 1 year (in the stack discharge) will be reached in each of the PWRs.

8. Relation with other Projects

Ageing and Poisoning of Iodine Sorption Materials.

9. References

Reaktortagung Mannheim, 29.3. - 1.4.1977, p. 813 - 816;
KFK 2405, p. 127 - 139;
KFK 2500, p. 119 - 140.

10. Degree of Availability of the Reports

Reaktortagung: ZAED, 7514 Eggenstein-Leopoldshafen;
KFK: Literature Department, KFK, Postfach 3640, 7500 Karlsruhe 1.

148-1 -02		5-3
Titre Accidents de réacteurs PWR - Transfert de la radioactivité à l'intérieur de la centrale et rejets hors confinement.		Pays France Organisme directeur CEA/ DSN
Titre (anglais) Accidents of PWR'S - Transfer of radioactivity in the plant and release out the containment.		Organisme exécuteur CEA/DSN/SESRS Responsable DSN/SESRS/FAR
Date de démarrage 01.01.76	Etat actuel en cours	Scientifiques
Date prévue d'achèvement 31.12.79	Dernière mise à jour 02.01.78	

1 - Objectif général :

En fonction des cas accidentels hypothétiques pouvant survenir sur un réacteur à eau pressurisée, ces études sont essentiellement basées sur le transfert de la contamination à l'intérieur de la centrale jusqu'au rejet dans le milieu naturel. Elles devront tenir compte des dispositions internes à la centrale permettant de réduire, d'une part l'ampleur de l'accident, et d'autre part ses effets.

2 - **Objectifs particuliers** :

1 - Elaboration d'un catalogue d'accidents types

2 - Mise au point d'un faisceau de données nécessaires au calcul des transferts et des conséquences radiologiques pour chaque accident type

3 - Etude des dispositifs de sauvegarde actuels et de leur aptitude à réduire les conséquences des accidents, propositions pour des dispositifs nouveaux, analyse des solutions étrangères

4 - Etude des mécanismes de transfert de la contamination à l'intérieur des différents milieux et de leur influence sur la composition, le niveau et la cinétique des rejets.

5 - Etude des modes de défaillance du confinement pour les accidents hors dimensionnement : ruptures hors-sol et traversée du radier par le corium

.../

3 - Installations expérimentales et programmes :

- piégeage des iodes dans les bétons
- programme expérimental BOUFFON pour la tenue du combustible, en particulier essai FLASH
- programme expérimental PHEBUS

4 - Etat de l'étude :

1) Avancement à ce jour :

Etude de quelques cas accidentels présentés dans le rapport WASH 1400 : séquences S_1 et S_3

Application du code ALICE pour des études de compréhension des cas précédents.

2) Résultats essentiels :

Etude préliminaire des différents cas de défaillance enceinte par rupture du radier.

Etude des problèmes de la contamination du sous-sol en cas de traversée du radier par le coeur.

5 - Prochaines étapes :

Etudes de différents cas accidentels (LOCA, petite brèche, transitoire).

Adaptation des études RASMUSSEN à la situation en France, compte tenu de la différence de conception des systèmes (en particulier des systèmes de sauvegarde) et du confinement.

6 - Relations avec d'autres études :

- Développement d'un code de calcul (ALICE) pour l'étude des conséquences radiologiques des accidents nucléaires.

7 - Documents de référence : rapports internes non disponibles

143-1 -03 /4165-40		5.3
Titre MIGRATION DE LA CONTAMINATION DANS LES PILES A EAU ORDINAIRE EN SITUATION ACCIDENTELLE		**Pays** FRANCE **Organisme directeur** CEA/DSN
Titre (anglais) MIGRATION OF THE CONTAMINATION IN PWR IN ACCIDENT CONDITION.		**Organisme exécuteur** CEA/DSN/SESTR/CAD. **Responsable** SESTR/CAD.
Date de démarrage 1/9/73	**Etat actuel** étude en cours	**Scientifiques**
Date prévue d'achèvement 31/12/81	**Dernière mise à jour** 15/11/77	

1 - OBJECTIF GENERAL -

Etude des conséquences des accidents de manutention de combustible dans les piles à eau.

Etude des rejets de contamination en cas d'accident de dépressurisation (participation au programme PHEBUS). Transfert des Iodes entre phase gazeuse et liquide dans es réacteurs PWR.

2 - OBJECTIFS PARTICULIERS

Expériences PIREE-MANUTENTION et REGARDE : du gaz sous pression est injecté au fond d'une cuve contenant 45 m^3 d'eau. On mesure l'efficacité de la barrière constituée par l'eau, et on suit l'évolution de la contamination dans l'atmosphère ventilée qui la surmonte.

Paricipation au programme PHEBUS : mesure de l'émission de radioactivité dans un accident de perte de réfrigérant. Mesures des produits de fission émis, mesure d'hydrogène.(Expérience NESSIE).

Etude des formes d'iode dans les phases liquides et gazeuses des circuits d'une pile à eau sous pression.

3 - INSTALLATIONS EXPERIMENTALES ET PROGRAMME -

Dispositif PIREE : cuve de 45 m^3. Injection sous eau et sous pression de Krypton, Xenon et Iode 131. Prélèvement, par filtres MAYPACK, d'eau et de gaz (Krypton et Hydrogène). Détermination de la forme chimique de l'Iode dans l'eau, influence sur le transfert dans les gaz.

4 - ETAT DE L'ETUDE -

1 - Avancement à ce jour -

1) EXPERIENCE PIREE MANUTENTION
- L'étude de la rétention de l'iode et du gaz dans l'eau dans le cas d'une injection est terminée.

2) PHEBUS
En cours d'achèvement.

3) NESSIE
Mise au point de la méthode électrochimique de caractérisation des formes d'iode.

2 - Résultats essentiels -

1) Dans le cas d'une injection, l'efficacité de décontamination de l'eau vis à vis de l'iode est de 500 ; vis à vis des gaz, elle varie avec les conditions opératoires (10 à 30 %), on observe un relâchement ultérieur dont la cinétique peut-être contrôlée.

2) Phébus : réalisation des appareils de prélèvement PF et H_2. (MAYPACK).

3) L'étude théorique complète des conditions thermodynamiques de l'existence des diverses formes d'iode est en voie d'achèvement.

5 - PROCHAINES ETAPES -

1 - Inluence du piégeage de l'Iode dans l'espace gaine combustible sur l'émission. Montage des appareils. Construction de l'installation (1978).

2 - Construction d'un banc d'essai et de contrôle des MAYPACK (1978).
   - Phébus.

3 - Mise au point de la méthode électrochimique de caractérisation des formes d'iode (1978) ; application à l'eau d'un circuit primaire du réacteur (1979).

6 - RELATION AVEC D'AUTRES ETUDES -

transfert de la contamination dans les piles en service.
contrôle des pièges à iode des centrales nucleaires.

7 - DOCUMENTS DE REFERENCE - Disponibles :

"Accident de manutention - expérience PIREE-MANUTENTION", E.DE MONTAIGNAC, L. ROUSSEAU, J. PORCHERON - Communication à la réunion des spécialistes sur la sécurité des éléments combustibles pour réacteurs à eau, Saclay 22-24/10/1973.

"Expérience PIREE-MANUTENTION", J. PORCHERON - Communication VII/34-01, Congrès de de la SFRP, Versailles 1974.

Title 1 (Original language) Tecniche per la misura della capacità di ritenzione di sistemi di filtrazione per iodio e derivati iodoorganici.	Classification 5.3
Title 2 (English) Techniques for Testing Charcoal Absorbers for Iodine and its Derivatives.	Country ITALY Sponsor ENEL Organisation Polytechnic Institute of Milan
Date initiated 1970 Date completed - Last updating April 1977	Project Leader G. Sandrelli

Description:

1. General Aim

Development of methods to test 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 Polytechnic Institute of Milan.

4. Project Status

The tests have been extended to low methyl iodide concentrations such as those expected in the annulus of a double containment system in the case of a LOCA.

5. Next Steps

The mathematical model will be adapted to the methyl iodide low concentrations.

Title 1 (Original language)	Classification
Trattamento dei gas nobili radioattivi prodotti per fissione.	5.3
Title 2 (English)	**Country** ITALY
Fission produced radioactive noble gases treatment.	Sponsor CNR - CNEN Organisation University of Pisa
Date initiated July, 1970 Date completed End of 1977 Last updating 1977	Project Leader CURZIO Giorgio

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1) General aim
Theoretical and experimental research on the general problems involved in the production, release and teatment of radioactive noble gases.

2) Particular objectives
a) Charcoal beds adsorption characteristics determination in ideal work conditions.
b) Evaluation of the dependence of characteristics on the bed size and grain size.
c) Evaluation of the effects of the decay heat, moisture, other impurities, pressure and temperature transients, etc.
d) Comparative analysis of treatment devices.
e) Charcoal filter tests in laboratory scale and full scale.
f) Granular charcoal characteristics determination.

3) Experimental facilities
G a) Charcoal bed testing facility
b) Nuclear detection devices
c) Granular charcoal testing facilities (to be completed)

4) Project status
Items 2 a, b are completed; 2 c, d, e, are near to be completed; 2 f are at the starting point.

5) 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.

TITLE 1 (original language)	Classification
Trattamento dei gas nobili radioattivi prodotti per fissione.	5.3

14. CASTELLANI F., CURZIO G., GENTILI A.
Analisi comparativa tra i sistemi di trattamento dei gas nobili radio attivi prodotti negli impianti nucleari. RP 280(77).

15. CASTELLANI F., CURZIO G., GENTILI A.
Metodologia di collaudo di un sistema di trattamento per il rilascio ritardato di gas nobili radioattivi prodotti negli impianti nucleari. RP 284(77).

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)

TITLE 1 (original language) Abbattimento di Iodio (attrezzatura PSICO 10)	Classification 5.3
TITLE 2 (english) Removal of Iodine from Containment Atmosphere by Sprays with PSICO 10 Facility	Country: ITALY Sponsor: CNR - CNEN Organisation: CAMEN - University of Pisa
Date initiated 1967 Date completed 1976 Last updating June 1976	Project Leader R. Mirandola (University) 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.

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.

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TITLE 1 (original language) Abbattimento di Iodio (attrezzatura PSICO 10)	Classification 5.3

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 Radioprotection 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). CAMEN report.

Title 1 (Original language)	Classification
Il controllo dei sistemi filtranti installati negli impianti nucleari per la rimozione delle particelle e dello iodio.	5.3

7. CUCCURU A., MAZZINI M.
Comportamento di filtri per particelle ad altissima efficienza in condizioni ambientali non usuali.
Istituto di Impianti Nucleari dell'Università di Pisa. RL 220(75).

8. Istituto di Impianti Nucleari dell'Università di Pisa: Relazione al CNEN sull'attività svolta nel campo della sicurezza nucleare:
- nell'anno 1970. RL 100(71) Tip. Edit. Pisana. Pisa, 1971
- nell'anno 1971. RL 119(72) Tip. Edit. Pisana. Pisa, 1972
- nell'anno 1972. RL 142(73) Tip. Edit. Pisana, Pisa, 1973
- nell'anno 1973. RL 172(74) Tip. Edit. Pisana, Pisa, 1974
- nel periodo luglio 74 ÷ giugno 75. RL 211(75) Tip. Edit. Pisana. Pisa, 1975.

9. CUCCURU A., KUNZ P., MAZZINI M.
Experiments on High Efficiency Aerosol Filtration.
Paper presented at the Seminar on High Efficiency Aerosol Filtration sponsored by CCE at Aix-en-Provence (F); 22-25 Novembre 1976.

10. MAZZINI M.
In Situ and in Laboratory Testing of HEPA Filters in Italy.
Paper presented at the Seminar on High Efficiency Aerosol Filtration sponsored by CCE at Aix-en-Provence (F); 22-25 Novembre 1976.

11. CUCCURU A., MAZZINI M., PRODI V.
Misure di efficienza di un sistema costituito da due filtri per particelle in serie.
Memoria presentata al Convegno dell'A.I.F.S.P.R., Pisa 28-29 Ottobre 1976.

8) Degree of availability

The previous references are free, the next ones may be available with the authorization of the CNEN.

Title 1 (Original language) Il controllo dei sistemi filtranti installati negli impianti nucleari per la rimozione delle particelle e dello iodio.	Classification 5.3
Title 2 (English) Testing of the Filter Systems used in Nuclear Plants for particle and iodine removal.	Country ITALY Sponsor CNEN and CNR Organisation University of Pisa - CAMEN
Date initiated: End of 1967 Date completed: End of 1978 Last updating: April 1977	Project Leader S. LANZA (University) M. MAZZINI (University) A. CUCCURU (CAMEN)

1) General Aim

To set up methods for testing the efficiency of HEPA and iodine filters, both in Laboratory and in situ, with reference to standard or accident conditions.

2) Particular objectives

a) Setting up and comparison of the methods, used for testing HEPA filters in laboratory and in situ.
b) Testing of HEPA filters under heavy environmental conditions (high temperature and relative humidity, overflow), with different particle size distribution.
c) Setting-up and comparison of the iodine and freon methods, used for testing charcoal filters in situ.
d) Testing the efficiency of materials used in nuclear plants for iodine removal in strictly controlled conditions of temperature, velocity and relative humidity of gas stream (standard and post-accident values).
e) Comparison of several types of Ag impregnated molecular sieves, with reference to nitric vapour poisoning.
f) Efficiency determination of two HEPA filters in cascade

3) Experimental facilities and program

- a rig and the related instrumentation for testing HEPA filters by NaCl, DOP, uranine and condensation nuclei methods.
- a rig and the related instrumentation for testing commercial charcoal filters by Freon and iodine methods.
- an apparatus to perform tests, indicated at the point 2.d) and 2.e) above.

Title 1 (Original language)	Classification
Il controllo dei sistemi filtranti installati negli impianti nucleari per la rimozione delle particelle e dello iodio.	5.3

4) Project status

The tests related to the points 2.a) and 2.e) are completed.
The first phase of the program related to the points 2.b), 2.c), 2.d) and 2.f) is completed. New more practical devices for in situ testing of charcoal filters with methyl iodine and Freon 112 methods were made.

5) Next steps

To run a set of experiments to compare methyl iodide and Freon 112 methods for testing commercial charcoal filters in laboratory and in situ.

6) Relation to other projects

"Removal of iodine" and "Fission Produced radioactive noble gases treatment".

7) Reference documents

1. LANZA S., MAZZINI M., MUZZI F., PERINETTI
   Il controllo dei sistemi filtranti installati negli impianti nucleari.
   Atti del Convegno sulle attività di ricerca nel campo della sicurezza degli Impianti Nucleari. CNEN-Roma, 1971.
2. MAZZINI M.
   Il collaudo "in situ" dei sistemi filtranti per particelle installati negli impianti nucleari.
   Normazione Nucleare (supplemento del Notiziario del CNEN); anno VI n. 2, 1972.
3. LANZA S., MAZZINI M.
   Influenza dell'avvelenamento con vapori nitrici sulla ritenzione di CH_3I da parte di zeoliti argentate.
   Istituto di Impianti Nucleari dell'Università di Pisa, RL 111(72).
   Tipografia Editrice Pisana. Pisa, 1972.
4. LANZA S., MAZZINI M.
   Il controllo dei filtri per iodio, installati negli impianti nucleari.
   Notiziario del CNEN anno 19 n. 6, 1973.
5. LANZA S., MAZZINI M. et al.
   Testing Methods for Iodine Filters of Nuclear Plants.
   Paper presented at the Seminar on Iodine Filter Testing sponsored by CCE at Karlsruhe (RTF), 4-6 december 1973.
6. MAZZINI M.
   Sull'influenza della granulometria e della natura dell'aerosol sulla efficienza di filtri FAEP.
   Giornale di Fisica Sanitaria e Protezione contro le radiazioni.
   Tip. Ed. Minerva Medica, Vol. 18, N. 4, Pag. 143-147 (ottobre-dicembre 1974).

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Title 1 (Original language)	Classification
Il controllo dei sistemi filtranti installati negli impianti nucleari per la rimozione delle particelle e dello iodio.	5.3

7. CUCCURU A., MAZZINI M.
Comportamento di filtri per particelle ad altissima efficienza in condizioni ambientali non usuali.
Istituto di Impianti Nucleari dell'Università di Pisa. RL 220(75).

8. Istituto di Impianti Nucleari dell'Università di Pisa: Relazione al CNEN sull'attività svolta nel campo della sicurezza nucleare:
- nell'anno 1970. RL 100(71) Tip. Edit. Pisana. Pisa, 1971
- nell'anno 1971. RL 119(72) Tip. Edit. Pisana. Pisa, 1972
- nell'anno 1972. RL 142(73) Tip. Edit. Pisana, Pisa, 1973
- nell'anno 1973. RL 172(74) Tip. Edit. Pisana, Pisa, 1974
- nel periodo luglio 74 ÷ giugno 75. RL 211(75) Tip. Edit. Pisana. Pisa, 1975.

9. CUCCURU A., KUNZ P., MAZZINI M.
Experiments on High Efficiency Aerosol Filtration.
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10. MAZZINI M.
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Paper presented at the Seminar on High Efficiency Aerosol Filtration sponsored by CCE at Aix-en-Provence (F); 22-25 Novembre 1976.

11. CUCCURU A., MAZZINI M., PRODI V.
Misure di efficienza di un sistema costituito da due filtri per particelle in serie.
Memoria presentata al Convegno dell'A.I.F.S.P.R., Pisa 28-29 Ottobre 1976.

) Degree of availability

The previous references are free, the next ones may be available with the authorization of the CNEN.

Classification

5.3.

Title 1	COUNTRY UNITED KINGDOM
GAS PHASE TRAPPING STUDIES (1)	SPONSOR UKAEA
	ORGANIZATION RDL WINDSCALE
Title 2	Project Leader J J HILLARY
Initiated 1973 Completed :	Scientists:
Status : Last updating	

Description:

1. 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 UNITED KINGDOM SPONSOR UKAEA ORGANIZATION RDL WINDSCALE
Title 2	Project Leader J J HILLARY
Initiated 1972 Completed : Status : Last updating	Scientists:

Description:

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 UNITED KINGDOM SPONSOR UKAEA ORGANIZATION RDL WINDSCALE
Title 2	Project Leader J J HILLARY
Initiated 1972 Completed : Status : Last updating	Scientists:

Description:

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 UNITED KINGDOM SPONSOR UKAEA ORGANIZATION RDL WINDSCALE
Title 2	Project Leader J J HILLARY
Initiated 1973 Completed : Status : Last updating	Scientists:

Description:

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 very 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 efficiency of trapping plant may be of interest down to 10^{-11} ci/m^3. These are very much lower than the levels of 10^{-5}kg/m^3 used in trapping experiments.

Preliminary work is aimed at the development of the measurement technique.

Reference Documents

Internal documents.

Classification: 5.4

Title: Konsekvenser af frigørelser af fissions produkter til atmosfæren	Country: DENMARK
	Sponsor: Risø National Lab.
Title: Consequences of Releases of Fission Products to the Atmosphere	Organization: Risø National Laboratory
Initiated date: 1972 Completed date: Status: Progressing	Scientists: O. Walmod-Larsen S. Thykier-Nielsen P. Hedemann Jensen

1. General aim

Calculation of consequences of releases of fission products to the atmosphere under various environmental conditions.

2. Particular objectives

Development of models for calculation of

a. Doses to individuals:
External gamma doses from airborne radioactive material.
Internal doses due to inhalation of radioactive material.
External gamma doses from radioactive material deposited on the ground.
Beta doses to the skin from airborne radioactive material and from radioactive material deposited on the ground.

b. Concequences of doses based on given dose-consequence relations.

c. Doses to individuals and population under specified meteorological conditions.

d. Probability distribution of doses to individuals and population for a given probability distribution of meteorological parameters.

e. Isodose curves: Shape, area and number of people receiving doses witin specific limits.

f. Number of concequences (i.e. number of people having e.g. early illness, cancer) for a given release.

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Both normal and accidential releases are considered.
Furthermore the parameters in the models are studied:
Duration of release, atmospheric stability, plume rise, ground roughness etc.

3. Experimental facility and programme

None.

4. Project status

A computer model, PLUKON1, based on the gaussian plume model and fulfilling the objectives a. - f. mentioned in section 2, has been developed. PLUKON1 can be used for calculation of doses and consequences in near-zone, i, e. the area witin 50 km from the release point.
A limited comparison between PLUKON1 and the models used in the other Nordic countries, Finland, Norway and Sweden has been made. On the basis of the calculation results from the models, it was concluded that there are no essential differences between the Nordic dose models. The results of the comparison is published in reference 1.
PLUKON1 has been used for a calculation of doses from hypotetical accidents at a nuclear power plant [2].

5. Next steps

Development of a model, PUFKON, for calculation of consequences of accidental releases in situations where the meteorological conditions varies with time. This model is based on a puff-model. The meteorological part of PUFKON i.e. the conversion of meteorological data to dispension parameters for puffs is finished.

6. References

1. Comparison of Nordic Dose Models, S. Thykier-Nielsen, Risø-M-1972.

2. Calculation of the Individual and Population Doses on Danish Territory Resulting from Hypothetical Core-melt Accidents at the Barsebäck Reactor, P. Hedemann Jensen, E. Lundtang Petersen, S. Thykier-Nielsen and F. Heikel Vinther, Risø Report No. 356.

3. Modeller til beregning af eksterne gammadoser og inhalationsdoser fra frigørelser til atmosfæren af radioaktive stoffer, S. Thykier-Nielsen, Risø-M-1725.
   (Description of the models for calculation of external gammadoses and inhalation doses).

4. Sammenligning af matematiske modeller til beregning af eksterne gammadoser hidrørende fra radioaktivitetsfrigørelser til atmosfæren, P. Hedemann Jensen, Risø-M-1726.
   (Comparison of different models for calculation of external gammadoses from a plume).

7. Degree of availability

Available on an exchange basis.
The computer programmes are written in Burroughs ALGOL for a B6700-computer.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 77 - 31. 12. 77	5.4	RS 239
Vorhaben/Project Title Aktivitätsüberwachung der Dosisbelastung in der Umgebung von Kernenergieanlagen Dose Rate Proportional Measurement of the Radioactive Gaseous Effluents from Nuclear Power Plants		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 131, Frankfurt
Arbeitsbeginn/Initiated 1. 10. 76	Arbeitsende/Completed 31. 12. 78	Leiter des Vorhabens/Project Leader Dr. Grosse-Schulte
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 77	Bewilligte Mittel/Funds 419.880,-- DM

1. General Aim

In order to facilitate more precise evaluations of the environmental dose rate in the vicinity of an nuclear facility, the methods of measuring radioactive noble gas discharge rates must be improved: A system capable of automatically analysing the type of nuclide and the quantities released will be coupled to a system yielding the meteorological distribution factors. These two sets of data will be used for the on-line calculation of the actual environmental burden. The on-line measurement and analysis of the airborne particulate will be included at a later point of time.

2. Particular Objectives

- improvement of on-line nuclide-identification systems for stack surveillance (noble gases, aerosols)

- development of computer programs for real-time (or at least at short intervalls) calculation of the actual environmental burden.

- development and improvement of the on-line calculation of distribution parameters from metereological factors (wind characteristics, temp. profile etc.)

- experimental verification of the means and methods conceived (prototype system at a nuclear power plant).

3. Research Program

3.1 Nuclide identification (& release rate measurement) programs - establish a test routines

3.2 Meteorological Data

3.3 Establish programs for the computation of environmental dose rates on the basis of data retrieved from the nuclide identification monitor and from the meteorological instrumentation.

3.4 Investigate the boundary conditions given by licensing requirements and plant conditions

3.5 Investigate the possibility of interconnecting the effluent monitors from several plants at one site

3.6 Test the programs and systems by means of setting up one prototype arrangement at a plant

3.7 Recalibrate the prototype system by means of test measurements (actual or mock releases from the stack).

4. Test Facilities

For Biblis A , a measurement device has been provided for specific nuclide-related monitoring or radioactive noble gases. The apparatus has been tested in Gundremmingen for approx. 4 months and has been operating in Biblis since Jan. 1976.

5. Progress to Date

According to licensing requirements on-line (or semi-automatic) identification of the noble gas nuclides released at the stack has to be performed in the F.R.G. First attempts to establish equipment of such kind have been made at the BIBLIS-A, and at the ISAR power plant, respectively.

Several discussions have resulted in the concept of a aerosol-identification monitor (step-mode air particulate monitor with a GeLi-Detector system).

The computational aspects of the programs have been covered by talks with experts in meteorological calculations and computer specialists. Storage capacity and the amount of programming still necessary have been determined.

6. Results

A prototype system for an air particulate monitor has been designed in cooperation with GRS, BGA (Federal Health Agency) and a manufacturer. The system is being manufactured at the moment.

The basic equipment has shown its reliability (technically proven step-mode device). The filter will accumulate aerosols in steps (approx. 30 min up to 2 hours) and the aerosol activity will be measured directly. Measurement will be performed by a Ge-Li-detector with high resolution and 5 % efficiency.

The scanning pattern for the calculation of the environmental burden was determined. In this respect computer storage demand has been clarified. Geometrically the "grid" will extend as far as 15 kilometres (approx. 10 mi) from the stack.

7. Next Steps

Specific nuclide-related noble gas measurement

Investigations will be made on the feasibility of a dose-rate proportional measuring device (gamma dose rate).

This instrument is to provide a signal directly proportional to the gamma dose rate in the environment (mR/h) and it is considered a back-up system to the mechanically sensitive and more sophisticated nuclide identification system. A concept considering radioecological factors will be discussed. It is based on the assumption that for various nuclides not only the integral Gamma dose rate but also the secondary and tertiary doses are worth considering (inhalation/ ingestion etc.).

Specific nuclide-related aerosol measurement

Manufacturing is almost finished, the first set of experiments is being discussed.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.77 - 31.12.77	5.4	PNS 4811 (4132)

Vorhaben/Project Title	
Investigation of the Physical and Chemical Environmental Behavior of Radionuclides Characterized by a Particular Biological Effectiveness Untersuchung des physikalischen und chemischen Verhaltens biologisch besonders wirksamer Radionuklide in der Umwelt	Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe Projekt Nukleare Sicherheit ASS

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.74	31.12.78	H. Schüttelkopf
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1977	

1. General Aim

Determination of the longterm exposure of the environment of the reprocessing plant by longlived ^{129}I.

2. Particular Objectives

The behavior of ^{129}I in the environment of the Karlsruhe reprocessing plant.

3. Research Programm

3.1 Development of analytical methods for the determination of ^{129}I.

3.2 Measurement of the concentration distribution of ^{129}I in the Karlsruhe reprocessing plant and of the ^{129}I release from the plant.

3.3 Measurement of the ^{129}I in the environment in the Karlsruhe reprocessing plant.

3.4 Measurement of the stable iodine in the environment.

4. Experimental Facilities, Computer Programs

The measurement of ^{129}I and ^{127}I is performed using neutronactivation; for this purpose the irradiation facilities of the Karlsruhe research reactor FR2 are applied.

5. Progress to date

In 1977 milk, soil, thyroid and air samples were analyzed which had been taken in the environment of WAK. Samples from WAK process solutions, air and effluent waters were checked for ^{129}I. The composition of organic, inorganic and aerosol ^{129}I in the WAK effluent air was determined. ^{127}I concentrations in the environmental air were measured in Kiel, Stade, Karlsruhe, Grundremmingen and Munich.

6. Results

Since the installation in 1975 of the iodine filter in the dissolver exhaust air system of WAK the ^{129}I emission of WAK has been maintained at a relatively constant level of 2 to 3 mCi/a. The ^{129}I concentration in goat milk samples has decreased since that time with a half-life of o.5 year. The ^{129}I concentrations in soil samples range from o.o5 to 40 fCi/g. The maximum values are found near the principal point of exposure to the WAK exhaust air plume. The concentration of ^{129}I in the upper 5 cm of soil is generally higher than in the underlying 15 cm.

The chemical composition of ^{129}I in the WAK exhaust air was determined during three months of sampling. The organic gaseous portion attains about 2o%, the inorganic gaseous portion being 80%. The concentration of the iodine aerosols amounts to 1% of the total iodine contained in the effluent air. The ^{127}I measurement - performed separately for gaseous organic iodine and aerosol iodine-was carried out in the environment of KfK from October 1976 until December 1977, using a method developed and tested in 1976. Similar sampling devices were installed in Kiel, Stade, Grundremmingen and Munich in 1977. The results of these measurements have shown that the inorganic gaseous ^{127}I content in the environmental air ranges from 5 to 1o ng/m^3 for the total area of the Federal Republic. The concentration of ^{127}I as an aerosol reaches 1 ng/m^3.

7. Next steps

In 1978 the vertical diffusion of ^{129}I and the contamination of soil samples taken in 1977 and 1978 will be measured. Up to the end of 1978 samples of goat milk will be analyzed for ^{129}I. The research program on ^{129}I will be terminated in 1978.

8. Relation with other projects

None

9. References

Proceedings of the seminar on "Radioactive effluents from nuclear fuel reprocessing plants", Karlsruhe, 22. - 24.11.1977, to be published.

1o. Degree of Availability of the Reports

Unrestricted distribution.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1977	5.4	PNS 4320
Vorhaben/Project Title Theoretical and Experimental Investigation of the Atmospheric Dispersion of Radioactive Gases and Aerosols Theoretische und experimentelle Untersuchung der Ausbreitung radioaktiver Gase und Aerosole		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum GmbH Projekt Nukleare Sicherheit ASS
Arbeitsbeginn/Initiated 1969	Arbeitsende/Completed 1978	Leiter des Vorhabens/Project Leader W.Hübschmann,H.Schüttelkopf
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1977	Bewilligte Mittel/Funds

1. General Aim

Improvement of the knowledge about the atmospheric dispersion of radioactive emissions in the micro-scale (up to 15 km distance) and in the meso-scale (> 15 km).

2. Particular Objectives

Short range atmospheric diffusion, long range atmospheric transport and diffusion, diffusion models for accidental releases, influence of surface roughness,meteorological information system.

3. Research Program

3.1 Tracer diffusion experiments at the different stability categories; chemical tracer gases are emitted at heights between 50 and 200 m.

3.2 Tetroons are started and tracked by radar.

3.3 Turbulence studies over areas of different roughness.

4. Experimental Facilities

4.1 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.

Sampling stations are operated automatically to collect air-tracer samples. Tracer concentrations are measured by a gas-chromatograph.

4.2 Tetroons: Helium-filled balloons of constant volume.

4.3 Field measurements are performed over areas of different surface structure, using a 15 m mast.

5. Progress to Date

5.1 The meteorological date measurement at the tower, the data registration and record have been continued during the report period. A small field measurement station has been installed. 11 diffusion experiments have been performed in 1977. The chemical tracers have been emitted at heights from 60 m up to 190 m. In most experiments two trancers have been released simultaneously at two levels. Six experiments have been performed at night during stable stratification.

5.2 To investigate the transport and diffusion in the meso-scale (> 15 km distance) tetroons are started and tracked by radar. Several tetroon flights have been performed at Philippsburg (Rhine Valley) and at Bremen. These flights showed the limitations of radar tracking without transponder. It is planned, therefore, to equip the tetroons by a transponder (light-weight receiver-sender unit), which is developed now, to make it compatible with tetroon requirements.

5.3 A small data-logging station is operated in an area of small surface roughness (grade II). The collected data on turbulence parameters are compared to those generated in the KfK (roughness grade III).

6. Results

On the basis of the tracer experiments a preliminary curve family of the diffusion parameters σ_y and σ_z for 60 m emission height has been compiled. Further experiments are necessary to confirm this curve family.

To confirm the σ_y, σ_z-curve family for 100 m emission height some supportive work has been done. The influence of the evaluation method, of the sampling time and of the surface roughness have been investigated. The parameters based on experiments of the KfK and of the KFA Jülich have been combined to a common σ_y, σ_z-curve family.

7. Next Steps

7.1 The tracer experiments will be continued and concentrated on emission heights above 100 m.

7.2 Tetroon flights will be performed in the Rhine Valley after the development of a lightweight transponder has been successful.

7.3 Field measurements are continued in 1978.

8. Relation with Other Projects

9. References

P. Thomas, W. Hübschmann, L. A. König, H. Schüttelkopf, S. Vogt, M. Winter; Experimental Determination of the Atmospheric Dispersion Parameters over Rough Terrain, Part I., KFK 2285, July 1976

P. Thomas, K. Nester;
Experimental Determination of the Atmospheric Dispersion Parameters over Rough Terrain, Part II., KFK 2286, June 1976

K. Nester, W. Hübschmann, P. Thomas;
The Influence of Ground Roughness on Atmospheric Diffusion,
The 4th International Clean Air Congress, Tokio 1977

J. Hiller, W. Hübschmann, K. Nester, H. Schüttelkopf, P. Thomas; S. Vogt;
in: Jahresbericht 1976 der Abteilung Strahlenschutz und Sicherheit, KFK 2433 (1977), pp. 108

H. Dilger, J. Hiller, W. Hübschmann, K. Nester, H. Schüttelkopf, P. Thomas, S. Vogt; im 2. Halbjahresbericht 1976 des Projektes Nukleare Sicherheit, KFK 2345 (1977), pp. 456

H. Dilger, W. Hübschmann, D. Nagel, K. Nester, P. Thomas;
Die Parameter der atmosphärischen Ausbreitung in der Umgebung des Kernforschungszentrums Karlsruhe, KFK 2499 (1977)

10. Availability of the Reports

Reports are available through Kernforschungszentrum Karlsruhe GmbH, Karlsruhe Zentralbücherei

120-1 -01/4170-10 220-1 -01/4170-10		5.4
Titre Etude des caractéristiques démographiques des sites sous l'angle de la sûreté. Etablissement de critères comparatifs de classement et de sélection des sites.		Pays FRANCE Organisme directeur CEA/DSN
Titre (anglais) Demographic studies of sites characteristics with respect of safety.		Organisme exécuteur CEA/DSN/SESRS Responsable SESRS/FONTENAY
Date de démarrage 1/01/74	Etat actuel En cours	Scientifiques
Date prévue d'achèvement 31/12/82	Dernière mise à jour 12/77	

1 - OBJECTIF GENERAL

Acquisition de données concernant la population, l'urbanisme et le développement économique autour des sites nucléaires.
Détermination et utilisation de critères quantitatifs permettant de prendre en compte ces caractéristiques pour l'examen des dossiers de sûreté des installations nucléaires.

2 - OBJECTIFS PARTICULIERS

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tablissement et mise à jour permanente d'un fichier informatique susceptible de f nir à tout moment l'implantation sous une forme normalisée des populations sédentaires, temporaires et prévisionnelles autour de tout site métropolitain.

2.2
Etablissement et mise à jour permanente de dossiers d'information concernant l'urbanisme et le développement économique autour des sites nucléaires.

2.3
Etablissement et utilisation de critères permettant un jugement rapide et objectif des caractéristiques démographiques d'un site en liaison avec ses autres caractéristiques.

3 - INSTALLATIONS EXPERIMENTALES ET PROGRAMMES - Néant

4 - ETAT DE L'ETUDE

4.1 - Avancement à ce jour

. La realisation d'un premier fichier-programme "DISPO" susceptible de donner la répartition des populations sédentaires autour d'un site quelconque est achevée.
. Le programme MERADE permet l'utilisation comparative de divers critères de pondération des données météorologiques, radiologiques et démographiques.

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4.2-Résultats essentiels :

. Le programme "DISPO" contribue à la réalisation des analyses de sûreté depuis Août 1976 et a servi à actualiser les données concernant la distribution de population autour de quelques sites CEA (LA HAGUE, BRUYERES LE CHATEL, CADARACHE, FONTENAY-AUX-ROSES, SACLAY).

. Le programme "MERADE" a permis d'effectuer la comparaison des principaux sites électro-nucléaires français en fonction de leurs caractéristiques démographiques et météorologiques.

5 - PROCHAINES ETAPES

- à court terme (1978)

. Acquisition des schémas Directeurs d'Aménagement et d'Urbanisme (S.D.A.U.) et des Plans Régionaux de Développement Economique (P.R.D.E.).

. Dénombrement plus précis de la population à proximité des sites nucléaires (dans un rayon de 5 km).

. Amélioration du fichier-programme DISPO en fonction de nouvelles données et réalisation d'un découpage plus précis pour les villes de plus de 100 000 habitants.

- à moyen terme (1978-1979)

. Utilisation comparative de divers critères pour le classement des sites nucléaires français.

. Amélioration du fichier-programme DISPO (régions frontalières).

- à long terme (1979)

. Réalisation de fichiers-programme pour les populations temporaires et prévisibles.

. Réalisation d'un critère prenant en compte les données socio-économiques.

6 - RELATION AVEC D'AUTRES ETUDES

. Identification et caractérisation des agressions externes liées aux activités humaines. Détermination de leur probabilité d'occurence.
Etude des transferts atmosphériques.
Evaluation des conséquences radiologiques pour la population en cas de rejet accidentel de produits radioactifs dans l'environnement d'une installation nucléaire.

7 - DOCUMENTS DE REFERENCE

1."Recherche d'un critère d'évaluation de site par la pondération des données radiologiques, météorologiques et démographiques", A.DOURY, R.GERARD - Rapport DSN R 82, Symposium AIEA Vienne, 9-13/11/1974 .

2 - Documents internes CEA .

8 - DEGRE DE DISPONIBILITE :

1 disponible .

123-1 -01/4171-10 223-1 -01		5.4
Titre Etude des transferts atmosphériques		Pays France Organisme directeur CEA/ DSN
Titre (anglais) Studies on atmospheric diffusion and transport.		Organisme exécuteur CEA/DSN/SESRS Responsable DSN/ SESRS/ FAR
Date de démarrage 01.01.72	Etat actuel En cours	Scientifiques
Date prévue d'achèvement 31.12.81	Dernière mise à jour 02.01.78	

1 - Objectif général :

Cette étude a pour objectif général l'élaboration de codes de calcul qualifiés afin de prévoir quantitativement (concentrations moyennes et maximales, concentrations intégrées dans le temps, etc. le transfert par l'atmosphère vers l'environnement d'un polluant minoritaire résultant de rejets permanents ou accidentels des installations industrielles, nucléaires notamment.

2 - Objectifs particuliers :

2.1. Objectifs théoriques

a) Développement et perfectionnement du modèle de transfert atmosphérique utilisé au DSN par la prise en compte de nouveaux éléments de la dispersion atmosphérique :

- variation temporelle du vent
- variation spatiale du vent
- surélévation des nuages ou panaches
- relief

b) Développement et perfectionnement des méthodes d'utilisatio des statistiques météorologiques afin d'affiner les prévisions effectuées à l'aide des codes de calcul par une meilleure prise en compte :

- des durées de rejets envisagés (Accidents)
- des spécificités du site considéré sur le plan de la dispersion atmosphérique
(en particulier, étude des persistances de situations météorologiques données pendant une durée déterminée).

c) Comparaison des résultats des applications du modèle DSN avec ceux que l'on obtient, dans les mêmes conditions, avec d'autres modèles français ou étrangers

2.2. Objectifs expérimentaux

a) Qualification du modèle DSN à partir d'expériences de simulations des transferts atmosphériques in situ (traceurs) ou sur maquette en veine hydraulique (en particulier, détermination des limites éventuelles de validité pour les paramètres de diffusion actuellement utilisés, à savoir les écarts-types σ_x, σ_y, σ_z de la SANDIA-CORPORATION déterminés en 1966.

b) Adaptation éventuelle du modèle DSN pour des situations météorologiques et/ou orographiques complexes. (vents faibles, brises cotières, terrain construit) afin d'étendre son domaine d'application (en particulier, ajustement éventuel des écarts types).

c) Développement et perfectionnement des deux techniques expérimentales.

3 - Installations expérimentales et programmes :

Les moyens expérimentaux comprennent :

- Une station météorologique complète située à PIERRELATTE et équipée d'un pylône de 100 m qui permet d'acquérir et de mettre en forme les données nécessaires aux études purement météorologiques (voir II.1.b). Cette station permet également d'assurer la couverture météorologique nécessaire lors de certaines campagnes de simulation in situ par traceurs.

- Deux mâts télescopiques de 30 m qui permettent de constituer deux stations météorologiques simplifiées mais déplaçables. Ces stations sont utilisées pour assurer la couverture météorologique des campagnes de simulation in situ et pour étudier les particularités météorologiques locales d'un site.

- Le matériel de prélèvement et d'analyse des traceurs (SF_6 et Fréon 13 B 1)

Ce matériel comprend :

- les dispositifs d'émission des traceurs à débit réglable
- les dispositifs de prélèvement d'échantillons gazeux.
- le matériel d'analyse des prélèvements (chromatographes en phase gazeuse et intégrateur)

- Une veine hydraulique installée à EVIAN et associée à une chaine de mesures des paramètres caractéristiques des phénomènes turbulents (profils de vitesse, intensités de turbulence, spectres énergétiques turbulents, hauteur de couche limite, hauteur de rugosité). Cette installation sert de support aux études de simulation de la dispersion atmosphérique sur maquette

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4 - Etat de l'étude :

4.1. Avancement à ce jour

- Objectif 2.1.a : Le problème de la variation temporelle du vent est résolu. Ce résultat a donné lieu à l'élaboration du code de calcul ICAIR
- Objectif 2.1.b : non avancée en 1977
- Objectif 2.1.c : non engagée
- Objectif 2.2.a :

  - les interprètations de toutes les expériences de simulation in situ par traceurs, effectuées depuis 1975, seront terminées au cours du premier trimestre 1978. Il s'agit des expériences suivantes :
    - "Longue et Moyenne distance" (Vallée du Rhône + IRAN)
    - "Terrain construit" (PIERRELATTE)
    - "Vents faibles" (CADARACHE)
  - une série d'expériences a été effectuée dans la veine hydraulique d'EVIAN afin de s'assurer qu'une simulation correcte de la turbulence atmosphérique pouvait être obtenue en veine hydraulique (conditions de similitude). Des limites à l'applicabilité de la méthode ont été dégagées.
  - une maquette du centre de PIERRELATTE a été réalisée. Une première expérience de simulation a été effectuée sur cette maquette dans les conditions de dispersion des expériences in situ "terrain construit" afin de comparer les résultats des deux méthodes (sur la maquette, les isoconcentrations au sol sont visualisées par l'application d'une méthode colorimétrique. Brevet Sécuripol)
  - une maquette du centre de CADARACHE a été construite afin d'effectuer une comparaison analogue à la précédente pour les expériences in situ "vents faibles".
- Objectif 2.2.b : non engagée
- Objectif 2.2.c :
  - un dispositif d'émission régulée des traceurs a été mis au point.
  - un appareil de prélèvement à 10 voies avec commutation automatique d'une voie à l'autre et durée des prélèvements réglable, est en cours de construction. (Prototype)

4.2. Résultats essentiels

- Code de calcul ICAIR

Ce code permet de tracer, à un instant donné et à un niveau donné au dessus du sol, les lignes isoconcentrations résultant du rejet, variable dans le temps, d'une ou plusieurs sources ponctuelles de hauteurs quelconques, pour des conditions de transfert atmosphérique uniformes dans l'espace mais variables en fonction du temps.

- Résulats expérimentaux directement utilisables pour réaliser les objectifs des sous-tâches 2.2.a et 2.2.b.

- Démonstration de la faisabilité d'une simulation des phénomènes atmosphériques en veine hydraulique

- Perfectionnement et augmentation des moyens expérimentaux

5 - Prochaines Etapes :

- Objectif 2.1.a : Prise en compte de la variation spatiale du vent - Adaptation du code ICAIR (1978)

  Prise en compte de la surélévation des nuages ou panaches (1979)

  Prise en compte du relief (1980)

- Objectif 2.2.b : Recherche d'une méthodologie pour déterminer, sur un site donné, des suites d'une durée donnée de séquences de situations météorologiques différentes en affectant à ces suites une probabilité d'occurrence globale (étude liée à celle des persistances de situations météorologiques données). (1980)

- Objectif 2.2.c : Comparaison du modèle DSN avec les modèles français ou étrangers pour le calcul des concentrations moyennes annuelles (rejets permanents) (1er trimestre 1978).

- Objectif 2.2.a : Interprétation complète des expériences in situ déjà effectuées. Détermination des limites d'application du modèle dans son état actuel (1er trimestre 1978).

  Réalisation d'une expérience de simulation in situ par traceurs en bordure de mer (brises cotières, 1978).

  Réalisation d'expériences de simulation en veine hydraulique sur la maquette de CADARACHE pour les conditions météorologiques obtenues dans les expériences in situ "vents faibles". Comparaison des résultats (1978).

- Objectif 2.2.b : Attente des résultats des prochaines étapes de la sous tâche 2.2.a.

- Objectif 2.2.c : Construction en série des appareils de prélèvements à 10 voies (1978).

6 - Relations avec d'autres études :

Elaboration de codes de calcul décrivant les transferts physiques dans l'air, l'eau et le sol (170-11-10/4170-10, édition Avril 1977) (Fiche close le 31.12.77, l'objectif transferts physiques dans l'air étant reporté sur la présente fiche).

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Développement du code de calcul ALICE pour l'étude des conséquences radiologiques d'accidents .

7 - Documents de référence :

. Documents internes non disponibles .

123-2 -01/4175-20 223-2 -01		5-4
Titre Etude des transferts océaniques et hydrologiques		Pays FRANCE Organisme directeur CEA/DSN
Titre (anglais) Hydrologycal and oceanic environment diffusion		Organisme exécuteur CEA/DSN/SESRS Responsable DSN/SESRS/FAR
Date de démarrage : 1/1/1977	Etat actuel : en cours	Scientifiques (SESRS)
Date prévue d'achèvement: 31/12/1981	Dernière mise à jour :02.01.1978	

1. Objectif général

L'étude des transferts océaniques et hydrologiques doit permettre d'effectuer des prévisions quantitatives des transferts de pollution, par l'eau des mers, des fleuves et des lacs, des effluents susceptibles d'être rejetés en permanence ou accidentellement par les installations.

2. Objectifs particuliers

a) Qualification des modèles mathématiques prévisionnels de transferts dans les liquides.

Le transfert du polluant dans l'eau fait intervenir divers phénomènes physiques que l'on peut classer en distinguant deux parties dans la zone polluée:

- le champ proche sous la domination des conditions de rejet ; le transfert dans cette zone est dû à l'entraînement dans le jet du fluide ambiant,

- le champ lointain dans lequel le transfert dépend presque exclusivement des propriétés du milieu naturel lui-même.

Ce classement fait apparaître deux types de modèles qui s'appuient sur une certaine schématisation des phénomènes:

- un modèle mathématique en champ proche qui prend en compte certaines conditions aux limites,

- un modèle mathématique en champ lointain qui utilise comme données d'entrée les résultats du modèle précédent.

b) Mise au point de méthode expérimentale et réalisation d'essais "in situ" et sur maquette pour caler et ajuster ces modèles.

4. Etat de l'étude

1. Avancement à ce jour.

a) Etude du modèle mathématique en champ lointain.

Le modèle à "bouffées" a été appliqué pour évaluer les transferts dans l'eau, dans le cas d'une émission ponctuelle instantanée (rejets accidentels) et d'une émission continue.

b) La définition et l'estimation d'une expérience "in situ" sont en cours en collaboration avec différents laboratoires et organismes.

2. Résultats essentiels.

- Synthèse bibliographique concernant les coefficients horizontaux de diffusion turbulente à introduire dans le modèle.
- Qualification du modèle dans le cas d'une émission ponctuelle instantanée; les résultats de calcul ont été comparés à des résultats expérimentaux obtenus dans la baie de Saint-Brieuc.
- Calcul des coefficients de transfert dans l'eau (grandeurs proportionnelles à l'irradiation externe) et application aux rejets du Centre de la Hague; une comparaison est faite avec les résultats de l'expérience "RHODOLEIA-BETA" effectuée en 1964.
- Mise au point du modèle dans le cas d'une émission continue en fleuve.

5. Prochaines étapes

- Qualification du modèle à "bouffées" dans le cas d'une émission continue.

  Ce modèle sera appliqué à l'évaluation des concentrations radioactives en utilisant les trajectoires des masses d'eau, à partir du point de rejet, calculées par le modèle de courants EDF/LNH.
- La définition et la réalisation d'expériences "in situ" et éventuellement sur maquette, en collaboration avec différents organismes.
- Etudes pour la mise en oeuvre d'un modèle mathématique en champ proche.

6. Relations avec d'autres études

A partir des résultats obtenus sur les transferts, des études complémentaires sur les phénomènes de reconcentration des radionucléides au niveau de divers constituants (sédiments, organismes) sont à développer.

C'est finalement la résultante de ces deux processus, dispersion et reconcentrati[on] qui est à considérer du point de vue des risques encourus.

7. Documents de référence - disponibles

Une méthode pratique pour la prévision numérique des pollutions océaniques.
A. DOURY, C. BADIE
Rapport CEA 4512 - 1973

123-3 -01/4173-10 223-3 -01		5.4
Titre Etude des transferts hydrogéologiques		Pays FRANCE Organisme directeur CEA/ DSN
Titre (anglais) Studies of hydrogeological dispersion		Organisme exécuteur CEA/DSN/SESRS Responsable DSN/SESRS/FAR
Date de démarrage : 01/01/1976	Etat actuel : en cours	Scientifiques
Date prévue d'achèvement: 31/12/1981	Dernière mise à jour : 02.01.1978	

1. Objectif général

L'étude des transferts hydrogéologiques doit permettre de fournir des éléments techniques pour la prévision quantitative des transferts de pollution sous l'angle de la sûreté des installations nucléaires: prévision des temps de transferts jusqu'aux exutoires naturels ou artificiels et des concentrations à attendre à ces mêmes exutoires.

2. Objectifs particuliers

2.1 - Etude d'un modèle mathématique tridimensionnel de transferts hydrogéologiques.

(Adaptation du modèle à bulle atmosphérique aux conditions particulières du milieu géologique).

2.2 - Mise au point de méthodes expérimentales permettant d'obtenir les données nécessaires à la vérification du modèle mathématique prévisionnel tridimensionnel.

2.3 - Recherche des paramètres et des coefficients d'une série de terrains "types" pour l'établissement d'abaques de coefficients de diffusion.

2.4 - Mise au point de sous-programmes particuliers pour la transformation des mesures de "terrain" brutes en données utilisables par les modèles (Forage-Concentration)

2.5 - Mise au point de méthodes permettant d'établir les lois d'adsorption dans un terrain in situ (double traçage).

3. Installations expérimentales et programme

Les installations expérimentales sont mises en oeuvre pour effectuer les études correspondantes aux sous-tâches 2.2, 2.3, 2.4, 2.5 .

Ces installations sont constituées par:

- Une maquette métrique (cuve cylindrique) installée dans les locaux du laboratoire de radiogéologie de l'Université Bordeaux I.

- Un terrain d'expérimentation au Barp qui est utilisé à la simulation des transferts hydrogéologiques, à l'échelle décamétrique en milieu sableux fin.

- Un terrain d'expérimentation dont le choix est à faire, qui sera utilisé à la simulation des transferts hydrogéologiques à l'échelle décamétrique en milieu alluvionnaire grossier.

4. Etat de l'étude

4.1. Avancement à ce jour

- La mise au point du modèle mathématique prévisionnel tridimensionnel est terminée (TRIDISOL).
- Les expérimentations sur la maquette métrique sont terminées. Les résultats sont en cours de dépouillement.
- Les expérimentations sur le site du Barp sont en cours, elles sont interrompues pour trois mois en raison de l'équipement du camion laboratoire. Les premiers résultats sont en cours de dépouillement. Notons à ce sujet que l'étude hydrogéologique préliminaire du site est terminée.
- La recherche d'un nouveau terrain d'expérimentation est en cours.

4.2. Résultats essentiels

Le code "Forage-Concentration" est opérationnel. Les premiers résultats des expérimentations sur maquette montrent que la valeur des coefficients de diffusion est indépendante du temps de transfert.

5 - Prochaines étapes

- Mise au point d'une méthodologie pour la détermination in situ des coefficients de diffusion macroscopique pour différents milieux en vue de l'établissement d'abaques des coefficients.

- Etalonnage des appareils de détection pour la mise au point de sous-programmes pour la transformation des mesures "brutes" en données utilisables pour les modèles.

- Mise au point d'une "sonde tritium" utilisable en forage, en vue d'utiliser la technique du double traçage.

6. Relations avec d'autres études

Expériences sur la sûreté des stockages des déchets.

7. Documents de référence

- Internes non disponibles .

Title 1 (Original language)	Classification
Valutazione quantitativa del rilascio di sostanze radioattive naturali nel-l'ambiente	$\boxed{5.4}$ - 5.5 - 5.6
Title 2 (English)	**Country** ITALY
Quantitative evaluation of the release of natural radioactive substances into the environment	Sponsor Organisation } CNEN
Date initiated: January 1974 Date completed: In progress Last updating: April 1977	Project Leader M. Dall'Aglio

Description

In the vicinity of some nuclear plants has been carried out the study of distribution and circulation of the natural isotopes of radioactive elements which can be released by the nuclear plants, before the start of the industrial activity (e.g. Impianto "Fabbricazioni Nucleari", Bosco Marengo, Alessandria).

After the beginning of the nuclear activity an environmental check can supply quantitative information about the pollution level due to the nuclear plants.

Title 1 (Original language)	Classification
Studi sulla contaminazione del mare	$\boxed{5.4}$ - 5.5 - 5.6
Title 2 (English)	**Country** ITALY
Studies on the contamination of the sea	Sponsor: CNEN and Euratom Organisation : CNEN
Date initiated 1957 Date completed in progress Last updating December 1976	Project Leader A. Brondi

Description

Studies of the factors which 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.

Title 1 (Original language)	Classification
Ricerche sui radionuclidi nell'ambiente	$\boxed{5.4}$ - 5.5 - 5.6
Title 2 (English) Researches on radionuclides in the environment	Country ITALY Sponsor Organisation CNEN
Date initiated January 1961 Date completed In progress Last updating April 1977	Project Leader Giorcelli

Description Systematic measurements of environmental radioactivity. The main purpose of such measurements is to keep under a constant surveillance the radioactive contamination levels in the environment. Furthermore the data collected are utilized for a study on the distribution and propagation of radionuclides in the environment.

Title 1 (Original language)	Classification
Ricerche su elementi stabili nell'ambiente	$\boxed{5.4}$ - 5.5 - 5.6
Title 2 (English)	**Country** ITALY
Researches on some stable elements in the environment	Sponsor / Organisation: CNEN
Date initiated: January 1969 Date completed: In progress Last updating: April 1977	Project Leader Giorcelli, Clemente

Description Researches on some trace elements (Hg, Se, Cr, Cs, Fe, Zn, Sb, Co, Rb, Ni, Ag, etc.) in areas of particular interest in order to determine natural and artificial concentration levels of these elements in the different links of the food-chain.

The data deriving from these measurements will provide an information of some value under the point of view of a correct evaluation of the risk to the population deriving from artificial release of these elements into the environment.

Title 1 (Original language)	Classification
Modello analitico per la valutazione quantitativa della dispersione atmosferica di inquinanti radioattivi gassosi	5.4 -- 5.6
Title 2 (English)	**Country** ITALY
A mathematical model for the evaluation of the atmospheric diffusion concerning the radioactive gaseous effluents	Sponsor Organisation: Universita' Palermo
Date initiated: February 1977 Date completed: in progress Last updating: May 1977	Project Leader E. Oliveri

Description

The aim of the research program is to develope a mathematical model for the evaluation of the atmosferic diffusion concerning the radioactive airborne material released from a Nuclear Plant. Standard meteorological dispersion models, including local parameters as the frequencies of Pasquill stability categories and the elements of climatological matrix,are used and the effects of the dry deposition and of the washout are taken into account.

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N.V. KEMA	CLASSIFICATION : 5.4
TITLE : Gevolgen voor de omgeving van ongevallen bij kernenergiecentrales	COUNTRY: THE NETHERLANDS
	SPONSOR : KEMA ORGANIZATION : KEMA
TITLE (ENGLISH LANGUAGE): Environmental effects of nuclear power plants accidents	PROJECTLEADER : B.Th. Eendebak
INITIATED : - / LAST UPDATING : 1978 STATUS : - / COMPLETED : 1977	SCIENTISTS : B.Th. Eendebak

General aim

To analyse the risks of light water reactors on specific sites in the Netherlands.

Particular objectives

To study the effects of nuclear accidents as a function of site, population density, wheather conditions, etc.

Experimental facilities

Not applicable.

Project status

Computer code "MAKRO" is available.

Next steps

Not applicable.

Relation to 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".

Reference documents

Not available yet.

Degree of availability

Through the organization KEMA.

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX

Classification

5.4, 5.6

Title 1 THE DEVELOPMENT OF A COMPUTER CODE TO CALCULATE THE CONSEQUENCES OF A RELEASE OF RADIOACTIVE MATERIAL TO THE ATMOSPHERE	COUNTRY UNITED KINGDOM
	SPONSOR UKAEA
	ORGANIZATION SRD CULCHETH
Title 2	Project Leader G D KAISER
Initiated 1974 Completed : 1976 Status : Last updating	Scientists:

Description:

1. General Aim

Theoretical work is required to calculate the doses to various organs received by people standing at several distances downwind of a given release of radioactive material and the area contaminated by fission products and actinides. It is also required to produce consequence probability curves where the consequences may be death, illness, or a contaminated area.

2. Particular Objectives

For a variety of weather conditions and wind velocities, the following are calculated:

a. doses to thyroid, lung, GI-Tract, and whole-body following inhalation
b. whole-body gamma dose due to external radiation
c. beta dose
d. areas dangerously contaminated by deposited fission products by 137 Cs alone and by actinides, as a function of elapsed time since the occurrence of the release.

By combining the results of the above calculation overall weather conditions, consequence/probability curves are plotted where the consequences can be a. thyroid cancer, b. deaths due to lung dose, c. deaths due to dose to GI-Tract, d. deaths due to whole-body dose, e. total man-rem, f. area dangerously contaminated and h. area dangerously contaminated by actinides, the last three consequences being calculated at various times after the release.

The model used distinguishes between short and prolonged release and also incorporates plume rise in a simple way. The effect of surface roughness on the atmospheric dispersion is also included.

3. Project Status

The program in its present state is able to produce all the results described in Section 2.

contd...

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Classification

Title 1	COUNTRY
	SPONSOR
	ORGANIZATION
Title 2	Project Leader
Initiated / Completed :	Scientists:
Status : / Last updating	

4. Next Steps

It is intended to increase the number of body organs included in the program by adding for example, bone. The area dangerously contaminated by radio-iodine will also be included.

In the longer term, it is hoped to include a more realistic plume-rise model and to deal with the important subject of inversion lids in a satisfactory way.

5. Relationship with Other Projects

The program is a flexible tool with which it is possible to calculate various consequences of a release of radioactive material to the atmosphere. It is therefore very useful in the context of nuclear safety studies. It can be used to study the consequences of reactor accidents on a specific site, or of accidents to a particular reactor on a variety of sites, or of accidents to a nuclear ship, or to a reprocessing plant, or the consequences of routine releases of radioactive material. It is therefore anticipated that the program will be used in a wide variety of studies.

149-2 - 03 245-1 - 01		5:5
Titre D O S I M E T R I E N E U T R O N I Q U E		Pays FRANCE
		Organisme directeur CEA /DSN
Titre (anglais) N E U T R O N D O S I M E T R Y		Organisme exécuteur CEA - DSN/SESTR CADARACHE
		Responsable SESTR - CAD.
te de marrage 1964	Etat actuel en cours	Scientifiques
Date prévue d'achèvement	Dernière mise à jour 15/11/77	

1 -Objectif général

Etude et développement de moyens de dosimétrie et de spectrométrie des neutrons pour les champs faibles (dosimétrie d'ambiance) et les champs forts (dosimétrie en cas d'accident nucléaire)

2 OBJECTIFS PARTICULIERS

Dosimétrie d'ambiance : les centrales PWR, beaucoup plus compactes que les centrales graphite-gaz, posent, dè ce fait des problèmes relativement aigus de fuite et de diffusion des neutrons. Il en est de même pour le transport et le traitement des combustibles qui traitent, maintenant, des produits fortement irradiés. La géométrie particulière de ce type de source accroit notablement l'importance des neutrons d'énergie comprise entre 100 keV et 1 MeV, neutrons pour lesquels le pouvoir de résolution des appareils développés à ce jour est insuffisant. Par ailleurs, les interventions se faisant, le plus souvent à des niveaux de rayonnement relativement élevés, un effort est à faire sur la maniabilité des moyens de mesure à mettre en oeuvre.

Dosimétrie criticité : des moyens de mesure convenables ont été mis en place dans les installations du CEA. Une action est à mener pour un emploi correct de ces matériels : étalonnages, entrainement du personnel. Des recommandations ont été rédigées au plan national et sont en cours de rédaction au plan international.

3 - Installations expérimentales et programmes :

Les installations disponibles pour les expérimentations sont : la station d'irradiation STIRCA, avec un accélérateur SAMES de 150 keV et un bloc de graphite équipé de 6 sources américium-béryllium, la source solution SILENE à Valduc. Divers réacteurs français ou étrangers ont également été utilisés : HPRR, à Oak Ridge, RB, à Vinca, VIPER, à Harwell, ainsi que les sources du PTB, à Brunswick.

Les objectifs définis plus haut fixent l'orientation des programme d'études qui portent sur le développement de nouveaux détecteurs et d'une électronique fonctionnelle.

Etude de détecteurs complémentaires : Les détecteurs actuellement à l'étude pour couvrir correctement la bande d'énergies 100 keV - 1 MeV soni

- la sphère de polyéthylène de 3" sous bore 10 avec détecteur ^{3}He,
- les compteurs proportionnels à protons de recul,
- les chambres à fission à uranium 235 et uranium 238 et, égalemei à neptunium 237.

Un ensemble de sphères de polyéthylène à détecteur activable - utilisant le dysprosium - est également mis en oeuvre.

Réalisation d'une électronique fonctionnelle : le matériel à utiliser en centrale doit être aussi portatif que possible afin de limiter la durée des interventions. On a donc réalisé préamplificateurs et échelles miniaturisés ainsi que des alimentations sur batterie

Amélioration de la dosimétrie criticité : il est prévu l'étalonna des moyens de mesure béta utilisés par les différents SPR pour la dosimétri criticité. Les sources étalon seraient fournies par le LMRI. Enfin, dans ce même domaine, on a entrepris la comparaison des programmes de calcul TRIPOLI et CARNAC qui permettent tous deux le calcul des doses délivrées en tous points d'un local par une source de composition et de géométrie définies.

4 - Etat de l'étude

1 - Avancement à ce jour : les matériels nécessaires ont été acquis et l'on a procédé à des étalonnages préliminaires avec des sources Am-Be. Pour les sphères à détecteurs de dysprosium, des mesures ont pu être faites auprès du Van de Graaff du PTB à Brunswick.

Des chambres à fission au neptunium 237, à l'uranium 238 et à l'uranium 235 ont été acquises. Des écrans de cadmium et de bore 10 ont été réalisés pour cette dernière.

Un effort important a été fait dans le sens de la maniabilité et l'on dispose maintenant, pour les sphères de Bonner comme pour les chambres à fission, d'alimentations autonomes et d'échelles de comptage de poids réduit.

Un prototype d'échelle miniature à calculateur incorporé, particulièrement adaptée au traitement des détecteurs activables, est en cours de développement en liaison avec le SEIn/Cadarache.

En vue de la métrologie de SILENE avec écran de plomb la mesure des détecteurs fissiles sur compteur Geiger AMPEREX a été mise au point et ces détecteurs ont pu être utilisés avec succès par Mr MEDIONI du DPR pour la 14ème intercomparaison auprès du HPRR d'Oak Ridge.

Le chapitre IV des recommandations AIEA concernant la dosimétrie criticité a été rédigé et adressé à l'Agence. Il traite des divers détecteurs utilisables pour la dosimétrie criticité et des méthodes de mesure correspondantes.

2 <u>Résultats essentiels</u> : On dispose maintenant de matrices des réponses correctement établies pour les sphères de polyéthylène à détecteurs Hélium 3 et Dysprosium. L'expérience a montré les limitations de ce système, surtout en ce qui concerne les neutrons d'énergie intermédiaire.

Pour les chambres à fission, utilisées nues ou sous écran de cadmium ou de bore 10, on ne dispose encore que de courbes de réponse calculées, recalées par la mesure de sources Américium-Béryllium ou Californium. Les mesures en neutrons monoénergétiques restent à faire.

Pour l'évaluation des doses après un accident de criticité, on teste les possibilités des calculs Monte-Carlo et CARNAC. Un rapport très détaillé a été fourni par le SERMA sur les résultats d'un calcul Monte-Carlo concernant SILENE. Ce travail met en évidence la très grande importance des neutrons réfléchis par le plancher de béton.

5 - <u>Prochaines étapes</u> : Les études à venir porteront sur les nouveaux détecteurs - chambres à fission et sphères dysprosium -, leur mise au point, la vérification des courbes de réponse et l'utilisation

pour la spectrométrie et la dosimétrie.

Par comparaison avec les résultats du calcul Monte-Carlo, le programme CARNAC sera adapté, ses données modifiées si nécessaire.

6 - Relation avec d'autres études

Les études concernant les mesures neutrons sont menées en liaison avec le DPr/STEP qui traite les problèmes de mesure gamma en présence de neutrons et de dosimétrie individuelle neutrons.

Les résultats des mesures effectuées en centrale intéressent au premier chef, le SERMA pour comparaison avec les données des calculs de protection. Ils concernent également le DRP/EDF pour la vérification des mesures qu'il effectue en divers points accessibles lorsque le réacteur est en fonctionnement.

Enfin, Merlin-Gerin semble s'intéresser à la mesure de l'irradiation au niveau des têtes de mécanismes de controle.

7 - Documents de référence

M. BRICKA, NGUYEN VAN DAT, L. PORTHEOS - Le spectromètre neutron à activation SNAC - Rapport CEA R 4226 - (1971),

R. CAIZERGUES, G. POULLOT - Calcul de la réponse des sphères de Bonner pour les détecteurs ILi, ^{3}He et Mn - Rapport CEA R 4400 - (1972

M. DOLIAS - Réponse des sphères modératrices à des neutrons monoénergétiques - Rapport EUR 4791f - (1972),

GROUPE DE TRAVAIL N° 9 DES SERVICES DE RADIOPROTECTION - Recommandations pour la dosimétrie en cas d'accident de criticité - Rapport CEA R 4669 - (1975),

M. BRICKA, R. MEDIONI, M. MOURGUES, S. LORRAIN - Neutron and gamma measurements near Viper Reactor - Rapport DPr/DSN - (May 1976)

M. BRICKA - La formulation linéaire en spectrométrie des neutrons - Méthode des spectres modèles - Rapport CEA R 4825 - (1977)

M. MOURGUES - Réponse des sphères modératrices à compteur ^{3}He. A paraitre.

8 - DIFFUSION SANS RESTRICTION -

Title 1 (Original language)	Classification
Ricerche su elementi stabili nell'ambiente	$\boxed{5.4}$ - 5.5 - 5.6
Title 2 (English)	**Country** ITALY
Researches on some stable elements in the environment	Sponsor Organisation: CNEN
Date initiated: January 1969 Date completed: In progress Last updating: April 1977	Project Leader Giorcelli, Clemente

Title 1 (Original language) Ricerche sui radionuclidi nell'ambiente	Classification $\boxed{5.4}$ - 5.5 - 5.6
Title 2 (English) Researches on radionuclides in the environment	Country: ITALY Sponsor / Organisation: CNEN
Date initiated: January 1961 Date completed: In progress Last updating: April 1977	Project Leader Giorcelli

Title 1 (Original language)	Classification
Studi sulla contaminazione del mare	$\boxed{5.4}$ - 5.5 - 5.6
Title 2 (English) Studies on the contamination of the sea	Country ITALY Sponsor: CNEN and Euratom Organisation : CNEN
Date initiated 1957 Date completed in progress Last updating December 1976	Project Leader A. Brondi

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Title 1 (Original language) Valutazione quantitativa del rilascio di sostanze radioattive naturali nel-l'ambiente	Classification 5.4 - 5.5 - 5.6
Title 2 (English) Quantitative evaluation of the release of natural radioactive substances into the environment	Country ITALY Sponsor / Organisation: CNEN
Date initiated: January 1974 Date completed: In progress Last updating: April 1977	Project Leader M. Dall'Aglio

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N.V. KEMA	CLASSIFICATION : 5.5
TITLE : Bepaling van het aantal lekke splijtstofstaven en de kernpositie tijdens het reactorbedrijf	COUNTRY: THE NETHERLANDS
TITLE (ENGLISH LANGUAGE): Determination of the number of leaking fuel rods on the core position during operation	SPONSOR : KEMA ORGANIZATION : KEMA PROJECTLEADER : J. Hoekstra
INITIATED : – / LAST UPDATING : 1978 STATUS : – / COMPLETED : 1977	SCIENTISTS : J. Hoekstra

General aim

To reduce the wet-sipping time.

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.

Experimental facilities

Dodewaard nuclear power plant.

Project status

Still in progress.

Next steps

Not applicable.

Relation to other projects

None.

Reference documents

None.

Degree of availability

Through the organization KEMA.

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX

148-2 -01 158-2 -01		5-6
Titre Développement d'un code de calcul (ALICE) pour l'étude des conséquences radiologiques des accidents nucléaires.		Pays France
		Organisme directeur CEA/ DSN
Titre (anglais) Development of a calculation code (ALICE) for study of radiological consequences of nuclear accidents.		Organisme exécuteur CEA/DSN/SESRS
		Responsable DSN/SESRS/FAR
Date de démarrage 01.09.75	Etat actuel en cours	Scientifiques
Date prévue d'achèvement 30.06.78	Dernière mise à jour 02.01.78	

1 - Objectif général :

Etude pour toute installation nucléaire, de la dispersion des produits radioactifs depuis le point initial de rejet jusqu'à l'environnement et calcul, à l'intérieur et à l'extérieur de l'installation, des activités, débits de dose et équivalents de dose en fonction du temps.

Ceci doit conduire à l'évaluation réaliste des conséquences radiologiques du fonctionnement normal et des divers accidents envisageables, compte tenu des caractéristiques de l'installation et du site.

2 - Objectifs particuliers

2.1. Modélisation des phénomènes influant sur le transfert des produits actifs à l'intérieur de l'installation (piégeage, filtration, aspersion, dépôts d'aérosols)

2.2. Modélisation du transfert atmosphérique avec prise en compte du dépôt sec, du lavage, de l'existence de couches d'inversion.

2.3. Modélisation des effets biologiques sur l'homme : irradiation externe par le panache et les dépôts, contamination interne par inhalation; doses individuelles et collectives.

.../

3 - Installations expérimentales et programme :

Essai de piégeage des iodes (pièges à charbon, eau, béton)

Evolution des aérosols dans une enceinte après un feu de sodium

Feu de sodium contaminé, rôle des aérosols dans le transfert de la contamination.

Test du modèle à bulles de transfert atmosphérique.

4 - Etat de l'étude :

4.1. Avancement à ce jour :

L'étude est divisée en 2 grandes parties : l'intérieur et l'extérieur de l'installation qui se raccordent par la fonction débit de rejet à la sortie de l'installation. La partie concernant l'extérieur et traitant un rejet complexe à débit est complètement opérationnelle et a fait l'objet de nombreux passages de cas tests entrainant des corrections mineures. La partie concernant l'intérieur de l'installation est en cours d'achèvement, les présentations des données et des résultats ont été définies et des calculs de cas concrets de complexité croissante sont actuellement analysés et vérifiés. Un fichier comprenant les principales caractéristiques physiques et radiologiques des isotopes a été établi.

4.2. Résultats essentiels :

- Calculs paramétriques mettant en évidence les effets d'appauvrissement par dépôt sec et lavage.
- Intégration des doses (doses collectives en homme-rem sur des régions définies par l'intersection d'un secteur et d'une couronne, compte tenu des distributions des vents, des types de temps (stabilité, précipitations, inversions).
- Calculs de conséquences d'accidents hors dimensionnement de réacteurs PWR ; application à la comparaison de sites en approche probabiliste, **et à l'établissement des plans d'urgen**

5 - Prochaines étapes :

- Fin de la programmation de la partie du code traitant l'intérieur de l'installation, vérification des résultats et des possibilités réelles du programme.

- Application aux transferts internes des produits actifs dans le cadre des études des accidents hors-dimensionnement pour les PWR et les neutrons rapides.

- Etudes paramétriques pour juger des avantages des différents types de confinement.

- Raccordement des 2 parties actuellement programmées par la mise au point du modèle à bulles performant et rapide.

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6 - Relations avec d'autres études :

- Feux de sodium : modélisation et codes de calcul du comportement des aérosols sodés, fiche 156-1-01

- Etudes des transferts atmosphériques, fiche 123-1-01

- Evaluation des conséquences radiologiques en cas de rejets accidentels dans l'environnement d'une installation nucléaire. Application du code ALICE, fiche 148-2-02

- Accidents de PWR, transfert de la radioactivité à l'intérieur de la centrale, rejets hors confinement, fiche 148-1-02

- Accidents sur les réacteurs à neutrons rapides. Conséquences radiologiques internes et évaluation des rejets sortant du confinement, fiche 158-1-01

- Etudes des transferts hydrogéologiques, fiche 123-3-01

7 - Documents de référence : rapports internes non disponibles

148-2 -02		5-6
Titre Evaluation des conséquences radiologiques en cas de rejets accidentels dans l'environnement d'une installation nucléaire. Application du code ALICE.		Pays France Organisme directeur CEA/DSN
Titre (anglais) Assessment of radiological consequences in case of accidental release in the environment of a nuclear plant.		Organisme exécuteur CEA/DSN/SESRS Responsable DSN/SESRS/FAR
Date de démarrage 01.01.76	Etat actuel en cours	Scientifiques
Date prévue d'achèvement 31.12.78	Dernière mise à jour 02.01.78	

1 - Objectif général :

Evaluation des conséquences radiologiques potentielles à la suite d'un accident grave sur une installation nucléaire, compte tenu des caractéristiques de l'installation et des caractéristiques météorologiques, hydrogéologiques et démographiques du site.

Cette évaluation sera faite essentiellement en terme de nombre de personnes dont l'irradiation dépasse des niveaux de dose pré-établis ou de nombre de personnes subissant un dommage précis Elle peut être présentée de façon déterministe ou de façon probabiliste selon les objectifs visés.

2 - Objectifs particuliers

1 - Contribution à la mise au point des plans d'intervention en cas d'accident nucléaire.

2 - Mise au point d'une méthode de comparaison des sites basée sur l'évaluation des conséquences radiologiques par les différentes voies de transfert dans l'environnement

3 - Propositions pour l'adaptation de l'installation au site choisi, par exemple : limitation du nombre de tranches, amélioration des dispositifs de sauvegarde, type de confinement.

4 - Propositions pour l'adaptation du site à l'installation, par exemple : proposition de plans de développement contrôlé des activités économiques et de l'urbanisation, amélioration des moyens d'alerte et de communication.

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5 - Mise au point des programmes de calcul utilisant toutes les caractéristiques des rejets et du site.

3 - Installations expérimentales et programmes :

- Installations expérimentales: dans le mesure où ils existent, mâts météorologiques disposés à proximité immédiate de l'installation.

4 - Etat de l'étude :

4.1 - Avancement à ce jour :

Une première étude de comparaison de sites de réacteurs PWR a été effectuée au cours du dernier trimestre 1976. Le terme source était l'accident type PWR2 du WASH 1400 (rapport Rasmussen), et l'application en avait été faite à 15 sites français et 4 sites américains.

Cette étude a été reprise sur de nouvelles bases, toujours pour les PWR :

- terme source : 2 séquences dominantes (THLB'S et ABε) en probabilité mises en évidence par le WASH 1400 et en tenant compte du type de confinement (enceinte simple pour les 900 MWe, double enceinte pour les 1300 MWe).
- doses calculées à l'organisme entier, par irradiation externe, 24 h après le déclenchement de l'accident; définition d'une dizaine de niveaux de dose.
- prise en compte de 8 types de temps différents et de la forme réelle du panache
- grille fine de la population autour de l'installation.
- 20 sites français, à différents stades des procédures administratives.

Le programme CRAPP, permettant de quantifier les conséquences radiologiques sur un site donné, a été mis au point.

4.2. - Résultats essentiels

Les 20 sites français, traités avec 2 accidents hors dimensionnement et dont les conséquences ont été traitées de façon probabiliste, peuvent se diviser en plusieurs groupes assez nettement différenciés, avec des différences sensibles entre les extrêmes.

Parallèlement à cette comparaison de sites les mêmes séquences accidentelles ont été étudiées en vue de l'établissement des plans d'intervention.

5 - Prochaines étapes :

- Calculs poursuivis avec d'autres séquences accidentelles et sommation pondérée (pour les calculs probabilistes appliqués à la comparaison des sites) des résultats obtenus.

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- Prise en compte des doses par inhalation à différents organes et introduction des notions de dommage à court terme et à long terme, ces derniers pouvant être prépondérants en nombre.

- Poursuite des contributions à la mise au point des plans d'intervention.

6 - Relation avec d'autres études

- Accidents PWR. Transfert de la radioactivité à l'intérieur de la centrale et rejets hors confinement, fiche 148-1-02

- Etude des transferts atmosphériques, fiche 123-1-01

- Etude des caractéristiques démographique des sites sous l'angle de la sûreté, fiche 120-1-01 .

7 - Documents de référence : rapports internes non disponibles .

Title 1 (Original language)	Classification
Modello analitico per la valutazione quantitativa della dispersione atmosferica di inquinanti radioattivi gassosi	$\boxed{5.4}$ — 5.6
Title 2 (English)	Country ITALY
A mathematical model for the evaluation of the atmospheric diffusion concerning the radioactive gaseous effluents	Sponsor / Organisation: Universita' Palermo
Date initiated: February 1977 Date completed: in progress Last updating: May 1977	Project Leader E. Oliveri

Classification

$\boxed{5.4}$ 5.6

Title 1 THE DEVELOPMENT OF A COMPUTER CODE TO CALCULATE THE CONSEQUENCES OF A RELEASE OF RADIOACTIVE MATERIAL TO THE ATMOSPHERE	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION SRD CULCHETH
Title 2	Project Leader G D KAISER
Initiated 1974 Completed : 1976 Status : Last updating	Scientists:

6. FAULTS AND ACCIDENT COMBINATIONS

Classification: 6

Title: Unormale hændelser på A-værker	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: Classification System for Nuclear Power Plant Incidents	Organization: Risø National Laboratory
Initiated date: 1977 Completed date: Status: In progress	Scientists: H. Larsen H.E. Kongsø

1. General Aim

Incidents on nuclear power plants are analysed on the basis of Nuclear Power Experience Documents (NPE). NPE compiles and reports on the operating experience of Light Water Reactors, with emphasis on operating problems. A classification system has been set up, comprising a total number of 18 classification-criteria like: Docket No., Time of Commissioning, Primary Component, Secondary Component, Primary Fault Machanism etc. Each incident is registered on punchcards. Work on a computer program to analyse the data has been initiated. The analysis comprises all reports concerning PWR's and BWR's in 1977.

2. Particular objectives

3. Experimental facilities and programme

4. Project status

5. Next steps

Further steps will be to perform an analysis of the data for 1977 be means of the computer program and to extend the time-interval covered.

6. Relation with other projects

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7. Reference documents

Annual Progress Report, Department of Reactor Technology, Risø 1977.

8. Degree of availability

Classification 6

Title Common cause failure	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
	Project leader J.R. Taylor
Initiated 1974	

General aim

The object of the project is to gather and classify data concerning common cause failure, and to develop models using the data to predict common cause failure probability.

Project status

So far, data concerning some 500 failure incidents have been studied in detail, and classified data for 121 coupled failures recorded. The project is continuing.

Reference documents

J.R. Taylor, Common Mode and Coupled Failure, Risø-M-1826, October 1975.

Availability

Reports are available on request.

Classification 6

Title	COUNTRY Denmark
Reliability of computer based control	SPONSOR Risø National Laboratory
	ORGANIZATION Risø National Laboratory
	Project leaders J.R. Taylor S. Bologna R. D'Agostino
Initiated 1973	

General aim

Research on the topic of computer reliability was originally taken up at two separate institutions - Risø and CNEN, Cassacia. On the specific topic of developing software for deriving systematic testing data for control programs, the two institutes joined forces. The joint project dates from 1977.

The present work is aimed at producing an interactive program, which will produce sets of data capable of testing all paths or all branches in a computer program.

Project status

Progress to date (April 1977) includes completion of the basic support routines and program language analyser for the program.

Relation with other projects

The project is part of a larger program of work at CNEN.

Reference documents

One report has been published.

J.R. Taylor, Proving Correctness of a Real Time Operating System, 3rd European Real Time Conference, Budapest 1973.

Availability

Copies of reports are available on request.

Classification 6 and 14

Title 1 Fejlanalyse af tekniske Systemer og vurdering af deres pålidelighed	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
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.

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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.

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TITLE 1 (original language) Fault tree analysis for nuclear power plants	Classification 6-14
TITLE 2 (english)	Country: ITALY Sponsor: CESNEF, Politecnico di Milano Organisation: " "
Date initiated 1973 Date completed 1975 Last updating	Project Leader S. GARRIBBA

nal 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.

TITLE 1 (original language)	Classification
Fault analysis of the conventional island in a LWR nuclear power plant.	6 - 14
TITLE 2 (english)	Country: ITALY Sponsor: Franco Tosi S.p.A. Organisation: " " "
Date initiated April 15, 1975 Date completed April 15, 1976 Last updating June 1976	Project Leader V. 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 a… 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 VI nuclear power stations.