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 1977

COMMISSION
OF THE
EUROPEAN COMMUNITIES

Directorate-General
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XII/D/3

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BT/uk

12485

Mr. KRAG
Délégation de la Commission des Communautés
européennes aux Etats-Unis
2100 M Street N.W. (suite 707)
WASHINGTON D.C. 20037

Dear Sir,

European Atomic Energy Community
Water Reactor Safety Research Projects
Document XII/740/77-E

Please find enclosed document XII/740/77-E, this being the current compilation of nuclear safety research projects within the European Atomic Energy Community. It replaces document III/578/76-E.

Yours sincerely,

f.o. W. VINCK

Encl.

COMMISSION OF THE EUROPEAN COMMUNITES
Directorate-General for Research, Science and Education
XII/D/3

NUCLEAR SCIENCE AND TECHNOLOGY

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VOLUME I

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XII/740/77-E
WG2-77/7

INTRODUCTION

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

The formats are assembled in the order of the revised classification system agreed at the fifth plenary meeting of Working Group N° 2 on Light Water Reactor Safety (Research) of the European Community, held in Brussels in January 1976. Research projects on pressure tube type reactors can now be included.

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) Up dated 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.

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

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Classification system

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

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

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

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

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

6. Faults and accident combinations

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

8. Instrumentation, control and computerized protection

9. Other safeguards

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

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

      idem 11.2

   11.4 Containment
      11.4.1 Concrete structures

      idem 11.2

      11.4.2 Steel structures
   11.5 Coolant channels

12. Quality assurance

12.1 Formulation of quality assurance system

12.2 Fabrication methods

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

12.4 Human factors

13. Systems optimisation, standardisation, new concepts

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

14. Probabilistic methods of safety analysis

15. Interrelation between reactor plant and operating personnel

15.1 Behaviour of personnel (under normal and accident conditions)

15.2 Training of personnel

16. Environmental protection

16.1 Preparation for emergencies

16.2 Emergency equipment

17. Nuclear accident recovery and decommissioning

17.1 Decontamination

17.2 Removal of accident consequences

17.3 Decommissioning

18. Fuel cycle

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

19. Economics of safety

1. BLOWDOWN AND ENERGENCY CORE COOLING

Classification 1

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

1. General aim

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

2. Particular objectives

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

3. Experimental facilities and programme

4. Project status

   1. Progress to date

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

   2. Essential results

5. Next steps

   Development of a flooding version of RHC.

6. Relation with other projects

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

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

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

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

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

7. Reference documents

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

8. Degree of availability

   Available on exchange basis.

Classification 1.

Title 1	COUNTRY Denmark SPONSOR Risø ORGANIZATION Risø
Title 2 NORCOOL. A Model for Analysis of a BWR under LOCA Conditions.	Project leader: Jens G. Munthe Andersen
Initiated: September 1976 Completed: Status: under development Last updating:	Scientists: P.S. Andersen, P. Hansen P. Astrup R. Holt N. Bech J. Miettinen J. Eriksson H.V. Larsen M. Eget

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-32, December 1976.

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

8. Degree of availability.

Titre	Pays :
Accident de perte de caloporteur dans les réacteurs à eau pressurisée : Codes 1ère génération	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais)	Organisme exécuteur : CEA/DSN - SETSSR
LOCA and ECCS studies on PWR : first generation codes	Responsable : M. GOMOLINSKI (SETSSR)
Date de démarrage : 01/01/71 Date prévue d'achèvement : 31/12/79 Etat actuel : étude en cours Dernière mise à jour : 25/11/76	Scientifiques : N. TELLIER

Objectif général :

En s'appuyant sur les expériences Françaises et étrangères, mise au point et validation d'ensembles de codes décrivant tout l'accident dans le but de les utiliser pour l'évaluation de sûreté des réacteurs de puissance.

Objectifs particuliers :

Mise au point et comparaison de codes dits de 1ère génération : DANAIDES - CERES - CORINTHE - RELAP 4.
Qualification des codes sur des expériences OMEGA et problèmes standard CSNI.
Préparation et interprétation d'expériences à caractère global.

Etat de l'étude :

1) Avancement à ce jour :

Codes de 1ère génération opérationnels mais susceptibles d'être modifiés en ce qui concerne les corrélations utilisées.
Test des codes sur problèmes standard 1,2 et 3 (DANAIDES).
Corrélation en cours sur les expériences OMEGA (RELAP 4).
Début des calculs préliminaires PHEBUS.

2) Résultats essentiels :

Introduction d'une méthode de résolution numérique implicite qui réduit le temps de calcul d'un facteur important. Détermination des paramètres essentiels de la phase dépressurisation de l'expérience PHEBUS.

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Prochaines étapes :

Problèmes standards 5 et 6. Calculs des premières dépressurisations OMEGA. Suite et fin des calculs préliminaires PHEBUS avec détermination de la grille d'essais. Premiers calculs préliminaires LOBI.

Relation avec d'autres études :

Expériences OMEGA, ERSEC, CANON, MOBY-DICK, EDGAR, PHEBUS. Code FLIRA 2 et interprétation ERSEC pour le renoyage.

Documents de référence :

- "Programmes de calcul pour l'étude de l'accident de perte de caloporteur des réacteurs PWR", M.GOMOLINSKI, D.MENESSIER, N.TELLIER - Rapport SETS 31.
- "DANAIDES - programme de calcul de transitoire de décompression", D.MENESSIER, A.FORGE, A.PEDEL, D.LACOTTE.
- "Calculs préliminaires PHEBUS - résolution de quelques problèmes liés à l'utilisation d'un code à capacités et fonctions (RELAP 4 ou DANAIDES)", R.POCHARD, H.TARTU - Note Technique SETSSR-93.
- "Calculs préliminaires PHEBUS - étude des problèmes liés aux échanges avec les structures", R.POCHARD, A.PEDEL - Note Technique SETSSR 114.

Titre	Pays : FRANCE
Modèles et codes de calcul de 2ème génération pour l'étude de l'accident de perte de caloporteur dans les réacteurs à eau pressurisée.	Organisme directeur : CEA/DSN -EdF/SEPTEN
Titre (anglais) Advanced codes for the study of LOCA in P.W.R.	Organisme exécuteur : CEA - EdF/SEPTEN
	Responsable : M. REOCREUX (DSN-SETSSR)
Date de démarrage : 1/1/77 Date prévue d'achèvement 31/12/81 Etat actuel : en cours Dernière mise à jour : 01/04/72	Scientifiques : M. SUREAU (EdF-SEPTEN) M. CHABRILLAC (CEA/DRE) M. COURTAUD (CEA/DTCE)

Objectif général :

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.

Objectifs particuliers :

Description des phénomènes physiques intervenant au cours des diverses phases de l'accident.
Validation de ces modèles physiques sur l'ensemble des expériences françaises (OMEGA, ERSEC, MOBY DICK,...)
Intégration des modèles physiques dans un code de type modulaire et devant effectuer le calcul LOCA-ECCS (dépressurisation + renoyage) en un temps de calcul ne dépassant pas 2 heures.

Installations expérimentales et programme :

MOBY DICK	Débit critique
OMEGA	Transfert de chaleur en dépressurisation
CANON	Dépressurisation d'une capacité
ERSEC	Transfert de chaleur au cours du renoyage
EPIS	Injection de sécurité
REBECA	Ecoulement de brouillard entre les casemates d'un confinement.
MARVIKEN CFT	Début critique par une large brèche

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<u>Etat de l'étude</u> :

1) Avancement à ce jour :

Code CLYSTERE avec une modélisation axialle prenant en compte les les déséquilibres thermodynamiques. Ce code comprend des modules partiellement validés ou en cours de validation :
SERINGUE (MOBY DICK)
POMPES (EVA EPOPEE)
PSCHITT (ERSEC)
Il prend en compte la topologie complète du réacteur.
Modèle FLIRA : Premier calcul de renoyage avec un modèle axial, début d'interprétation des essais ERSEC

2) Résultats essentiels :

Nécessité de prendre en compte le déséquilibre thermodynamique dans les calculs d'accident.
Nécessité de compléter cette approche par l'introduction du glissement.

<u>Prochaines étapes</u> :

- Etablissement d'une modélisation prenant en compte le glissement et les déséquilibres thermodynamiques.
- Ecriture d'un système informatique, ossature du code LOCA à structure modulaire permettant
a) des changements de modules
b) le couplage de modules de types différents dans l'espace et dans le temps (Modélisation ou/et méthode numérique différente)

Titre	Pays :
Développement de moyens de calcul pour les études de sûreté des réacteurs à eau : CODE POSEIDON	FRANCE
	Organisme directeur : CEA - EdF/SEPTEN
Titre (anglais)	Organisme exécuteur :
Development of computation means for water reactor safety studies : POSEIDON CODE	CEA/DRE - SERMA (Saclay)
	Responsable : A.KAVENOKY (SERMA)
Date de démarrage : 01/01/74 Date prévue d'achèvement :31/12/78 Etat actuel : en cours Dernière mise à jour : 21/01/77	Scientifiques : M.LEWI J.P. L'HERITEAU

Objectif général :

Développement d'un ensemble de codes dits de 2ème génération, permettant le calcul de toutes les phases de l'accident de perte de réfrigérant du circuit primaire des réacteurs pressurisés.

Objectifs particuliers :

Ecriture d'un système de codes (POSEIDON) incluant des modèles physiques validés sur des expériences.

Etat de l'étude :

1) Avancement à ce jour :

Le système de codes POSEIDON, en projet, regroupe un ensemble de codes réalisés au CEA ou à EDF et qui sont à des états d'avancement différents.

2) Résultats essentiels :

Etude de faisabilité achevée. 1ère version de FLIRA en fonctionnement (code de renoyage).

Prochaines étapes :

Réalisation d'une première version de POSEIDON. Mise en service de la partie dépressurisation.

Relation avec d'autres études :

Les études expérimentales : MOBY-DICK, ERSEC, OMEGA servent de support pour le développement des modèles.

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Documents de référence :

- "Development of Numerical Methods for Thermohydraulic Problems in Reactor Safety", M. CHABRILLAC, and al., CSNI Meeting, Toronto 1976.

- "Reflooding Calculation Model Following an Accidental Primary Fluid Loss", JP. L'HERITEAU, D. MENESSIER, European Two Phase Flow Meeting, Haifa 1975.

Titre Méthodes de mesure en double phase	Pays : FRANCE Organisme directeur : CEA
Titre (anglais) Development of two phase flow instrumentation	Organisme exécuteur : CEA/DTCE-STT (GRENOBLE) Responsable : M. COURTAUD (STT)
Date de démarrage : 1974 Date prévue d'achèvement : 31/12/79 Etat actuel : en cours Dernière mise à jour : 1/4/77	Scientifiques : F. FRANCK J.C. ROUSSEAU

Objectif général :

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

Objectifs particuliers :

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

Installations expérimentales et programme :

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

Etat de l'étude :

1) 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.

Prochaines étapes :

Etalonnage en régime transitoire.

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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 branche froide de la boucle d'essai PHEBUS.

Documents de référence :

-"Mesure du titre et du débit massique d'un mélange diphasique à l'aide d'un moulinet et d'un venturi". R.FRANCK - Note TT 534

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

-"Etude générale de l'instrumentation destinée à la boucle PHEBUS" - Note technique DSN/SES n° 12/77.

Titre	Pays :
Etude et réalisation d'une installation (PHEBUS) pour tester le comportement d'une grappe d'éléments combustibles PWR en cas de dépressurisation	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais)	Organisme exécuteur : CEA/DSN-SES
Study and construction of a nuclear facility for the testing of the behaviour of a PWR fuel under depressurization condition,	Responsable : R. DEL NEGRO (SES)
Date de démarrage : 01/06/71 Date prévue d'achèvement : 31/05/77 Etat actuel : en cours Dernière mise à jour : 18/11/76	Scientifiques : C. LAGRANGE, C. GOLINELLI, A. TATTEGRAIN

Objectif général :

La réalisation d'une installation d'expérimentation permettant de :

- Reproduire les conditions d'environnement d'un réacteur PWR pour une grappe 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.

Etat de l'étude :

1) Avancement à ce jour :

Etudes terminées.

Construction en cours d'achèvement.

Prochaines étapes :

Divergence du réacteur.

Dépressurisation sans combustible.

Dépressurisation avec combustible.

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).

Documents de référence :

Rapport préliminaire de sûreté.

Titre	Pays : FRANCE
Programme PHEBUS : Moyen de calcul et calculs préliminaires.	Organisme directeur : CEA/DSN
Titre (anglais) PHEBUS EXPERIMENT PRELIMINARY CALCULATIONS	Organisme exécuteur : CEA/DSN-SETSSR Responsable : M. GOMOLINSKI (SETSSR)
Date de démarrage : 1/06/76 Date prévue d'achèvement : 31/12/77 Etat actuel : en cours Dernière mise à jour : 21/01/77	Scientifiques : N. TELLIER R. POCHARD

Objectif général :

Définition des différentes étapes du programme expérimental pour les prochaines années.
Utilisation des codes d'accident de lère génération pour définir le programme expérimental.

Objectifs particuliers :

Définition de la grille des essais - Définition des scénarios propres à chaque essai (manoeuvres de vannes, injection de secours...)

Etat de l'étude :

1) Avancement à ce jour :

Les premiers calculs préliminaires ont été réalisés à l'aide du code RELAP 4 MOD3.

2) Résultats essentiels :

Mise en évidence de l'importance de certaines séquences expérimentales (ouverture de vanne par exemple). Mise en évidence de l'importance des échanges thermiques avec les parois.

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Prochaines étapes :

Etude systématique des paramètres permettant de définir quantitativement la grille des essais.

Relation avec d'autres études :

Accident de perte de caloporteur des réacteurs à eau pressurisée : code de 1ère génération.
Etude de l'accident de perte de caloporteur primaire programmes expérimentaux.

Documents de référence :

"Calculs préliminaires PHEBUS - Résolution de quelques problèmes liés à l'utilisation d'un code à capacité et jonction", R.POCHARD, H.TARTU - Note Technique SETSSR 93.
"Calculs préliminaires PHEBUS - Etude des problèmes liés aux échanges avec les structures", R.POCHARD, A.PEDEL - Note Technique SETSSR 114.
"Préparation des données PHEBUS pour codes à capacités-jonctions", N.TELLIER, H.TARTU - Note Technique SETSSR 121.
"Boucle PHEBUS, 1ère phase des essais - proposition d'essais physiques hors neutrons", G.PAUMARD, R.POCHARD, N.TELLIER - Note Technique SETSSR 125.
"Notice de présentation de PHEBUS".

Titre	Pays : FRANCE
Expériences de dépressurisation et de renoyage d'une grappe combustibles PWR: Programme PHEBUS	Organisme directeur : CEA/DSN
Titre (anglais) Experiments on depressurization and reflooding of a PWR fuel assembly : PHEBUS project	Organisme exécuteur : CEA/DSN-SES. Responsable : A.TATTEGRAIN (SES)
Date de démarrage : 01/01/76 Date prévue d'achèvement : 31/12/81 Etat actuel : en cours Dernière mise à jour : 25/11/76	Scientifiques : P.DUTRAIVE

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.

Objectifs particuliers :

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 irradies.

Installations expérimentales et programme :

Ce programme sera entrepris sur l'installation PHEBUS en cours de construction au CEN/CADARACHE. Cette installation permet de réaliser les conditions thermodynamiques typiques d'un réacteur PWR sur une grappe de 25 crayons de 0,80m de longueur active. (1,20m 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.
Six à huit essais par an sont prévus à partir de mi-77.

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Etat de l'étude :

1) Avancement à ce jour :

L'installation est en cours de construction. Les dispositifs d'essais sont en cours d'étude. Des calculs préliminaires permettant de préciser quantitativement le programme sont en cours. Ils sont faits à l'aide des codes RELAP, DANAIDES, CERES.

Prochaines étapes :

Le démarrage des premiers essais est prévu au cours du second semestre de 1977. Les essais sur des dispositifs aiguilles doivent démarrer au début de 1979 et s'étendre jusqu'en 1982.

Relation avec d'autres études :

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

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/
Initiated 1973 Status	Completed 1975 Last updating : 20.01.75	Scientists M. SUREAU M. HOUDAYER E.D.F.

I - GENERAL AIM

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

II - PARTICULAR OBJECTIVES

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

- du modèle monocanal au multicanal,

- de la modélisation ponctuelle à la modélisation axiale (pour, en particulier, suivre les propagations d'ondes de pression),

- de l'équilibre au déséquilibre entre les deux phases du fluide.

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

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III - EXPERIMENTAL FACILITIES AND PROGRAMME

Toutes études expérimentales de support du programme français.

IV - PROJECT STATUS

4.1 - Progress to date

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

4.2 - Essential Results

V - NEXT STEPS

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

VI - RELATION WITH OTHER PROJECTS

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

VII - REFERENCE DOCUMENTS

Néant.

VIII - DEGREE OF AVAILABILITY

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

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

1. GENERAL AIM

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

This code was initially developped for pressure tubes reactors.

2. PARTICULAR OBJECTIVES

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

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

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

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

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

3. EXPERIMENTAL FACILITIES

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

4. PROJECT STATUS : Complete

5. RELATIONS WITH OTHER PROJECTS

DHTI loop in Ispra.

6. REFERENCE DOCUMENTS

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

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

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

7. DEGREE OF AVAILABILITY

To be discussed.

Title 1 (Original language)	Classification
Statistical analysis of randome signals	$\boxed{\text{I}}$ - 3 - 4 - 8 IO - I4
Title 2 (English)	Country ITALY Sponsor / Organisation } CNEN
Date initiated 1966 Date completed in progress Last updating April 1977	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

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:

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

4. Project status

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

5. Next steps

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

6. Relation with other projects

Not applicable.

7. Reference documents

Internal KEMA reports.

8. Degree of availability

Free on basis of exchange with other programmes and results.

Classification 1.1

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

1. General aim

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

2. Particular objectives

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

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

3. Experimental facilities and programme

4. Project status

Completed

5. Next steps

6. Relation with other projects

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

7. Reference documents

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

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

1. General Aim

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

2. Particular objectives

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

3. Experimental facilities and programme

4. Project status

1. Progress to date

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

2. Essential results

5. Next steps

6. Relation with other projects

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

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

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

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

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

7. Reference documents

8. Degree of availability

Available on exchange basis when completed

Classification 1.1

Title 1	COUNTRY Denmark SPONSOR Risø ORGANIZATION Risø
Title 2 NORHAV- P(B)WR blow down computer program (TINA)	Project leader: Peter Sten Andersen
Initiated: 1973 Completed: Status: Progressing Last updating:	Scientists: Peter Sten 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 distributions of coolant mass, -flow, -enthalpy and pressure as well as fuel rod temperature, cladding deformation and rupture in a P(B)WR core during the blow down phase of a loss-of-coolant accident. The model which is based on the subchannel approach includes slip and thermal non-equilibrium between steam and water. Opposite flow directions within a single channel (as well as different channels) are handled by the program.

3. Experimental facilities and programme

4. Project status

The thermo-hydraulic part of the program has been completed and is operational. It has been tested against the LOFT Semiscale blowdown experiments. Cladding deformation and rupture must still be specified by the user.

5. Next steps

Blowdown calculations will be performed for a PWR reactor for which data are avaiable. The influence of the radial power distribution and the non-uniform boundary conditions will be studied.

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.

7. Reference documents

8. Degree of availability

Available on exchange basis when completed.

Classification 1.1

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

1. But général

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

2. Objectifs particuliers

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

3. Installations expérimentales

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

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

4. Etat du projet

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

5. Prochaines étapes

Essais sur l'installation permettant d'atteindre 10 bars .... début 1975
Débits critiques avec des mélanges eau-air .... fev.mars 75
Extension de la gamme de titre (jusqu'à 20%) .... juin 75-Juin76
Influence de la géométrie élargissement brusque .... juin 75
effet de coude .... juin 76

A l'étude : boucle Super Moby Dick de 3 MW permettant d'explorer les régions à forte pression (110 bars) et faibles titres et moyennes pression (20 bars) et fort titres.

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6. Relation avec d'autres projets

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

7. Documents de références

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

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

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

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

CEN/G rapport interne TT/15 (1973)

Classification 1.1

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

1. <u>But général</u>

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

2. <u>Objectifs particuliers</u>

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

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

3. <u>Installations expérimentales et programmes</u>

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4. <u>Etat du projet</u>

   a) <u>code DANAIDES</u>

   Le code existe sous plusieurs versions :

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

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

   DANAIDES R : version réacteur de puissance.

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

   b) <u>code utilisant la méthode des caractéristiques</u>

   Première version en cours de tests.

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5. Prochaines étapes

Amélioration des modèles physiques des diverses versions de DANAIDES	1975
Amélioration de la méthode numérique afin de diminuer les temps de calcul	1975
Poursuite des tests du code des caractéristiques	1975
Dépouillement des essais OMEGA	1976
Dépouillement des essais PHEBUS	1978

6. Relation avec d'autres projets

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

7. Documents de référence

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

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

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

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 1.1. (1.2.)

Title 1	Country : JRC
Experimentelle Untersuchungen des Einflusses der DWR-Umwälzschleifen auf den Blowdown	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 : December 1977 (BMFT part A) Status : progressing 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 re-wetting of fuel rods heated up to high temperatures

RS- 77 : Investigation of the thermohydraulic non-equilibrium

RS- 81 : Mixing phenomena in parallel flow channels

RS-111 : Investigation of reactor pump behaviour during blowdown

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

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

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

7) 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

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:

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- 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. <u>Experimental facilities, computer codes</u>

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° 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° C operating pressure and temperature respectively) of different volume content (280 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 210° and 270° 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. <u>Progress to Date</u>

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

Classification: 1.1.1	
Title 1 (Original Language): Vorgänge bei der Druckentlastung wassergekühlter Reaktoren. Modellversuche mit einem 11,2 m hohen Stahlbehälter mit Einbauten. (RS 0016B - I.1.1., Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Battelle-Institut e.V., Ffm.
Title 2 (English): Investigation into the Phenomena Involved in the Depressurization of Water-Cooled Reactors. Experiments Using a Steel Vessel 11.2 m in Height with Internals.	Project Leader: Dr. T.F. Kanzleiter
Initiated (Date): July 15, 1972 Status: Continuing Completed (Date): July 31, 1977 Last Updating (Date): December 31, 1976	

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.

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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 external institutions. Some of these computer codes are also being used for licensing procedures.

5. Progress to Date

Ad 3.1: The four preliminary BWR and PWR experiments without internals (Nos. SWR 1R, SWR 2R, DWR 1R and DWR 2R) were carried out in 1974. Experiment SWR 2R was repeated in 1976 under improved conditions.

The main data of the latter experiment were as follows:

Experiment No.	SWR 2R
Accident simulated	steam line break
Cross section of rupture (diameter)	64 mm (sharp-edged orifice)
Rupture time	~ 1-3 ms
Length of discharge nozzle	555 mm (143 mm dia.)
Inside height of pressure vessel	11.2 m
Level of discharge nozzle	10.0 m
Inside diameter of pressure vessel	0.78 m
Initial pressure	71.1 bar
Initial temperature	286.8 °C (saturation)

Initial water level at 286.8 °C	7.0 m
Power of electric heater during blowdown	0

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 1976.

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

5. Results

Ad 3.1: The main goal of experiment SWR 2R was the investigation of depressurization, mass flow and behavior of the water level during the blowdown after a steam line break. The measured results showed an expansion of the steam volume with a pressure transient of approximately 30 bar/s and a mass flow of 230 kg/s during the first 250 ms. Subsequently the mass flow remains unchanged for about 2 s and violent generation of steam bubbles occurs so that the water level rises from its initial height of 7.0 m to the level of the discharge nozzle (10.0 m) at an approximately constant velocity of 1.45 m/s.

Ad 3.2: The evaluation of the PWR experiments with "flexible" internals furnished the following results:

- The loads exerted on the internals by depressurization waves are of approximately the same magnitude as calculated with computer codes using "best estimate" assumptions. (Model calculations with assumptions as used in licensing procedures give rather conservative results.)
- Fluid-structure interaction occurs during the period of maximum differential pressures.
- The core barrel is subject to circumferencial oscillations.
- Experiments with a long rupture time (e.g. 30 ms instead of 3 ms) showed no remarkable pressure waves and led to rather small loads on the internals.

7. Next Steps

Ad 3.3: - Preparation of the experimental facility for BWR experiments. Special emphasis will be laid on the construction of an improved loop and on the development of a new differential pressure transducer for direct installation inside the vessel.
- Carrying out the BWR experiments.

Ad 3.4: Specification of the experimental PWR program part II.

8. Relation to Other Projects

9. References

(1-4) Quarterly Reports in the Series "IRS-Forschungsberichte" (in German)

IRS-F-30 January to March 1976

IRS-F-31 April to June 1976

IRS-F-33 July to September 1976

IRS-F-34 October to December 1976

(5) IRS-F-29 "Research Reports (Annual Reports)" (in English)

(6) BF-RS0016B-32-6 "Technischer Bericht. Versuchsergebnisse vom Druckentlastungsvorgang im Druckbehälter mit flexiblen DWR-Einbauten. Versuch DWR 5A." August 1975

(7) BF-RS0016B-32-7 "Technischer Bericht. Versuchsergebnisse vom Druckentlastungsvorgang beim Referenzversuch SWR 2R." July 1976

(8) BF-RS0016B-31-4 "Technischer Bericht. Aufbereitung der auf Digitalband dargestellten rohen Meßdaten zu physikalischen Meßgrößen." June 1976

10. Degree of Availability of the Reports

The above-cited reports are available upon request from

Gesellschaft für Reaktorsicherheit - Forschungsbetreuung
Glockengasse 2, D-5000 Köln 1.

Reports (6) to (8) can be made available only by special agreement.

Classification : $\boxed{1.2}$ 1.1.1

<table>
<tr><td>Title 1 (original language)

EVA PROGRAM</td><td>Country : FRANCE

Sponsor : CEA
FRAMATOME

Organization

CEA
FRAMATOME
WESTINGHOUSE</td></tr>
<tr><td>Title 2 (English)

Two-phase flow pump test program.
Joint R & D program between FRAMATOME and CEA
with the WESTINGHOUSE Participation.</td><td rowspan="2">Project leader:

Mr. DELAYRE CEA
Mr. DUBOURG FRAMATOME

Scientists :

Mr. FAJEAU CEA
Mr. MARINI FRAMATOME</td></tr>
<tr><td>Initiated (date): JUNE 1974
Completed (date): DECEMBER 1976

Status: PROGRESSING
Last updating (date): JULY 1975</td></tr>
</table>

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

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.

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)	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

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
1.1.1 (1.1.2.)

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

1) General aim

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

2) Particular objectives

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

- a sudden expansion of water
- a periodic volume variation
- injection of cold water in a 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 :frequenc

- 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: 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	

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 major accidents such as Loss Of Coolant Accidents
- The deformation in the reactor internals immediately following a LOCA.

Experimental facilities and program : -

Project status
Although the development of a special computer program is not excluded the first computational approach of the problem will be done with programs such as NASTRAN, MARC-CDC and TOODY. Detailed computation of construction parts can be done with the latter two programs where local effects such as non-linear material properties can be included.

Next steps : -

Relation 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 1977 + 1978 US $ 48.000

Personnel : 1977 + 1978 : 0.8 manyear
Technical assistance not included

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.

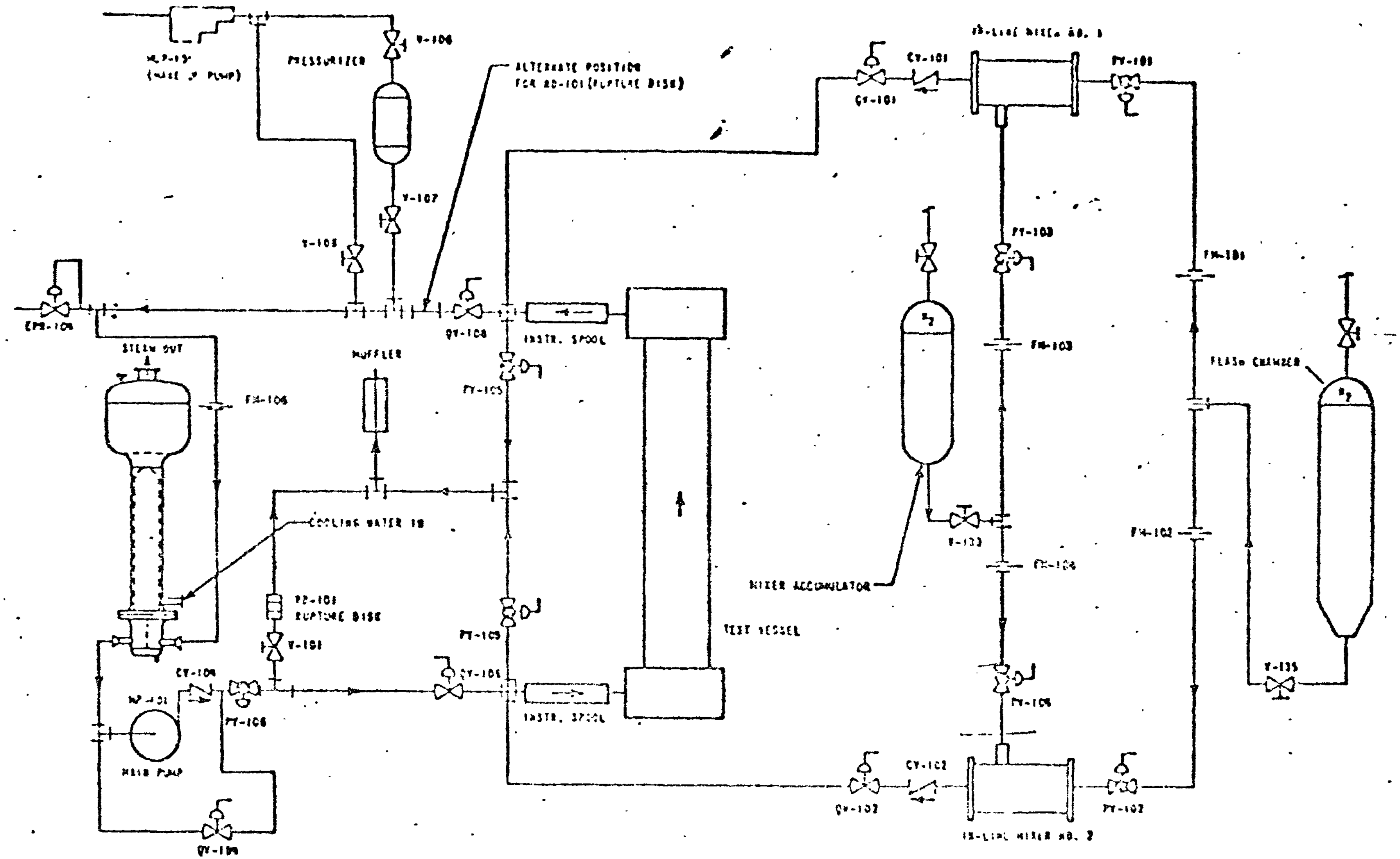


FIGURE 1 : BLOWDOWN HEAT TRANSFER TEST LOOP SCHEMATIC

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

1. General Aim

The main objective of this experimental program was to study the thermohydraulic and heat transfer behaviour in multi-rod bundles during blowdown processes.

Experimental data were requested to determine

- the time, when critical heat flux (DNB) occurs (DNB-delay time)

- the heat transfer behaviour immediately after DNB occurs

- the thermohydraulics and heat transfer behaviour in the Post-DNB region of the blowdown.

2. Particular Objectives

During a blowdown process with large leaks two phases of transients are characteristic. In the first seconds of a postulated rupture in the primary coolant system steep transients of pressure and mass flow in the core as well as heat flux density of the fuel rods occur. In the later part of the blowdown a phase of small gradients follows.

In order to reach sufficient high accuracy of the measurements the initial phase of large pressure gradients (DNB experiments) had to be investigated separately from the region of small pressure gradients (Post-DNB-experiments).

All experiments have been carried out with an electrically heated 25 rod bundle of PWR dimensions. The experimental investigations under hot channel conditions required special adaption of the test facility.

3. Research Program

The first part of the experimental program - the DNB-tests - included blowdowns under 11 different specified conditions, during which DNB-delay time, heat transfer behaviour and thermohydraulic behaviour in the first phase of the blowdowns was studied. The experiments were concentrated on hot channel conditions. Main variables of the conditions specified were the mass flow rate, the inlet enthalpy and the exit enthalpy at the beginning of the blowdowns.

In the Post-DNB-experiments data were received about the heat transfer coefficients in the later part of a blowdown at film boiling conditions. A wide range of mass flow, pressure, steam quality and pressure gradients was studied to simulate conditions in the rod bundle similar to those expected by assumed ruptures of different rupture area of the heat transfer in the Post-DNB region.

4. Experimental Facilities

The experimental facilities are described under "test equipment for controlled bundle measurements (BET)", Project RS 37, 37/1 and 37/2 I.2.2, Annual Reports No. A 74.

The facility was modified by adjusting a cold water injection system down stream of the bundle. Thus fast transients could be achieved at the beginning of the blowdown with hot channel conditions. In addition the volume upstream of the test section was reduced, which means less flashing volume.

5. Progress to Date

The first part of the experimental program included a total of 21 DNB experiments. In 11 different specified conditions transients have been simulated under conditions which may occur in assumed ruptures between pressure vessel and steam generator or steam generator and circulating pump of PWR's. The experiments have been started at steady state hot channel conditions and 141 bars.

After the DNB tests the heat transfer properties were studied in the Post-DNB-region.

44 tests runs were started with 13 different test parameters:

Rel. mass flow density:	0,04 - 0,4	
Pressure:	110 - 10	bar
Blowdown-time:	15 - 50	s
Pressure gradient:	up to 15	bar/s
Initial enthalpy:	1085 - 1520	kJ/kg

Work was concentrated on low mass flow densities. The control of the experiments acured in accordance with precalculated data by a process-computer.

The Post-DNB-situation with heater temperatures between 500°C and 700°C was adjusted some seconds before the pressure transient was started.

After this the test equipment was converted to run a special standard blowdown-test for checking computer programs.

Finally some tests were run to determine the stream resistance value ζ of the test loop.

All measured data were evaluated.

65 blowdown-tests have been run with the bundle within 6 months, without major failures in the instrumentation. Only during the last 5 blowdowns some of the thermocouples at the electrically heated rods failed.

6. Results

The experimental study of the heat transfer behaviour and thermohydraulic behaviour within a 25 rod bundle was completed successfully. Detailled information were received about DNB-delay time, DNB location and heat transfer behaviour right after DNB under hot channel conditions. Many experimental data on heat transfer were received in the later part of the blowdown when film boiling occurs.

In addition the temperature, pressure and pressure difference curves of the instrumented test section include many informations of the local heat transfer behaviour and the hydraulic behaviour within the test bundle.

With regard to the test bundle the behaviour was very satisfactory. Including the pre-testing phase a total of more than 100 blowdowns were carried out without any damage. After about 50 blowdowns a leak occurred in 1 rod.In order to close this leak the instrumentation of the rod had to be cut. Apart from that only about 5 of about 100 thermocouples within the bundle failed in the very late part of the experimental works.

Classification: 1.1.2

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

1. General aim

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

2. Particular objectives

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

3. Experimental facilities and program

Two loops had been available for the experimental part of this investigation. The different aggregates of the circuit and the used measuring techniques had already been described in detail in the reports IRS-F-12 and IRS-F-18.

The test program during the whole period was separated into parts:

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

4. Project status

4.1 Progress to date

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

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

4.2 Essential results

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

References

/1/ KWU (AEG), Notkühlprogramm DWR and SWR, AEG-E3-1941, Mai (1971)
/2/ KWU/R5-2882, Abschlußbericht über die Abblaseversuche mit 4-Stabbündeln Großwelzheim, Jan. 1974
/3/ Rohsenow W., Griffith P.: Correlation of maximum heat transfer data for boiling of saturated liquids
Chem. Engng. Progr. Symp. Ser. 52, Nr. 18, 47 (1956)

5. Next steps

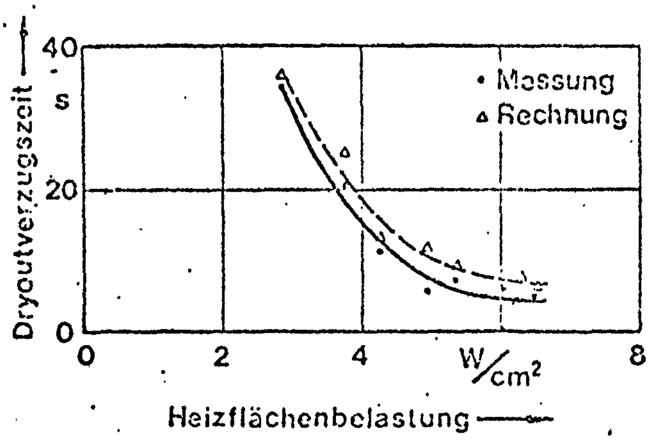
The investigations had finished in March 1975.

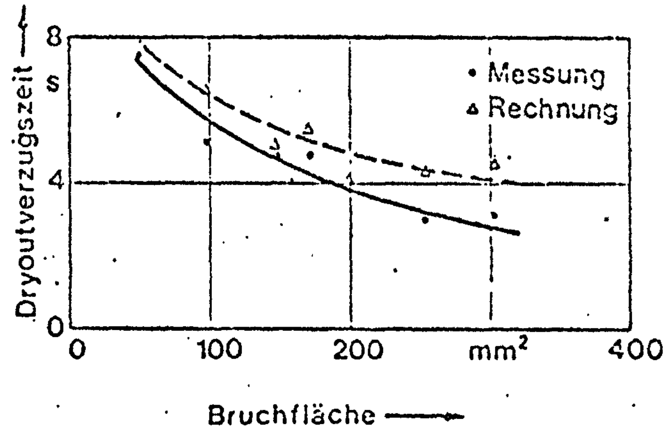
6. Relation with other projects

RS 37, RS 37-1 and RS 37-2
"Investigations of the Events within the Reactor Core under loss of Coolant and Emergency Cooling Conditions, High Pressure Experiments" at KWU, Großwelzheim.
RS 64: "Investigations of the steady and transient Critical Heat Flux of Multirod Bundles for PWRs and BWRs with Freon as Model Fluid".

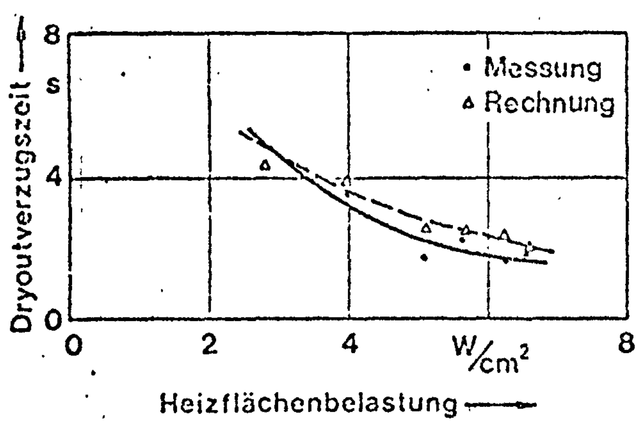
Comparison of measured and calculated dryout delay (4-rod bundle) as function of

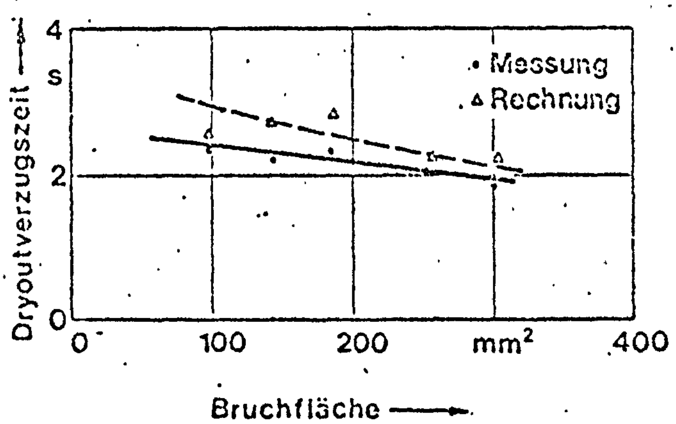
hot leg break: heat flux ↓ cross section of break area ↓

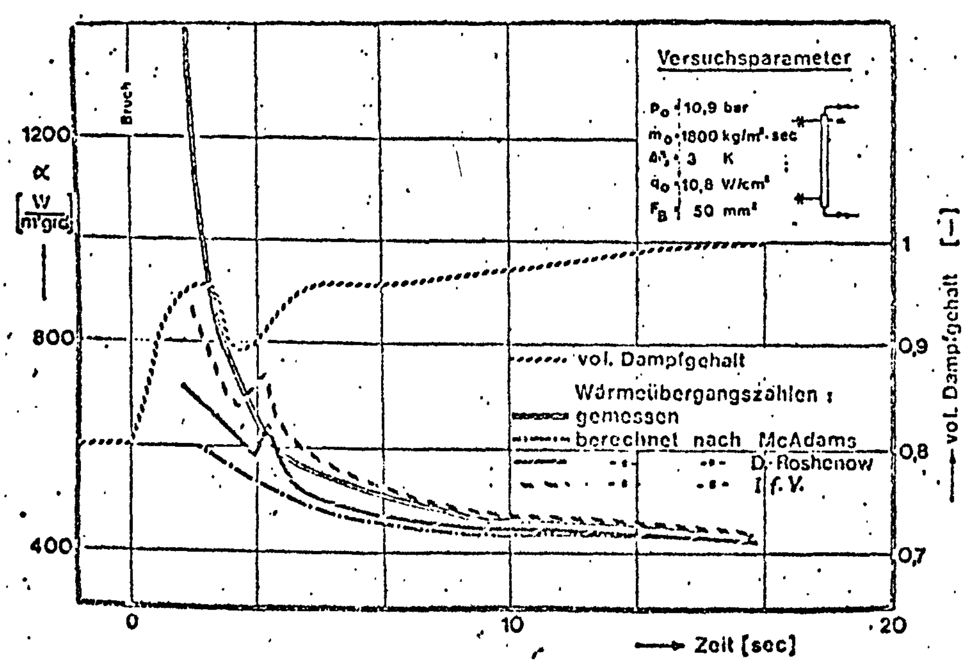




cold leg break: heat flux ↓ cross section of break area ↓







comparison of measured and calculated heat transfer coefficients

7. Reference documents

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

8. Degree of availability

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

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

1. General aim

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

2. Particular objectives

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

3. Experimental facilities and program

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

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

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

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

4. Project status

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

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

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

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

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

4.2 Essential results

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

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

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

$$\Pi_1 = \frac{\dot{m}_{tot\ inlet} \cdot \varepsilon \cdot d''}{\nu'' \cdot \rho''} \cdot Re''$$

considers the influence of mass flow

$$\Pi_2 = \frac{\partial p/\partial z \cdot dz - \tau}{\sigma} \cdot \ell$$

considers the influence of test section geometrie and the most important physical properties of the fluid

$$\Pi_3 = \frac{\dot{q}_0}{\dot{m}_{tot.\ inlet} \cdot r_0} \left(1 - \frac{\Delta h}{r_0}\right)$$

considers the influence of heat flux and initial subcooling

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

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

5. Next steps

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

6. Relations with other projects

RS 37; 37-1; 37-2

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

RS 48

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

RS 64

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

RS 179

Phaseseparation

7. Reference dominant

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

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

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

8. Degree of availability

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

9. Budget

1 year (10 month) DM 284.600.00

10. Personell

References

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

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

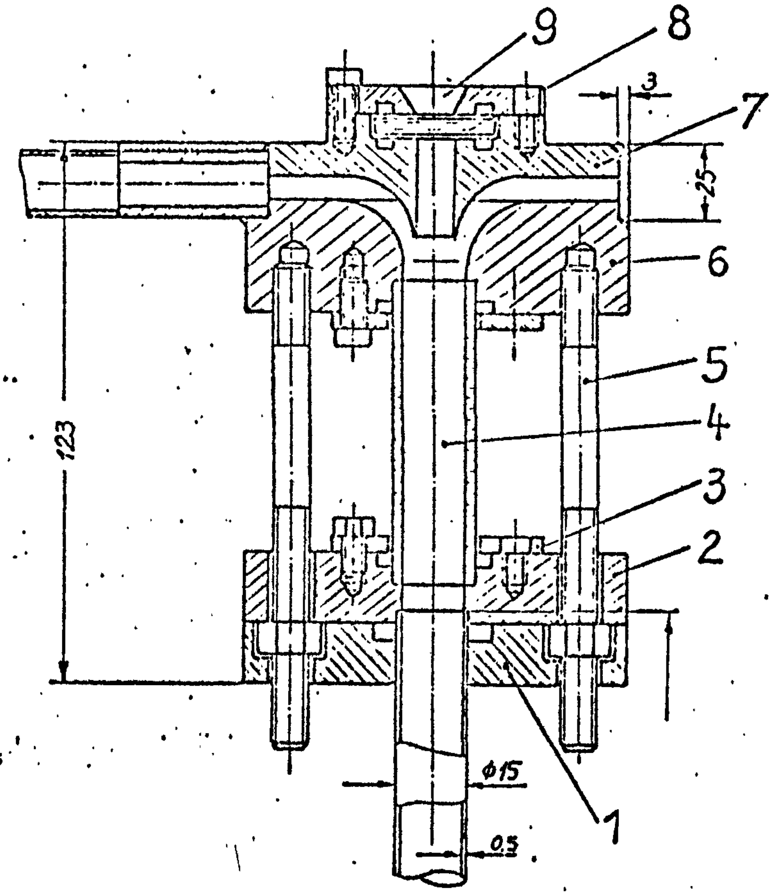


Fig. 1: Slit device at the end of the heated test-section

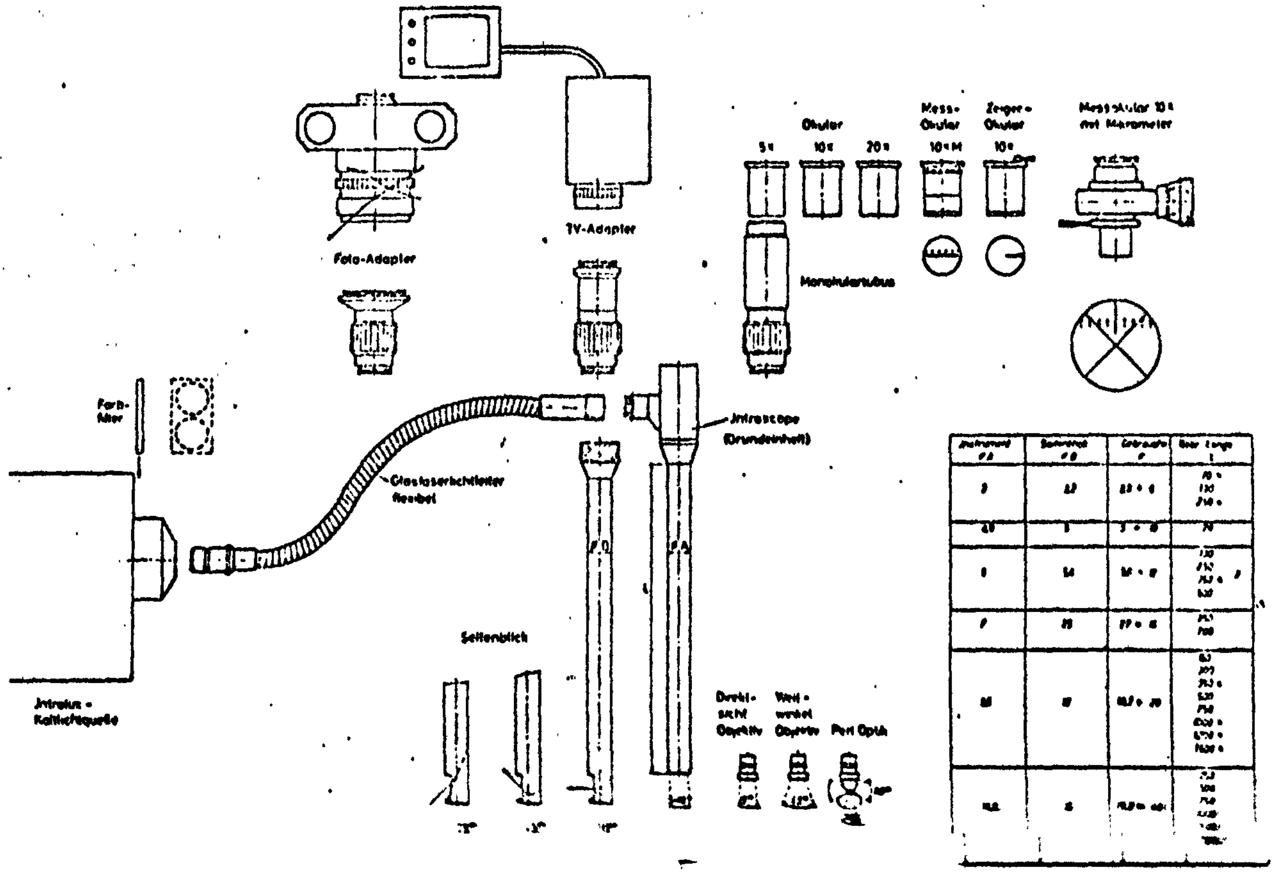
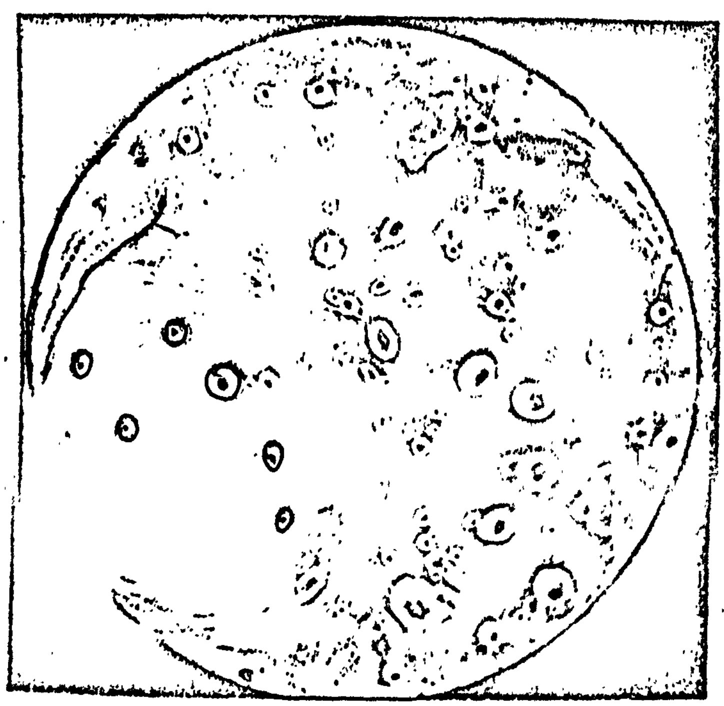


Fig. 2: Glass fibre optic equipment



$\dot{m}_{inlet}$ = 250 kg/m² sec

p = 8 bar

l_{heated} = 5 m

ε_{out} = 0,9

ϑ_{out} = 308 K

$\dot{Q}$ = 5,4 KW

Fig. 3

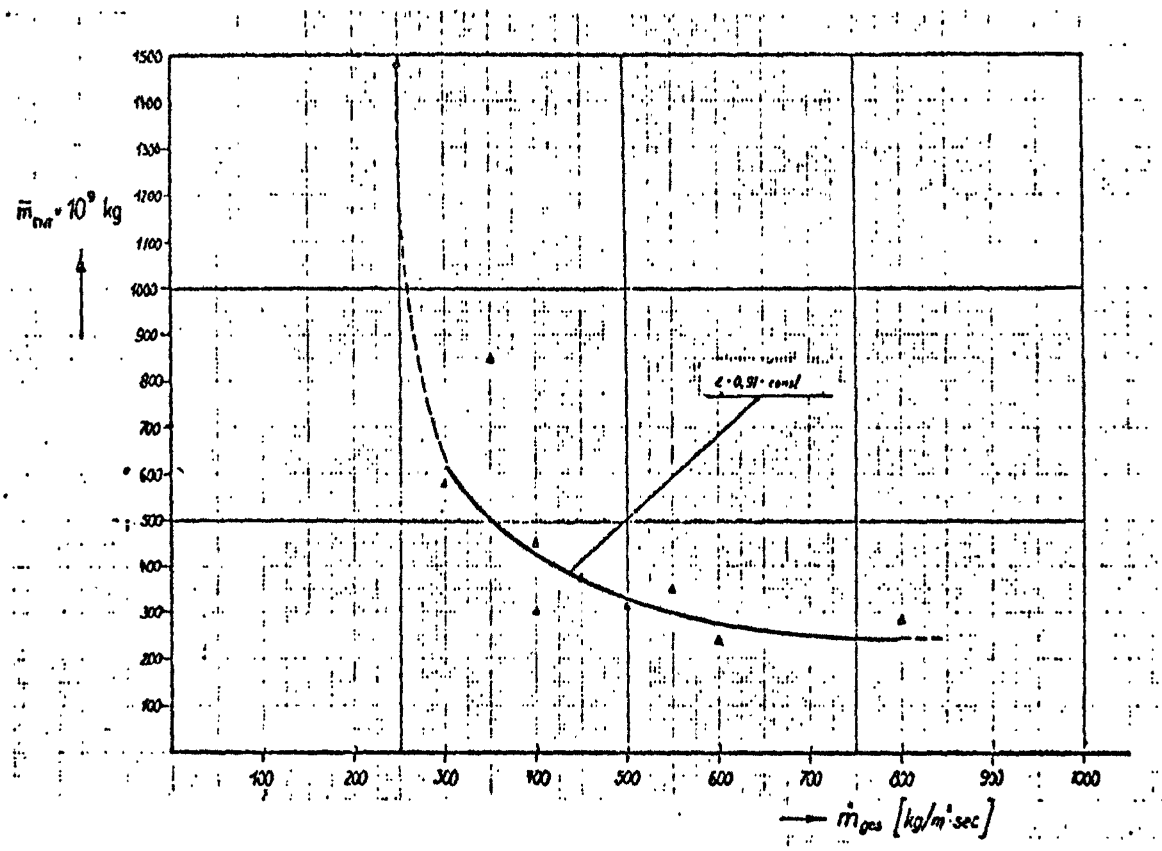


Fig. 4: Entrainment as a function of total mass flux

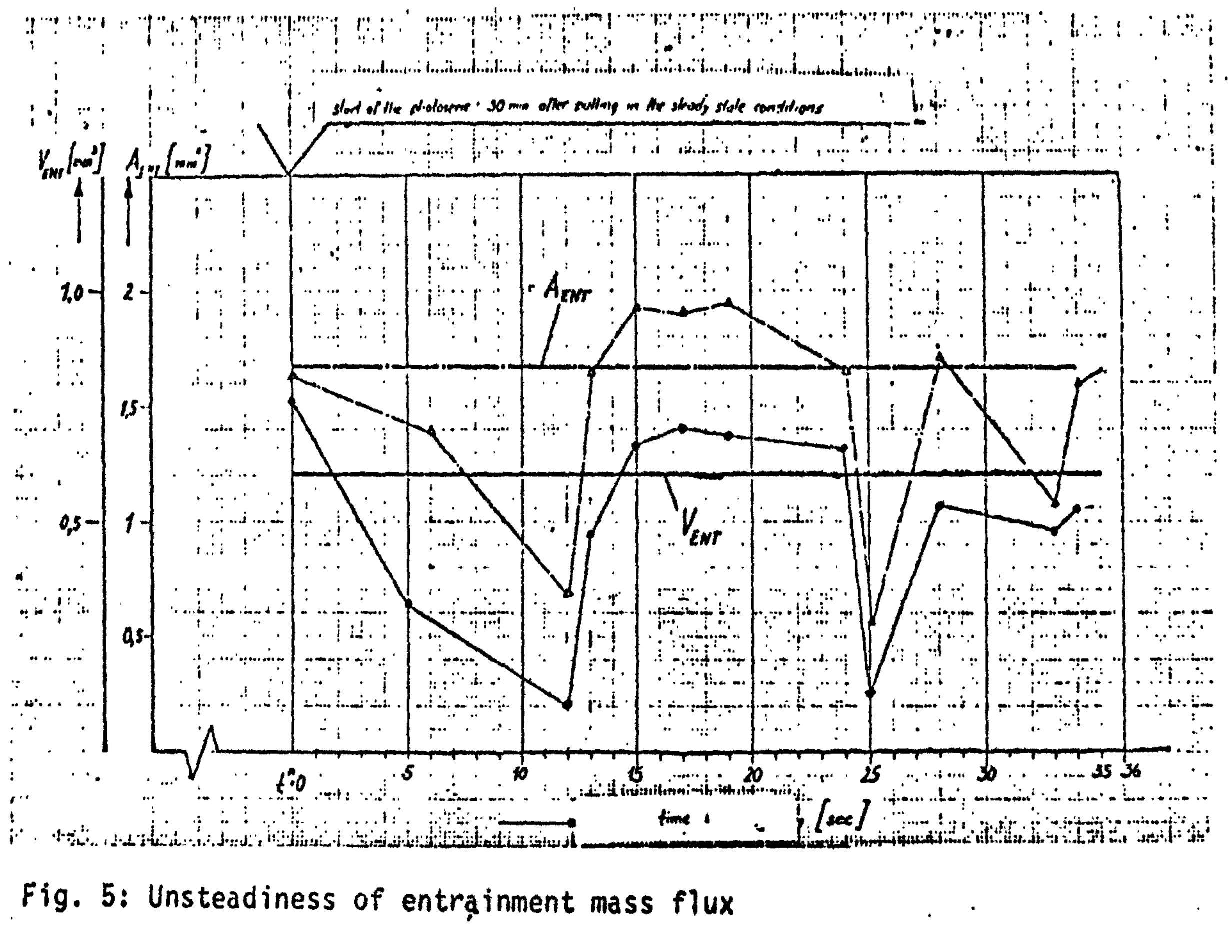


Fig. 5: Unsteadiness of entrainment mass flux

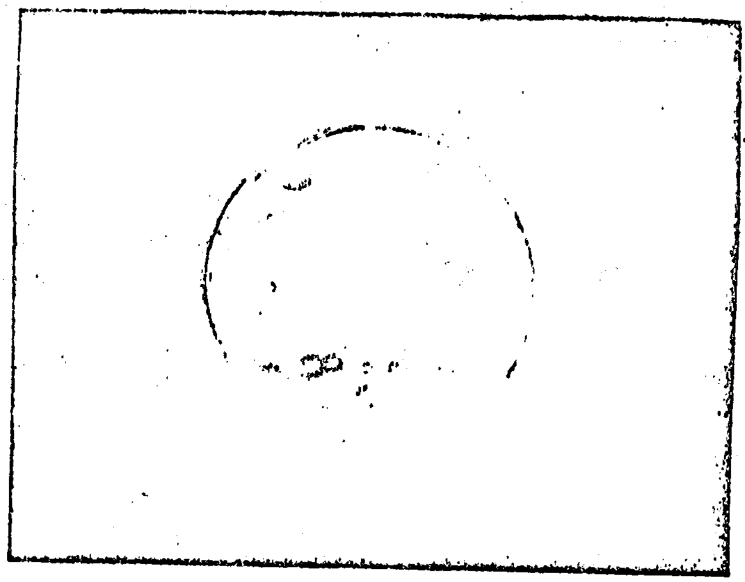
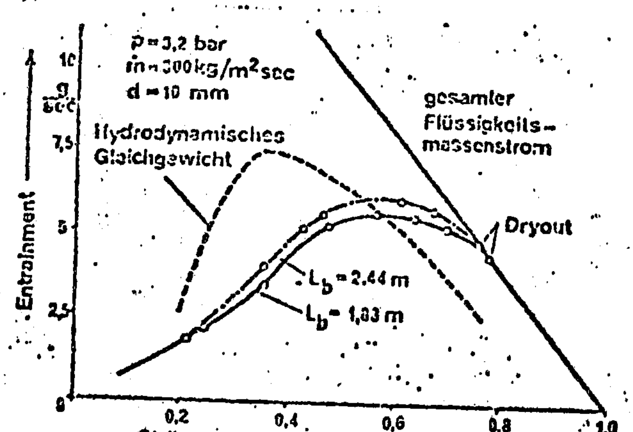


Fig. 6: Bubble flow - picture taken with a glass fibre optic



Entrainmentverlauf im beheizten Rohr (nach Bennet et al

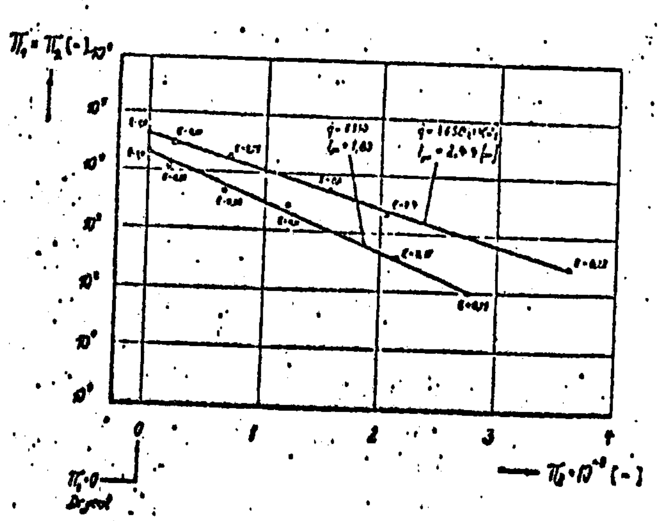


Fig. 7/8: Entrainment measurements of Bennet /2/

Classification: 1.1.2	
Title 1 (Original Language): Theoretische und experimentelle Untersuchungen zum thermohydraulischen Verhalten des Reaktor-Cores in der ersten Blowdown-Phase (RS 163 - I.1,1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: T.U. Hannover
Title 2 (English): Theoretical and Experimental Investigations on Thermo- and Fluiddynamic Behaviour of the Reactor Core during the first Blowdown Phase	Project Leader: Prof. Dr.-Ing. F. Mayinger
Initiated (Date): March 75 Status: Continuing Completed (Date): Dec. 78 Last Updating (Date): Dec. 1976	

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 for the loss of coolant accident and the following emergency cooling phase.

2. Particular objectives

The main problems are the experimental and theoretical investigations of dryout-delay time and heat transfer under consideration of entrainment- and mixing-processes. Furthermore the influence of loop components on the hydrodynamic behaviour in the reactor core during LOCA-conditions must be tested at a 16-rod bundle

3. Experimental facilities and program

During this report period first blowdown tests had been carried out at a single tube test section in the parameter range

$$\dot{m}_{tot} = 1000 \ldots 2000 \text{ kg/m}^2\text{sec}$$
$$\dot{q} = 1{,}75 \ldots 6{,}642 \text{ W/cm}^2$$
$$h_E = 3 \text{ kJ/kg}$$
$$(p/p_k)_{H_2O} = (p/p_k)_{R12}$$
$$A_q = 10 \ldots 70 \text{ mm2}$$

with the aim to evaluate a flow pattern map for transient conditions.

To get more information about the parameters influencing the complex procedure of droplet entrainment and -deposition steady state tests at fixed conditions had been carried out contemporaly.

At the beginning of the mixing investigation the steady state macroscopic crossflow was tested with air-water mixtures in a special mixing test section. By use of a transparent test section material the high frequent mixing movement could be made visible with the aid of high speed cinematography. Synchronously with the films the variations of the static pressures in subchannels were recorded. Therefore detailed information of the mixing phenomena could be gained.

4. Project status

In the following tabular a specified survey of the investigations duri this report period are given.

1. Blowdown tests to evaluate a flow pattern map for transient conditions
2. Entrainment experiments at steady state conditions with miniature camera and high speed films
3. Analysis of the test results
4. Mixing mechanism and local cross flow along the test section
5. Integral cross flow behaviour of the two phases and pressure drop behaviour
6. Test calculations with the modified COBRA-IIIC computer program

In the first seconds after the simulated break a flashing and foaming of the fluid in dependance of the temporal pressure gradient can be recognized, with a rapid increase in phase velocity, so that a homogenous flow can be assumed. As quantitative optical measurements - so e.g. high speed films with the s.c. "axial view method" in direction of the flow - failed, in this region of the flow a capacitive measuring method had been tested to describe the void fraction distribution across the cross section area. Besides these experiments, optical investigations of the entrainment behaviour at steady state conditions were carried out to get an idea of the parameters influencing the entrainment behaviour.

In the mixing investigations the void fraction was measured by means of a gamma ray attenuation method at different locations along the height of the test section. The flow terms were obtained by photographic method and by measuring the quality as a function of the void fraction. Summing up these components the crossflow along the channel length could be fixed.

The analyzing of high speed films supplied the qualitative interpretation of the mixing phenomena. A physical performance could be obtained from this.

5. Essential results

An example for a transient flow pattern map gained in the blowdown test by the aid of high speed films is shown in fig. 1. It is interesting that the burnout effect in all tests even at the smallest break area of 10 mm^2, which corresponds to a 0,25 F break in reality, could be identified as a dryout. The assumption that the boiling crisis especially at small break areas could occur in form of a DNB could not be corroborated. But with regard to this special case, further investigations with break areas of 1 mm^2 and 2 mm^2 are planned for the next time.

An equation to evaluate the entrainment mass flow from the high speed pictures is given in fig. 2. Results of these tests under steady state conditions are shown in fig. 3, where the entrainment mass flow related to the total mass flow is plotted against the quality. We found that the entrainment mass flow is nearly independant from system pressure and subcooling at the inlet of the test section but only a function of total mass flow and quality.

The entrainment behaviour along a heated channel can be described by aid of fig. 4. The changing film flow rate is caused by two different effects.

1) by evaporation
2) by entrained droplets

$$\left(\frac{\partial \dot{M}_{FILM}}{\partial z}\right)_{TOT} = \underbrace{\left(\frac{\partial \dot{M}_{FILM}}{\partial z}\right)_{Q}}_{EVAPORATION} + \underbrace{\left(\frac{\partial M_{FILM}}{\partial z}\right)_{E}}_{ENTRAINMENT}$$

The evaporation term could be calculated by a simple energy balance leading to the equation

$$\left(\frac{\partial \dot{M}_{FILM}}{\partial z}\right)_{Q} = \frac{\dot{q} \cdot \pi \cdot d_0}{r_0}$$

The total film flow rate can be calculated in dependance of the entrainment curves in fig. 2, so that the total change of the film flow rate is defined too.

Fig. 4 shows the total change of the film flow rate and the change of the film flow rate due to the evaporation as a function of local quality. The difference of both must be the change of the film flow rate caused by entrainment droplets from the film surface.

At low quality - this means at relatively large film thickness - the total change of the film is much higher than predicted as a result of evaporation - equation (2) - so that in this part of the tube there is net liquid entrainment. Above a certain quality, where the changing of the film flow rate is exactly equal to the evaporation rate calculated by equation (2), the changing of the film flow rate is lower than the change in dependance of heat flux. Accordingly, in this part of the tube must be net liquid deposition.

Furthermore, another single effect was investigated: the slip ratio between the droplets and the vapour phase. Therefore pictures with a special double exposing method had been taken, to measure the way a droplet travels in a certain time. In fig. 5 the slip ratio $S'=W_{ENT}/W_D$ is plotted versus the droplet diameter. In the theoretical part of this work an equation to describe the slip ratio S' was evaluated, basing on the drag force acting on a particle being transported in a gas stream. This equation is

$$S' = 1 - \frac{d_{Tr}}{w''} \cdot 9{,}1 \cdot 10^{4} \cdot \left(Re_{TR}\right)^{-0{,}25}$$

droplet diameter d_{dr},
vapour velocity w''

$$Re_{TR} = \frac{w'' \cdot d_{TR}}{\nu''}$$

With the knowledge of the multiplicity of the thermohydraulic behaviour in relation to the subchannel analysis it will be tried to obrain a mathematic physical model conception by experimental single effect studies. The mixing investigations with an adiabatic air-water loop already carried out during this report period give essential indications to the magnitude of influencing parameters on cross flow behaviour simulating steam-water at saturation condition.

The cross flow for both kinds - one side and both sides injection -

could be determined. For the case of one-side injection figures 6 and 7 show the cross flow of both phases along the channel length in relation to the mass flow rate at the inlet. The gaseous diversion cross flow could be found to be nearly independent of the mass flow rate of liquid. The behaviour of the gaseous diversion cross flow is nearly linear and appears only in the direction of the compensating volumetric void fraction, but the macroscopic cross flow of liquid changes the direction. Only at relatively low injected mass flow rates a cross flow of two phases in the opposite direction could be observed. In contrary to the measurements under condition of one-side injection the results of both-sides injections show a lower gaseous cross flow based on lower injected differences of mass flow shown in figures 8 and 9. But the diversion cross flow of liquid shows only the same behaviour at both-side injections as one-side injection at the end of the channel length. By means of the design principle of the test section and the isocinetic entrance- and exit-conditions the mixing behaviour can be reproduced over a cross flow length up to the total mixing.

Figures 10 and 11 show the measured total cross flows of two phases. Behind a passage region the influence of the injection kind is negligible and behaves parallel to the defined total cross flow. A connection between the cross flow of liquid and the partial pressure terms resulting in the total pressure drop at one-side injection is shown in figures 11 and 12. In the graph it can be seen that at low injection and at equally low quality the decrease in static head is larger than the increase in two phase friction pressure loss. For this case the increase in total pressure drop is negligible. Referring to figure 11 it can be remarked that the opposite cross flow of two phases occur only in this section. The behaviour of liquid cross flow tries here mainly to compensate the hydrostatic pressure until total pressure balance between the channels is obtained.

At higher injected mass flow the same cross flow directions could be measured for two phases i.e. a redistribution avoids the axial friction forces in favour of the cross flow.

6. Next steps

In the next steps measurements of the void fraction distribution across the cross section area at steady state and transient condition will be performed. The results of these measurements must be compared with the obtained photos and γ-ray measurements. The investigations in the physical phenomena of entrainment must be continued to evaluate an entrainment calculation model.

Further experimental studies of the mixing investigations will be carried out with the aid of a boiling two channel test section and the coolant Freon 12. In this connection steady and transient investigations are provided with diverse gap widths. At the COBRA-IIIC version the input-output will alternatively be modified from American units to SJ-units. Parallel to this the routines of physical properties will be tested in order to take them up into the MJT-MOD-COBRA-IIIC-Code.
For the experimental investigations in the loop components behaviour a 4-rod-test-section will be installed for the first measurements. These measurements are planned for the next time.

7. 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

8. 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

9. Degree of availability

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

10. Budget

This report period: 238.000,-- DM

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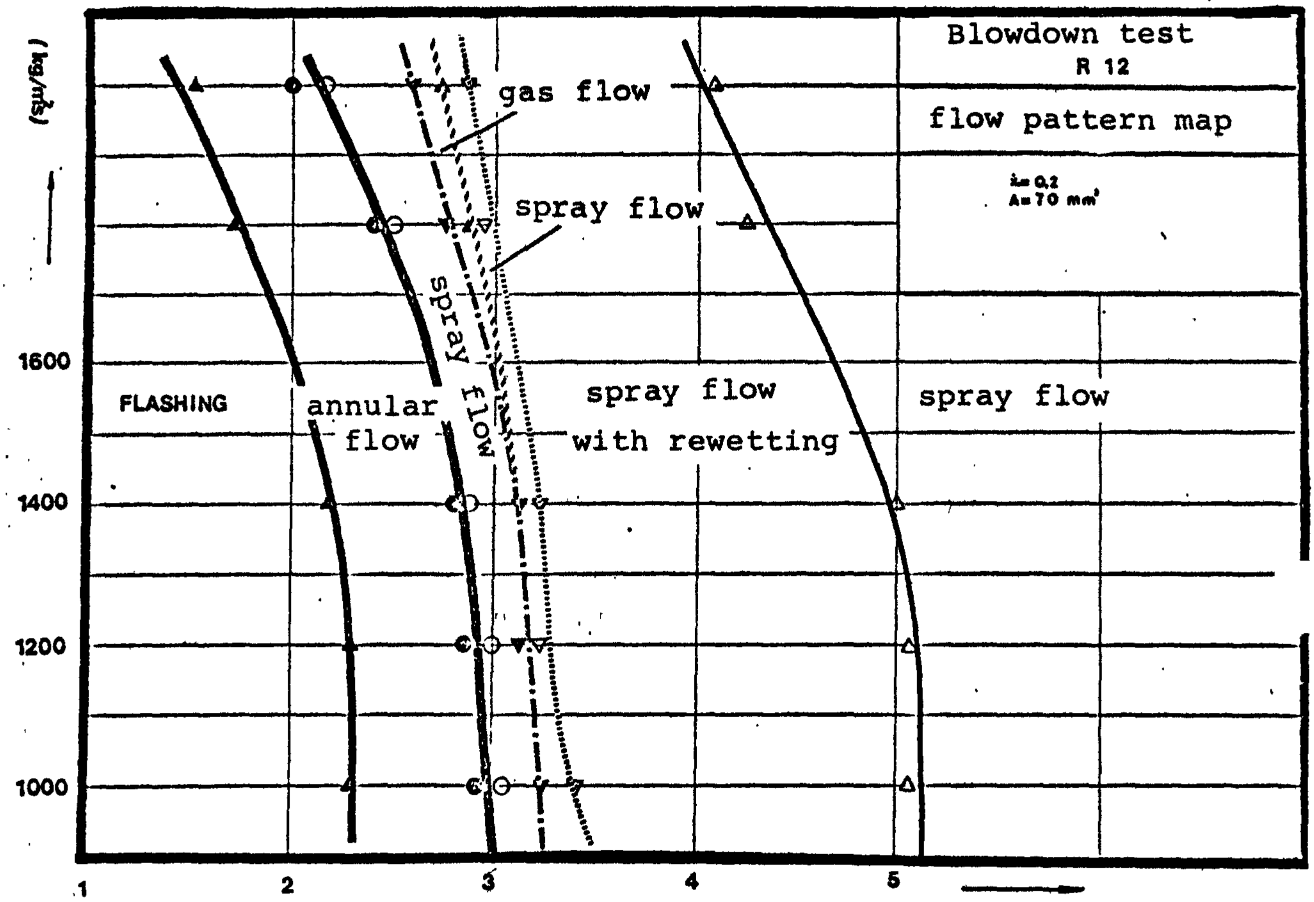
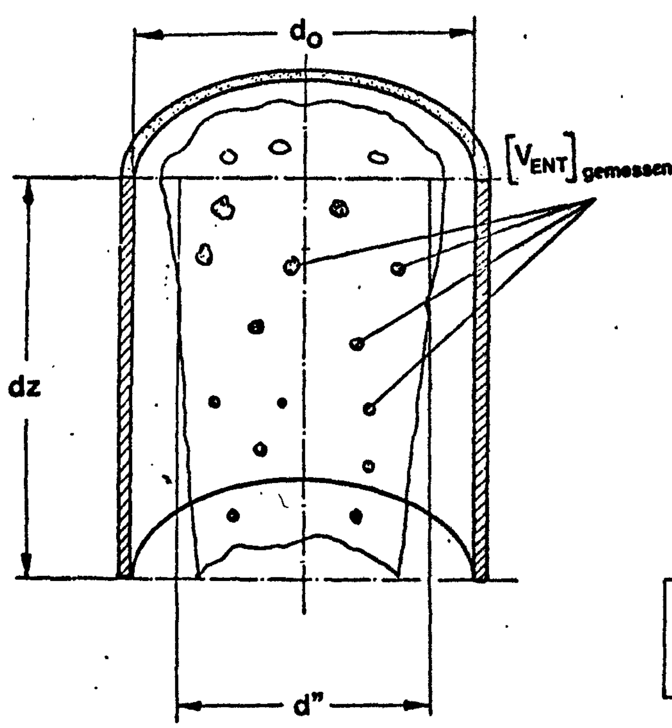


Fig. 1: Flow pattern map for transient conditions



$$S' = \frac{w_{Ent}}{w_D}$$

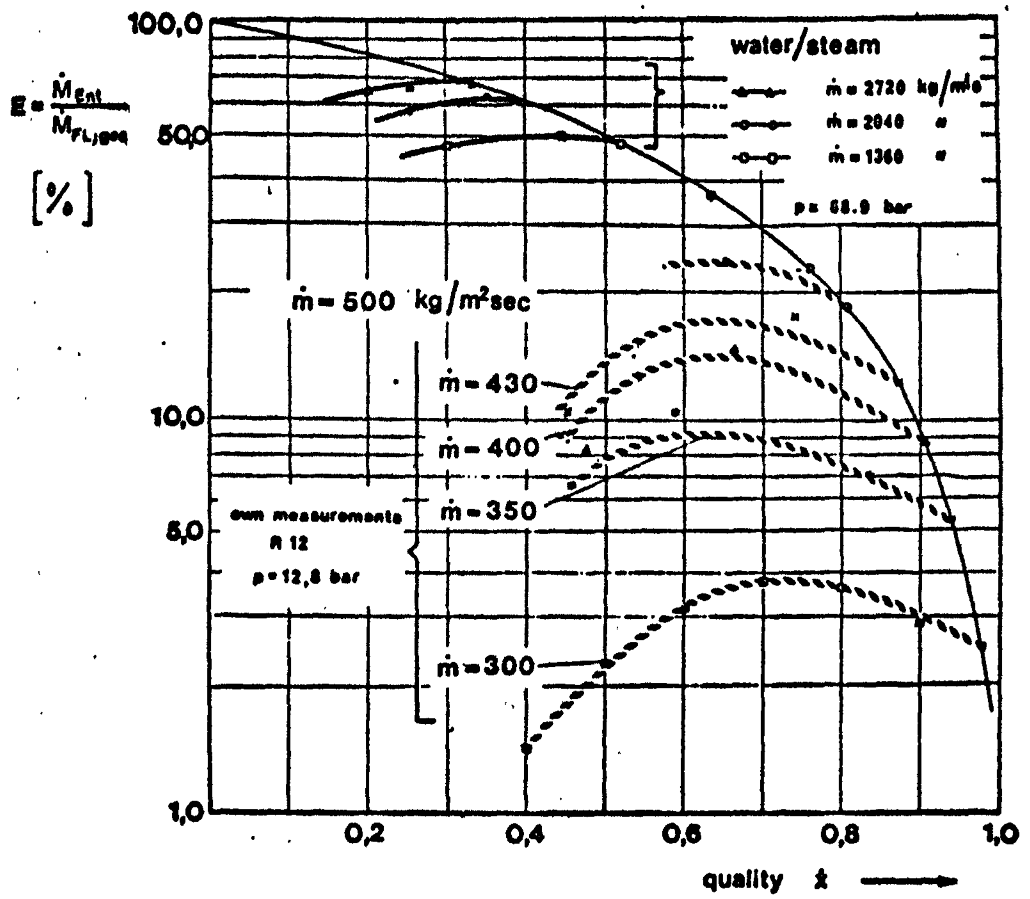
$$S' = \frac{\dot{V}_{Ent} \cdot A_D}{A_{Ent}\,\dot{V}_D} = \frac{\dot{M}_{Ent} \cdot \varrho_D\, A_D}{\varrho_{Ent}\, A_{Ent}\, \dot{M}_D}$$

$$S' = \frac{\dot{M}_{Ent} \cdot \varrho_D \cdot A_{ges} \cdot \varepsilon}{\varrho_{Ent} \cdot A_{Ent} \cdot \dot{M}_{ges} \cdot \dot{x}} \;\Big|\; \frac{dz}{dz}$$

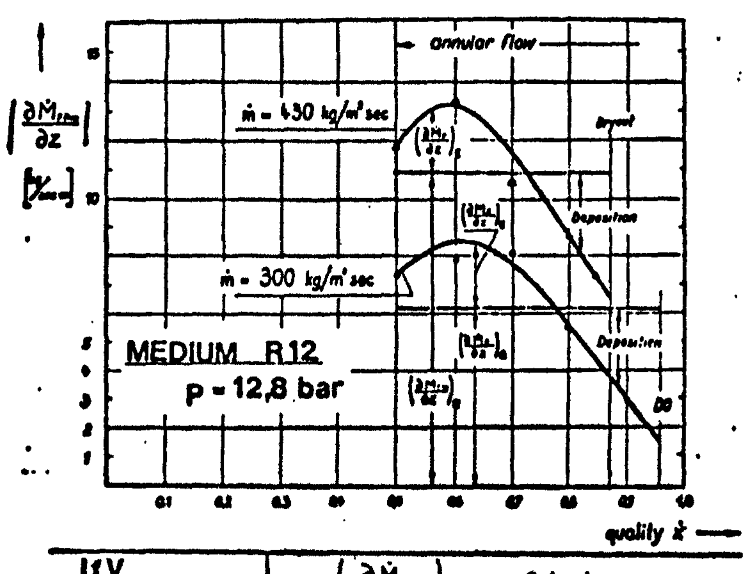
$$\boxed{\dot{M}_{Ent} = S' \cdot \dot{M}_{ges}\, \frac{[V_{Ent}]_{gem} \cdot \varrho_{Ent}}{A_{ges} \cdot dz \cdot \varrho_D} \cdot \frac{\dot{x}}{\varepsilon}}$$

$$\boxed{E = \frac{\dot{M}_{Ent}}{\dot{M}_{Fl.ges}} = \frac{S'}{(1-\dot{x})} \cdot \frac{[V_{Ent}]_{gem} \cdot \varrho_{Ent}}{A_{ges} \cdot dz \cdot \varrho_D} \cdot \frac{\dot{x}}{\varepsilon}}$$

IfV TU Hannover	**Entrainment mass flow** **- calculation model -**	Fig. 2



IfV TU Hannover | $E = \frac{\dot M_{ENT}}{\dot M_{Fl;ges}} = f\,(\text{quality})$ fig. 3



IfV TU Hannover | $\left(\frac{\partial \dot M_{Film}}{\partial z}\right)_{TOT} = f(\dot x)$ fig. 4

drag force on a particle:

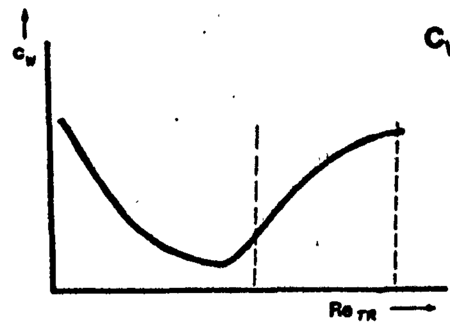
$$F = c_W \cdot A_{PROJ} \cdot \frac{\varrho'}{2} \left(|w'' - w_{TR}|\right)^2$$

definitions:

Slip ratio between droplets and vapour: $S' = \frac{w_{TR}}{w''}$

$$A_{PROJ} = d_{TR}^2 \cdot \frac{\pi}{4}$$

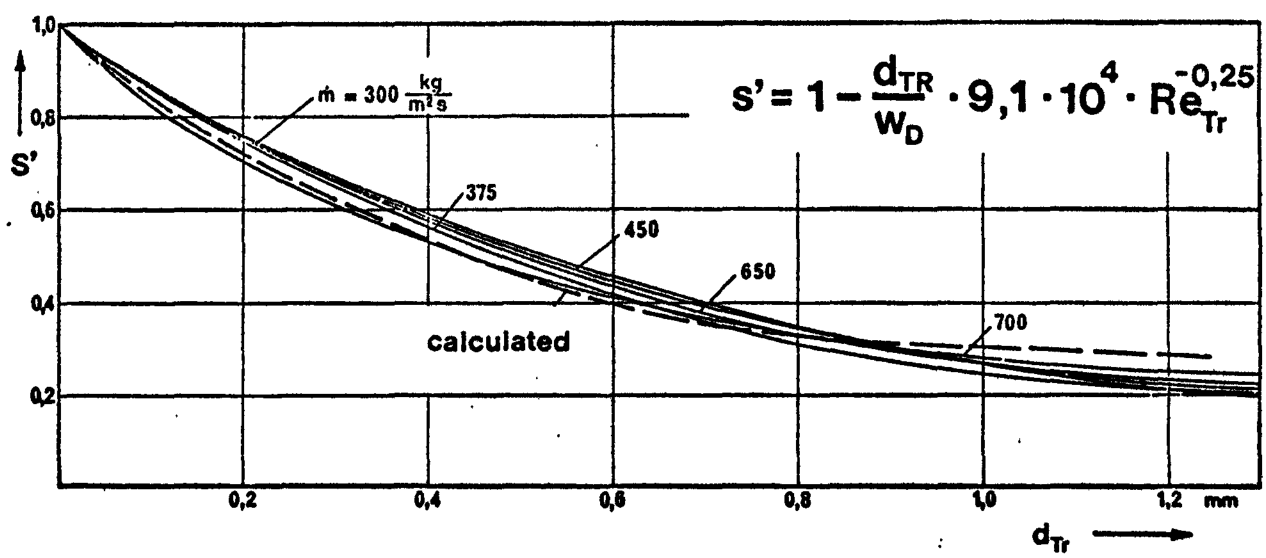
$$c_W = C_1 \cdot Re_{TR}^n \quad ; \; Re_{TR} = \frac{w'' \cdot d_{TR}}{\nu''}$$



drag force at steady state: $F = G$

$$\boxed{S' = 1 - \frac{d_{TR}}{w''} \cdot 9{,}1\ 10^4 \cdot Re_{TR}^{-0{,}25}}$$

fig. 5



Slip ratio between vapour and droplets
- calculation model -
$s' = \frac{w_{Tr}}{w_D}$

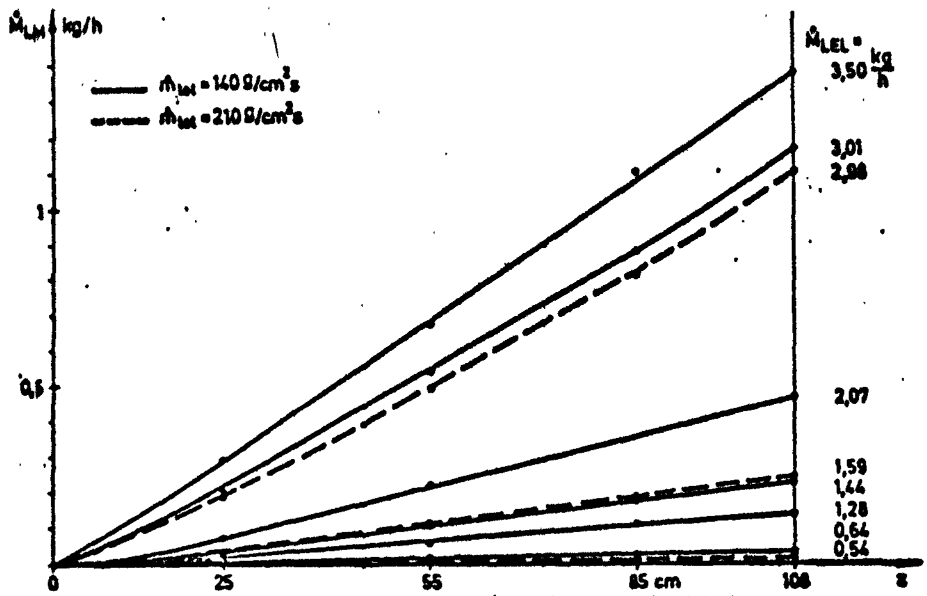


Fig. 6 : Mass flow of diversion gas versus channel length on one-side injection

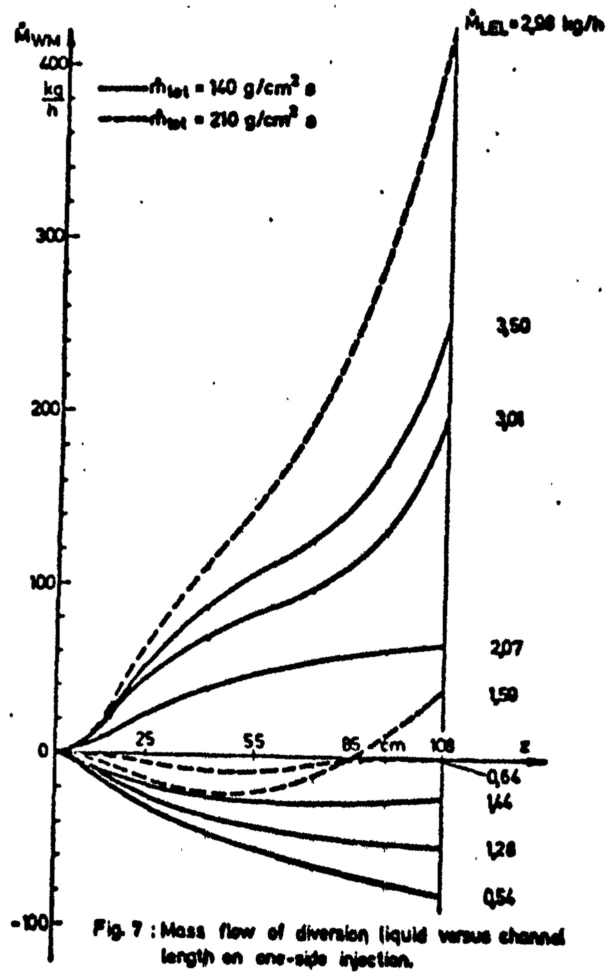


Fig. 7 : Mass flow of diversion liquid versus channel length on one-side injection.

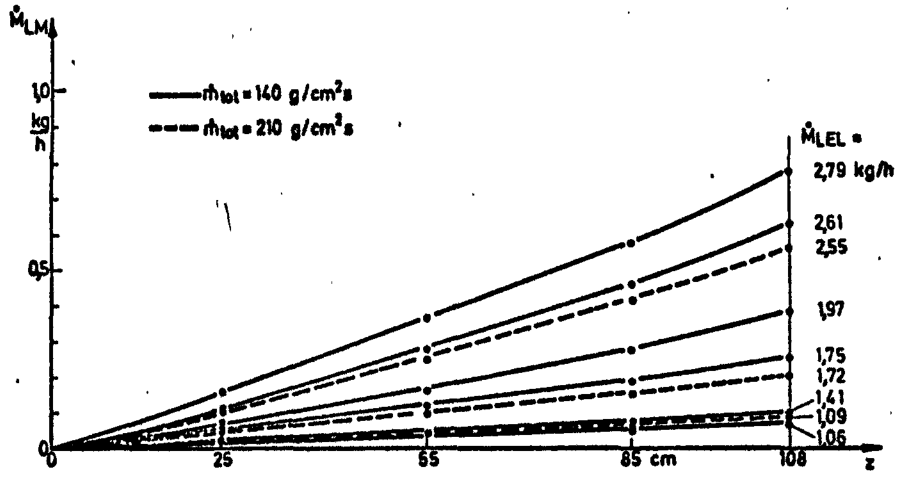


Fig. 8 : Mass flow of diversion gas versus channel lenght on both-sides injection, $\dot{M}_{LER}$ = 0,49 kg/h

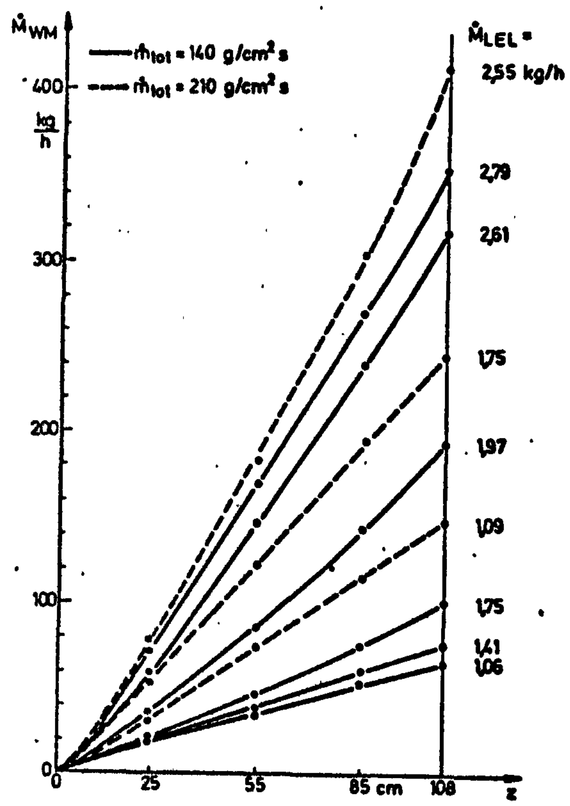


Fig. 9 : Mass flow of diversion liquid versus channel lenght on both-sides injection, $\dot{M}_{LER}$ = 0,49 kg/h

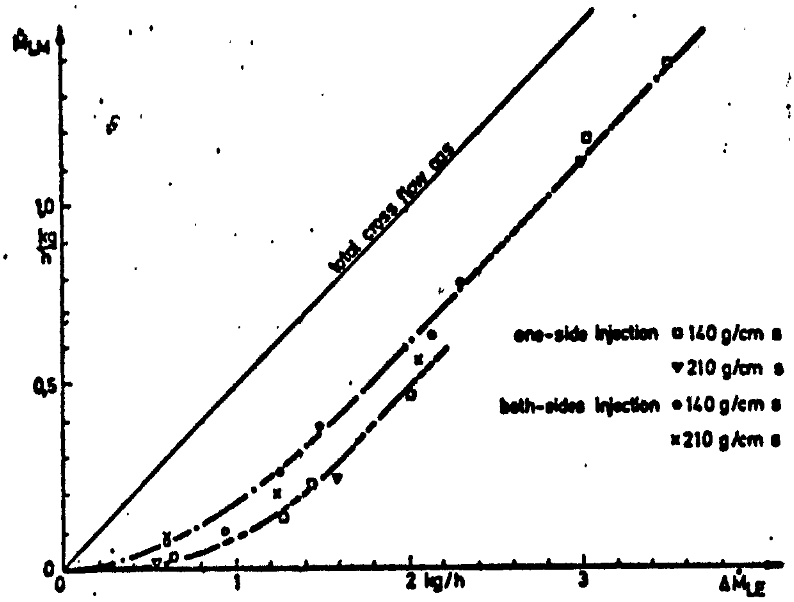


Fig. 10 : Mass flow of diversion gas versus difference mass flow of injected gas, $l_{mix} = 1{,}08$ m

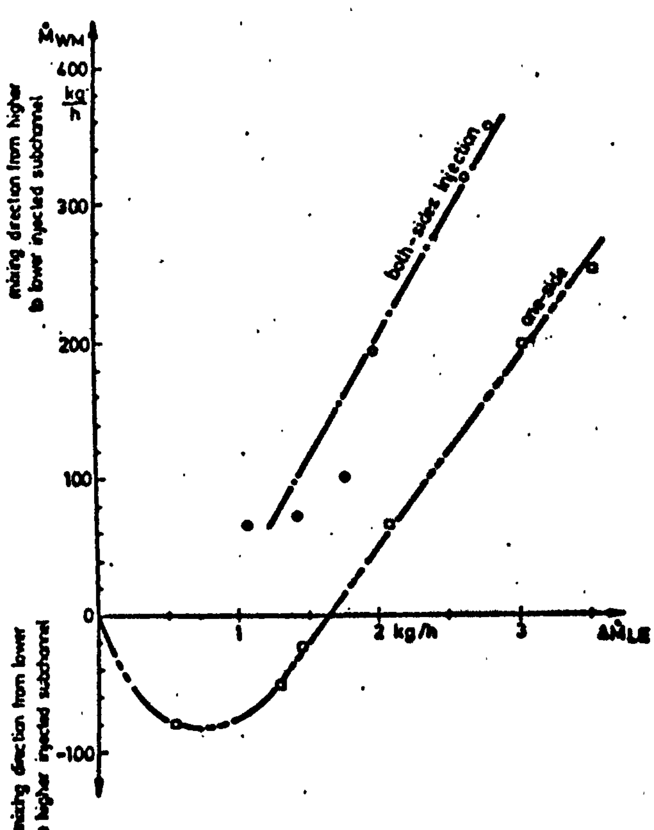


Fig. 11 : Mass flow of diversion liquid versus difference mass flow of injected gas, $\dot{m}_{tot} = 140$ g/cm²s, $l_{mix} = 1{,}08$ m

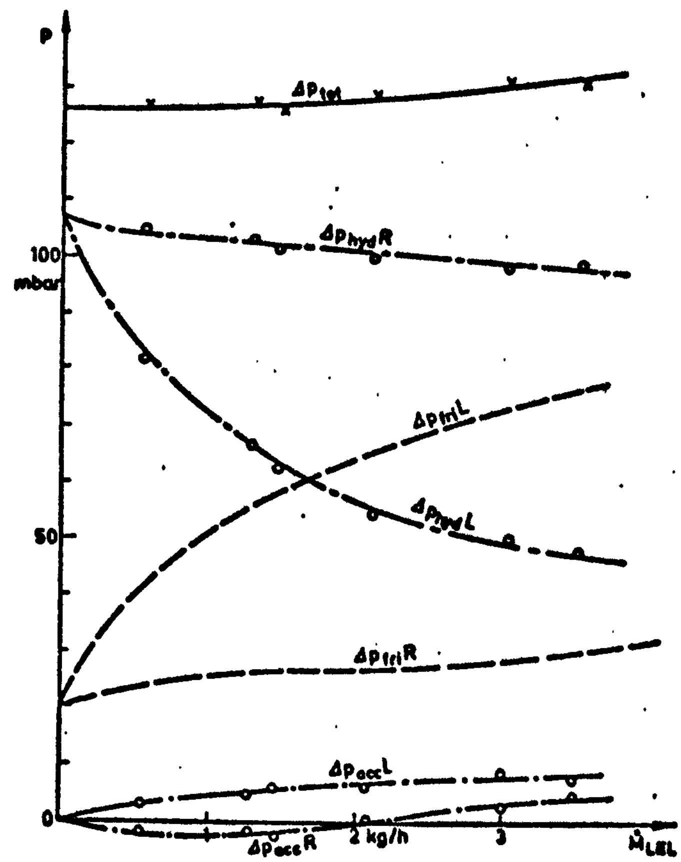


Fig. 12 : Pressure drop components versus mass flow of injected gas $\dot{m}_{tot} = 140$ g/cm²s, $l_{mix} = 1{,}08$ m

Classification: 1.1.2	
Title 1 (Original Language): Erstellung eines theoretischen Phasenseparationsmodells zur Erfassung des Wassermitrisses und Vergleich der theoretischen Vorhersage mit experimentellen Ergebnissen (RS 179 - I.1.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: T.U. Hannover
Title 2 (English): Development of a theoretical phase separation model with respect to liquid entrainment. Comparison of theoretical prediction with experimental data.	Project Leader: Professor Dr.-Ing. F. Mayinger
Initiated (Date): Aug. 1975 Status: Continuing Completed (Date): July 1978 Last Updating (Date): Dec. 1976	

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 R 12 (CF_2Cl_2).

2. Particular objectives

Within the activities a theoretical phase separation model has to develop leading to a better understanding of fluid behaviour during blowdown conditions. This model should be simple enough to be used in a computer code to predict the mass discharge, liquid level settlement and liquid entrainment from a horizontal surface with respect to the vapour generation due to pressure loss. 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 including a 4-rod bundle simulating the reactor core.

3. Research program

In the first step of this research work vapour is injected at constant pressure into a vessel filled up with liquid. While penetrating through the surface droplets are entrained by the upstreaming vapour. In the next steps this phase separation and entrainment process has to be investigated during blowdown conditions. For these tests the vessel model has to be connected with a low pressure tank. In the third step a second test section with a heated 4-rod bundle will be constructed to investigate the phase interaction and separation process in the core under reactor similar geometrical conditions.

4. Experimental facilities, Computer Codes

For the experimental investigations a 28:1 scaled down model of a reactor pressure vessel with a supporting loop for steady state experiments and a low pressure tank for blowdown tests was constructed as already reported in RS 179/V75. By aid of some high speed cinematographic devices optical investigations were performed of the phase separation and the droplet entrainment mechanism at steady state and in the first blowdown period. By this it was possible to gain detailed information about the droplet entrainment by counting and planimetering the recorded droplets from the pictures. From a single photo liquid hold-up was got as a function of the height above the surface. Comparing two following high speed photos it is possible to calculate the drop velocities and to determine droplet trajectories. For the first blowdown period the fluid behaviour could be detected and the flashing process could be studied.

Additionally void fraction behaviour depending on time and height of the test section may be investigated by aid of an installed γ-ray equipment as well as the total mass discharge can be recorded by a weight measurement using a force transducer. The system pressure and the liquid and vapour temperature transients can be determined by pressure gauges and thermocouples.

For the theoretical work a calculation model has to be developed to calculate the pressure loss combining the mass discharge at the break area with different separation models in the two-phase mixture or at the mixture surface.

5. Progress to Date

Within this reported period the following investigations and activities were conducted.

a) Steady state experiments
- Optical studies of the phase separation mechanism by vapour injection into saturated liquid.
- Optical investigations of liquid entrainment by vapour injection depending on the initial liquid level, injected vapour rate and system pressure.

b) Blowdown tests
- Investigations of the pressure transient, void fraction, temperature and dynamic fluid behaviour depending on the initial liquid level and the cross section area of the break.

c) Development of a calculation model to verify pressure decrease and mass discharge by aid of some different phase separation models and theories for critical outflow.
- Recalculation of water data from a Battelle Institute Frankfurt blowdown test.
- Calculation and comparison of pressure loss data with own experimental freon test results.

6. Results

In the first steady state experiments the phase separation mechanism was investigated for the very simple case of vapour injection into saturated liquid. Fig. 1 shows some essential scenes of a high speed movie taken with 4000 frames per second.

The upstreaming bubble reaches the surface, penetrates through the interface and forms an amounting jet of liquid. From the top droplets of different diameters are ejected and carried away by the upstreaming vapour.

A more detailed quantitative investigation has been carried out for steady state injection tests with regard to some parameters of expected main influence like the initial liquid level, the injected vapour rate and the system pressure.

Fig. 2 shows the influence of a varied liquid level. An increase of the liquid level from 6 mm via 21 mm to 36 mm above the injector nozzle corresponding to a variation of about 150 % gives a smaller dispersion in droplet entrainment. The drop diameters become larger and the amount of the compact liquid jet is greater. The maximum height of the droplet trajectories decreases and the liquid surface is stimulated to stronger oszillations. A quantitative investigation of the liquid hold-up due to droplet entrainment performed at different distances from the surface gives for the highest liquid level the highest hold-up, decreasing very quickly within the distance from the surface. For the lowest level the liquid hold-up is more uniform but lower within height, showing more droplets in greater distances than for higher liquid levels, see fig. 3.

A change of the injected vapour rate from 9,06 l/sm^2 via 11,33 l/sm^2 to 13,6 l/sm^2 i.e. a variation range of about 50 %, does not strongly influence the volumetric entrainment rate, but penetrating through the liquid surface the upstreaming vapour carried away bulks of droplets amounting higher with increasing injection rate. The quantity of liquid entrainment becomes more violent at henced injection rates, but the statistic drop diameter becomes smaller.

A variation of the system pressure in a range of 50 % from p = 5,33 bar to p = 7,61 bar gives a higher entrainment rate at lower pressure level. With smaller system pressure the density ratio between liquid and vapour increases leading to a larger bubble size in the entering vapour jet. This induces a larger momentum exchange area at the boundary layers causing a larger entrainment rate.

These steady state experiments could only give some information about the phase separation behaviour above a quiet liquid surface and in a range of only low droplet entrainment. In the real case of a reactor blowdown more complicated and complex mechanisms are expected interfering together, so the thermohydraulic fluid behaviour was investigated under pressure transients with regard to different inital liquid levels and different cross section areas of the break.

Fig. 4 shows some characteristic scenes from the onset of a blowdown taken from a high speed movie. Before the test the liquid was at saturation conditions. The break corresponds to the whole outlet tube cross section area.

The flashing effect starts at the liquid surface and at the boiling nuclei, formed by the tips of the nozzles (used for steady state vapour injection tests) and at some roughness at the bottom of the vessel model. Both vaporization fronts adhere together and form a homogeneous mixture in the vessel, climbing up to the break. During this flashing period no more optical investigation method yield to quantitative results because of the many phase boundaries, reflecting the light. Therefore the time depending void fraction behaviour was recorded in many different heights and just before the break by use of the γ-ray attenuation method. Furthermore the pressure decrease was measured by pressure gauges.

In fig. 5 some recorded pressure curves are presented. In the first 200 msec the pressure decreases rapidly because of the escaping vapour. This causes a violent flashing in the vessel implied by the onset of evaporation. The outflow to the break is limited by the critical velocity, so an increase of vapour generation yields to a pressure increase in the vessel until at the second maximum the mass of the generated vapour and the escaping fluid is equal. From this point the mass discharge is bordered by the critical homogeneous outflow. Depending on the initial liquid level, a smaller pressure increase could be measured with decreasing initial liquid level.

A variation in the break cross section area gives no different qualitative result for the pressure curves, but the time depending values for example the second maximum is reached after a longer time for a smaller break.

The maximum liquid entrainment measured before the break decreases with increasing initial liquid level, see fig. 6, while a smaller break of only 64 % of the total outlet tube cross section area yields to values of about 10 - 15 % lower ones as for the total break, because of the lower momentum exchange of the two phases in the mixture.

Starting with this knowledge of the fluid behaviour during the blowdown given by the optical results, a calculation model could be developed, using some phase separation models mentioned in RS 179/V75 together with a theory of critical outflow like Moody or Köberlein /1/. Combining the models of a total phase-separation, that one following the Relap Code and that one for a homogeneous mixture together with Moody's theory, experimental water data from the Battelle Institute Frankfurt were recalculated, see fig. 7. Compared to own freon tests good agreement could be given using the same separation models but together with Köberlein's theory, see fig. 5.

7. Next steps

For the prediction of the time depending liquid entrainment and phase separation behaviour it is necessary to measure the total mass discharge. Therefore a force transducer has been installed to weight the total test section. A comparison of measured mass discharge together with the measured and calculated data of void fraction will be used to develope and to test the calculation model. From all these informations it will be tryed to give a prediction of the time depending liquid entrainment and the phase separation behaviour.

8. Relation with other projects

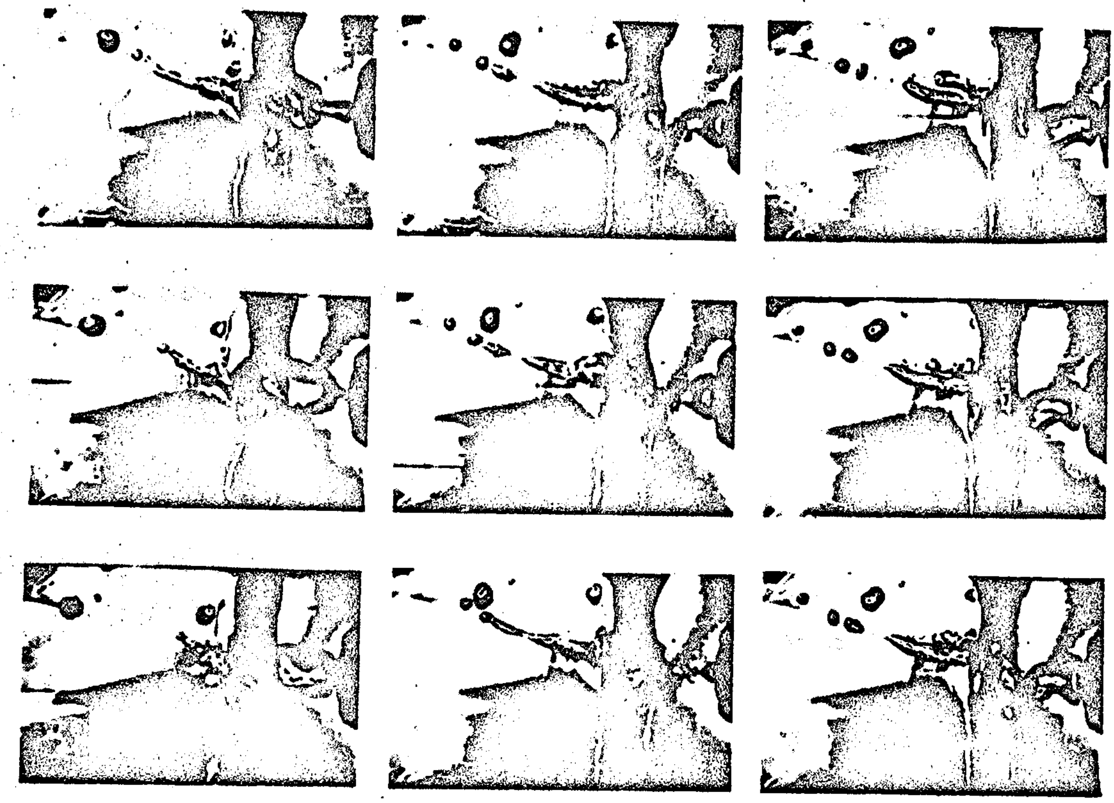
See RS 179/V75

9. References

/1/ Köberlein, K.: Die verzögerte Einstellung des thermodynamischen Gleichgewichts als Grundlage für die Druckwellenausbreitung in der Zweiphasenströmung. Ber.Nr. MRR 106, April 1972, LRA Garching

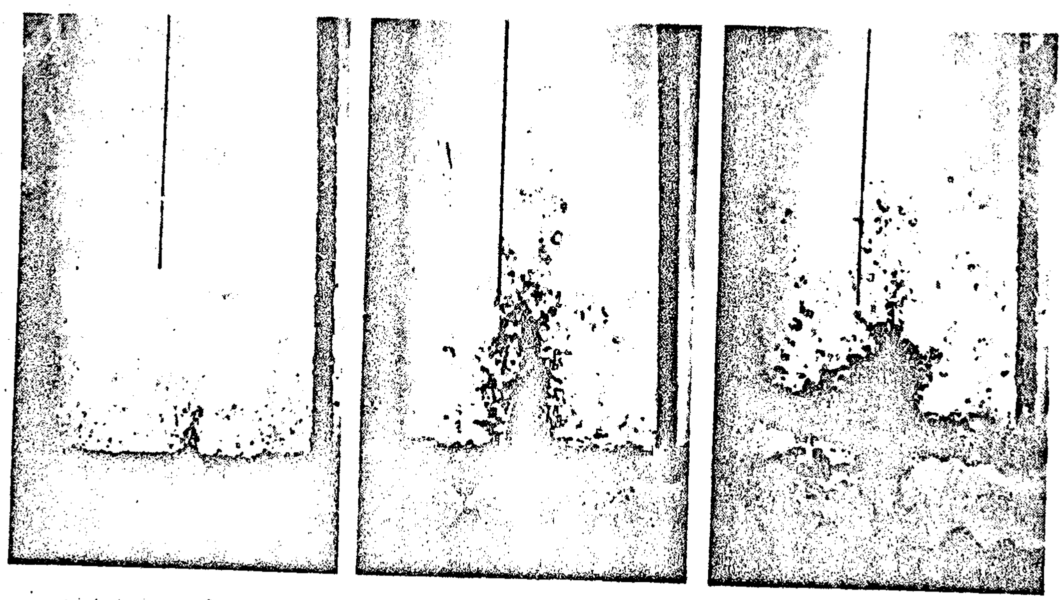
10. Degree of availability of the Reports

LRA Garching



Phase separation mechanism and droplet entrainment

Fig. 1



liquid level z = 6 mm z = 21 mm z = 36 mm

Influence of varied liquid level on phase separation and droplet entrainment behaviour

Fig. 2

120

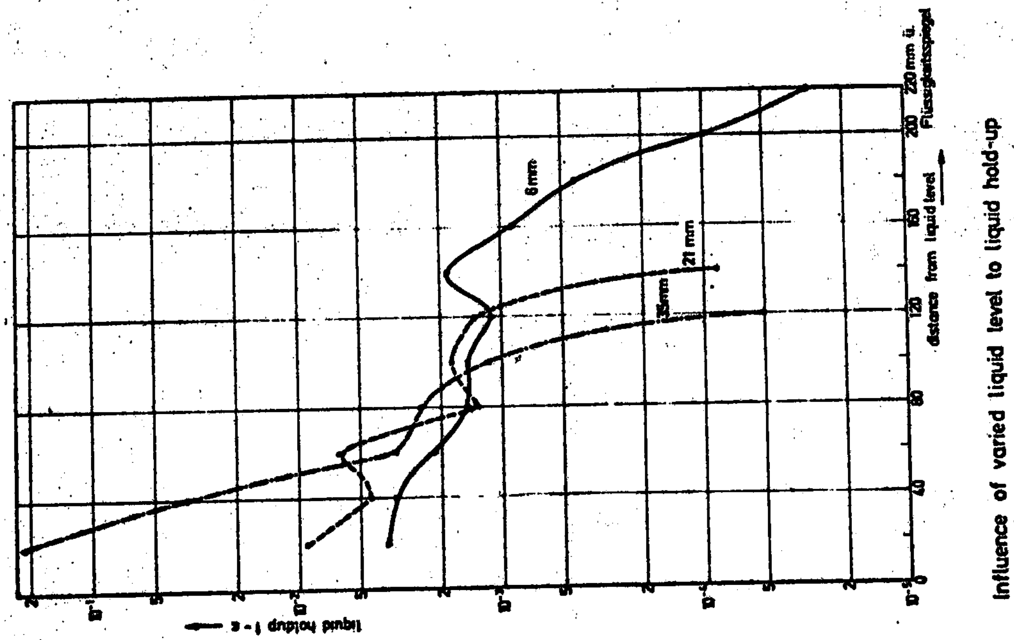


Fig. 3

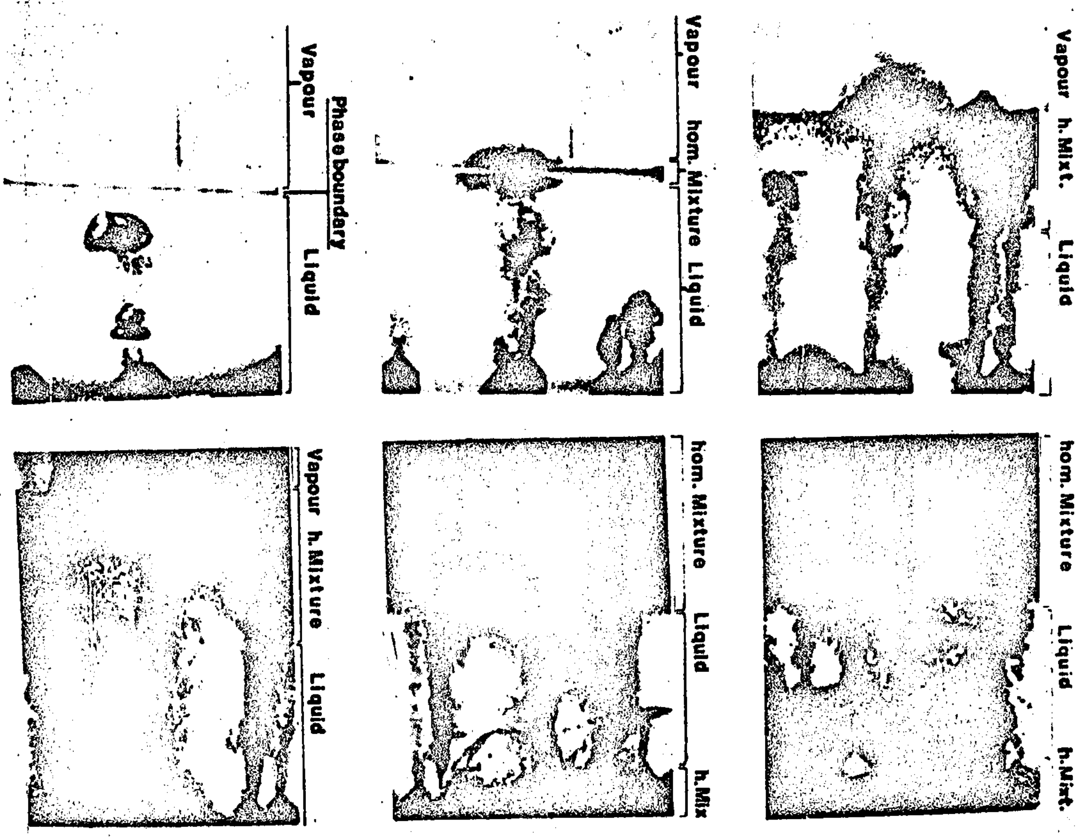


Fig. 4 Fluid behaviour during blowdown

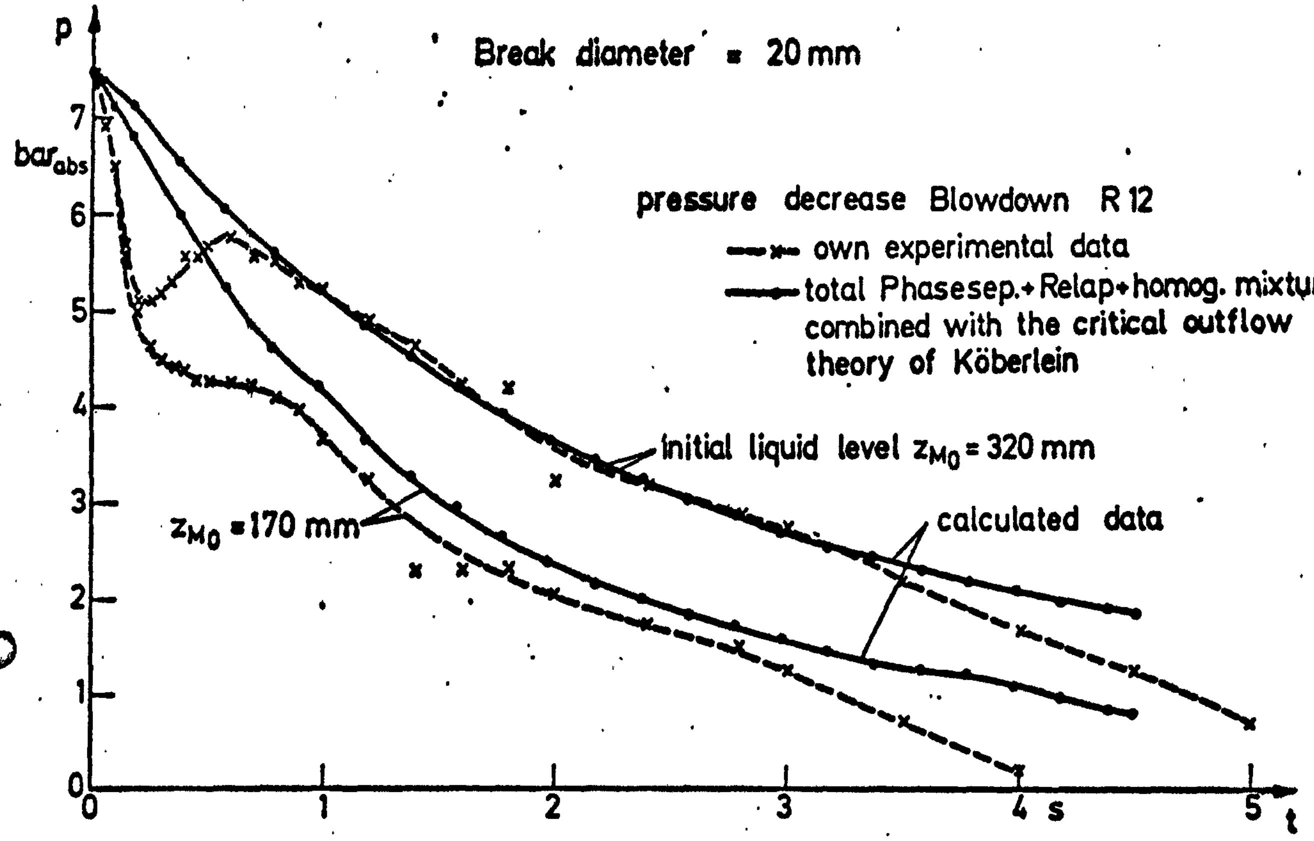


Fig.5 Vessel blowdown, 64% break area, initial liquid level 170, 320 mm, p_0=7,45 bar$_{abs}$
ϑ_0 = Saturation temperature

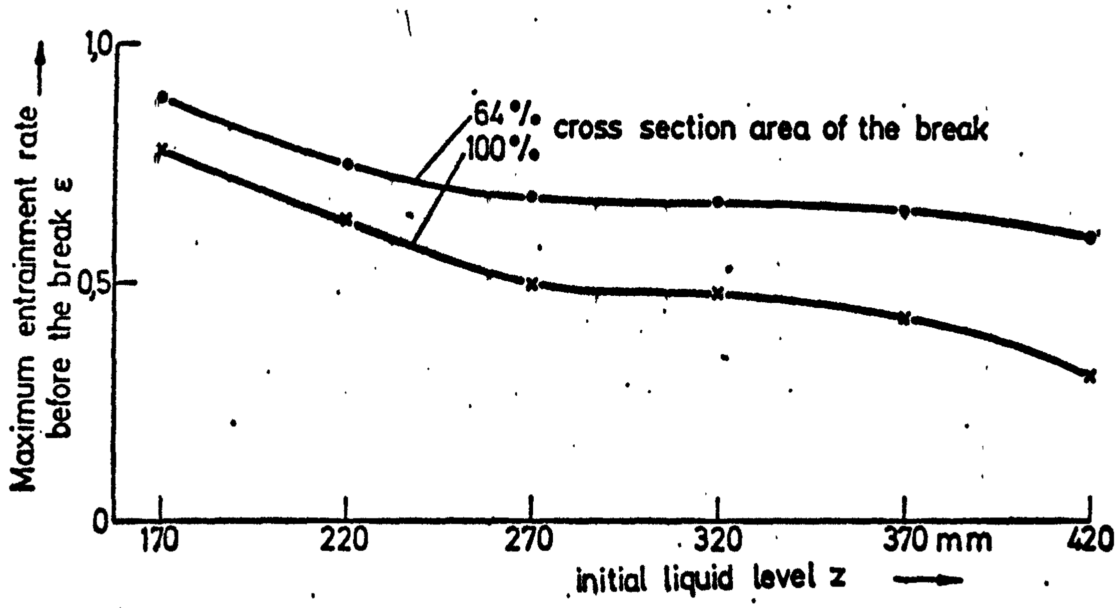


Fig. 6 Maximum entrainment rate as function of the initial liquid level and the cross section area of the break

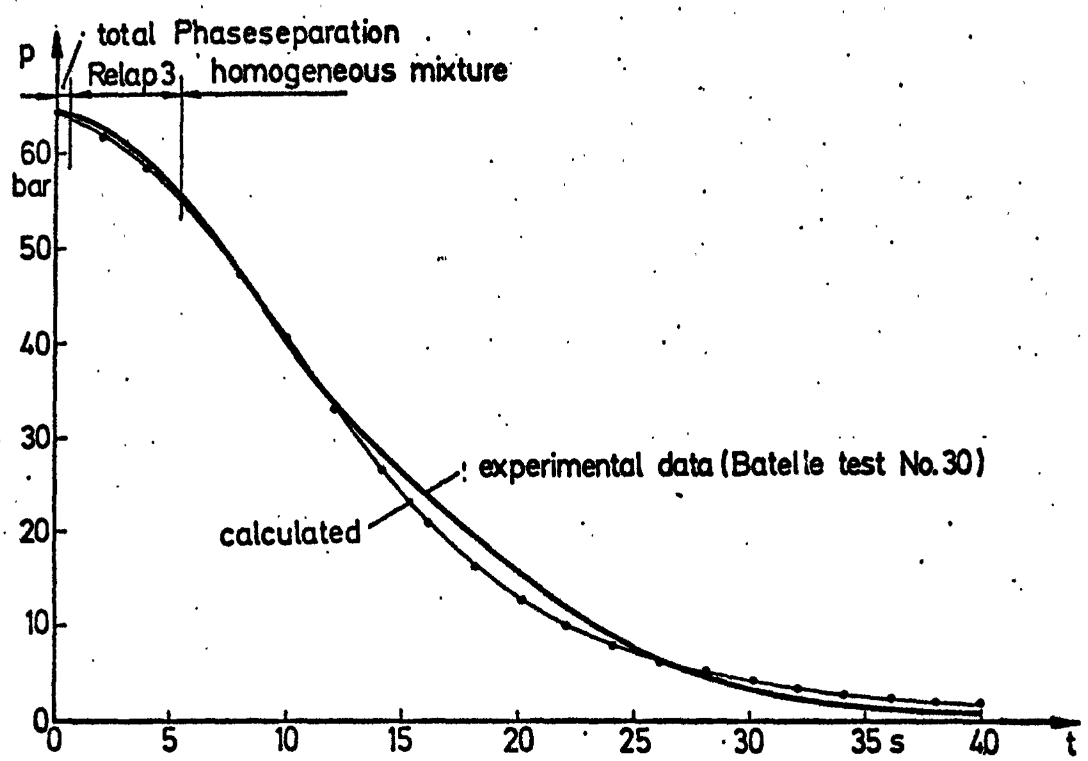


Fig.7 Comparison of experimental and calculated pressure decrease curves of a vessel blowdown

Classification: 1.1.2	
Title 1 (Original Language): Entwicklung von Meßverfahren zur Bestimmung transienter Massenströme (Dampf/Wasser) durch Signalkorrelation RS 135 - I.1.1, Jahresbericht A 1976	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: IKT, TU Berlin
Title 2 (English): Development of methods for measuring transient two-phase mass flows (steam/water) by signal correlation	**Project Leader:** Prof. U. Wesser
Initiated (Date): Jan. 1, 1975 Status: continuing Completed (Date): Dec. 31, 1977 Last Updating (Date): Dec. 31, 1976	

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 transit time is determined by using cross correlation techniques, while the temperature fluctuations are detected by thermocouples.

3. Experimental facilities and program

3.1 Experimental facilities

There is a one- and two-phase-flow water loop for low fluid velocities up to 5 m/s which is used to study and calibrate the measurement ap-

paratus.
Beyond that a small blowdown facility (50 bar) was built up to run the measurement assembly under real blow-down conditions.

3.2 Research Program

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

4. Project Status

4.1 Progress to Date

4.11 Density Measurements

To verify the theoretical analysis concerning density measurement methods with x-ray attenuation, experiments with low energy x-rays (= 30 keV) have been extended on air-water mixtures using scattered radiation to get the mean value by calculation in case of inhomogeneous density distributions.

4.12 Velocity measurements

Transient fluid velocity measurements have been performed at a two phase loop with velocities between 1 and 5 m/s and further velocity gradients of 1 m/s^2 at a blow-down facility special signal conditioners developed for this measurement technique.
The off-line cross correlation of the temperature signals was realized on a time series analyzer.

4.2 Essential Results

4.21 Density Measurements

It is supposed that in case of blow-down experiments the density distribution across the flow area is approximately homogeneous so that the application of one x-ray beam (if necessary with use of the scattered radiation) should yield sufficient accuracy.

4.22 Velocity Measurements

The cross-correlation technique applied to signals due to natural temperature fluctuations measured at two fixed points along the flow path proves to be a suited method for determining fluid velocities in pipes even in case of two-phase flows and transient conditions. At the first blow-down test series two-phase flow velocities in the range of 50 to 80 m/s could be detected. The void fraction was about 0.7.

5. Next Steps

The blow-down tests will be continued with higher pressures. Different configurations of the measurement assembly (variation of the distances between thermocouples, change of densitometer geometry) will be investigated.

The influence of increasing the corner frequency of the thermocouples by electonical means on the results of time series analysation shall be checked.

6. Relations with other Projects

It 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).

7. References

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

/2/ Lübbesmeyer, D.; Ulber, M.:
Messung von Fluidgeschwindigkeitstransienten auf der Basis von Kreuzkorrelationsanalysen
Vortrag: KTG-Fachtagung "Experimentiertechnik auf dem Gebiet der Reaktor-Fluiddynamik", Berlin, 10.-12.3.1976

8. Degree of availability

/1/ free

Schedule of the Research Program

Activity

X-Densitometer

(I) Theoretical analysis of different densitometer principles

(II) Design optimization

(III) Calibration and test of a design using a scattered beam

Velocity measurement with thermocouples

(I) Development of electrical circuitry for processing thermocouple signals

(II) Mechanical layout of the TC attachments

(III) Transient velocity measurements on a two -phase loop (stationary)

(IV) Experiments on a two-phase loop at GfK in Karlsruhe

(V) Experiments with groups of thermocouples for average velocity measurements

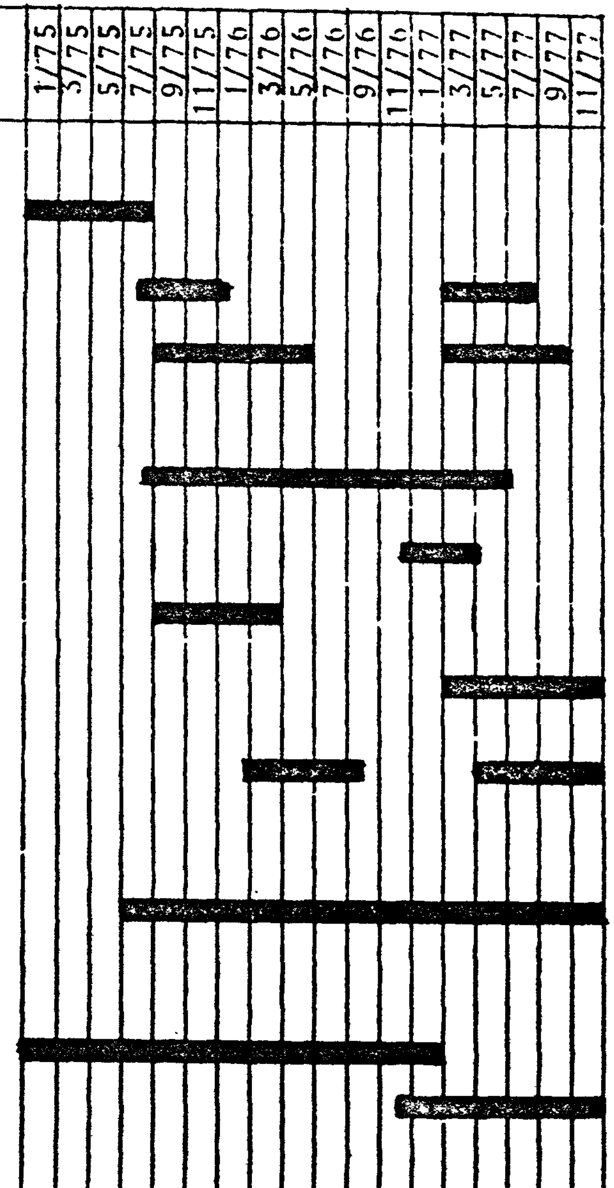
Time series analysator

(I) Software for automatic computation and plotting of the transient two-phase mass flow versus time

Blow-down test loop

(I) Planning and built up phase

(II) Transient two-phase experiments on the blow-down loop



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

General Aim

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

Particular Objectives

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

Experimental Facilities/Research Program

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

Project Status

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

Project Status/Essential Results

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

Next Steps

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

Relation with Other Projects

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

Reference Documents/Degree of Availability

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

Classification: 1.1.2	
Title 1 (Original Language): Gemeinsamer Versuchsstand zum Testen und Kalibrieren verschiedener Zweiphasen-Massenstrommeßverfahren (RS 145 (PNS 4215) - I.1.1 , Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT /GfK/PNS **ORGANIZATION:** GfK, Karlsruhe Projekt Nukleare Sicherheit
Title 2 (English): Joint Test Rig for Tests and Calibration of Different Methods of Two-Phase Mass Flow Measurement	**Project Leader:** J. Reimann
Initiated (Date): 1.10.1974 **Completed (Date):** 1978 **Status:** continuing **Last Updating (Date):** December 1976	

1. General Aim

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

2. Particular Objectives

Besides the knowledge of the mass flow-rate and the quality, the knowledge of the flow pattern and the apparent density is of special interest.

3. Research Program

Different measuring methods that are being developed in other institutions are to be tested and calibrated in steady-state steam-water flow, air-water flow and transient steam-water flow. Parameters are for steady-state flows: mass-flow-rate, quality, pressure; for transient flow: pressure at the beginning, critical flow area.

4. Experimental Facilities

For the comprehensive tests, the following facilities must be available: a test loop for steady-state air-water flow, a test rig for transient steam-water flow.

The test section and the position of the inserted measuring devices is not changed when changing the operating conditions. The flow regime is detected by a local measuring probe (impedance probe), for the apparent density a multiple γ-beam densitometer is developed.

5. Progress to Date

The building up of the test loop for steady-state steam-water flow has been finished. The measure- and control-system (pressure, temperature, mass flow measurement) has been checked at working conditions up to 150 at, different mass-flow and qualities.

Some tests have been made with a special visual inspection unit (sapphire window) for flow detection at pressures up to 150 at by means of a high-speed camera. Simultaneously, signals obtained from the impedance probe have been stored on magnetic tape.

The extension of the test loop for air-water mixtures began in summer, the first runs are now being performed.

The essential parts of the multiple γ-beam-densitometer were built.

6. Results

The planned maximal values of the mass flow rate as a function of quality and pressure are essentially reached. It is expected for the future experiments that between 5 and 10 stable operating conditions per day will be attainable.
The development of the impedance probe for the use in steam-water flows at high pressures was successful. The probe signals could easily be interpreted even at conditions where the high speed photography was no longer interpretable. The first experiments for building up a flow chart for horizontal steam-water flows have been carried out.

7. Next Steps

The construction of the densitometer will be finished, and testing of the electronic equipment will start.
The testof the first mass flow measuring methods at steady-state steam-water and air-water flow will begin.
The design of the test loop for transient conditions will be finished and the pressure vessel will be ordered.

8. Relation with Other Projects

RS 109; 135; 146; 147; 188; PNS 4236

9. References

Reports in the series IRS-Forschungsberichte
IRS - F - 29 (Annual report 1975)
IRS - F - 30
IRS - F - 31
IRS - F - 33

Report KFK 2262 (1976) P. 183

Classification: 1.1.2	
Title 1 (Original Language): Entwicklung eines Radionuklidverfahrens zur Massenstrommessung in instationären Mehrphasenströmungen (RS 146 (PNS 4214) - I.1.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT/GfK/PNS ORGANIZATION: PNS GfK Karlsruhe
Title 2 (English): Development of a Radionuclide Method of Mass Flow Measurement in Non-Steady State Multiphase Flows	**Project Leader:** R. Löffel
Initiated (Date): 1974 **Completed (Date):** 1978 **Status:** Continuing **Last Updating (Date):** Dec. 1976	

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 250 °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 densities.

3. Research Program

The experimental program consists of:
- measurement of gas velocity in the blowdown channel during the MARVIKEN/MXIICRT experiments
- test of the radionuclide method in the discharge pipe of the MARVIKEN reactor
- final 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-experiments (RS 123).

4. Experimental Facilities

Installation of a radiotracer gas velocity measurement device at one of the four blowdown channels and of a radiotracer two-phase velocity measurement device at the discharge pipe of the Marviken reactor was terminated in January 1976. In June a Gamma densitometer was additionally installed at the discharge pipe (cf. Figs. 1 and 2). This allowed to evaluate the void factor.

The radiotracer measurement device for the performance of the experiments at the joint test rig (RS 145) was completed.

5. Progress to Date

The Marviken II blowdown tests lasted from February until October and included 9 blowdowns. Besides the gas velocity measurements agreed by contract with the MXII-CRT Project, velocity and density measurements were performed in the discharge pipe as preliminary tests of HDR experiments.

6. Results

In the blowdown channels a gas velocity of 0 to 100 m/s was measured. Comparison of the gas phase velocity (measured with the radiotracer method developed by LIT) with the velocity of the liquid phase (measured by the infrared method of Institut für Thermische Strömungsmaschinen der Universität Karlsruhe) shows good agreement of the results. This demonstrates that there was practically no slip between the gas and liquid phases.

First results are available of the velocity and density measurements performed in the discharge pipe:
- the velocity ranges from 0 to 40 m/s
- values from 0 to 0.5 were determined for the void factor

7. Next Steps

In 1977 the method will undergo final testing under conditions resembling blowdown. The tests will be performed 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 1977 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

RS 145 (PNS 4215) Joint Test Rig for Tests and Calibration of Different Methods 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)
VDI-Berichte Nr. 254 (1976)(German)

Reports in the series IRS-FORSCHUNGSBERICHTE
Annual report (1975): IRS - F - 29 (English) .
Quarterly reports : IRS - F - 30 (German)
IRS - F - 31 (German)
IRS - F - 32 (German)

10. Degree of Availability of the Reports

Unrestricted distribution.

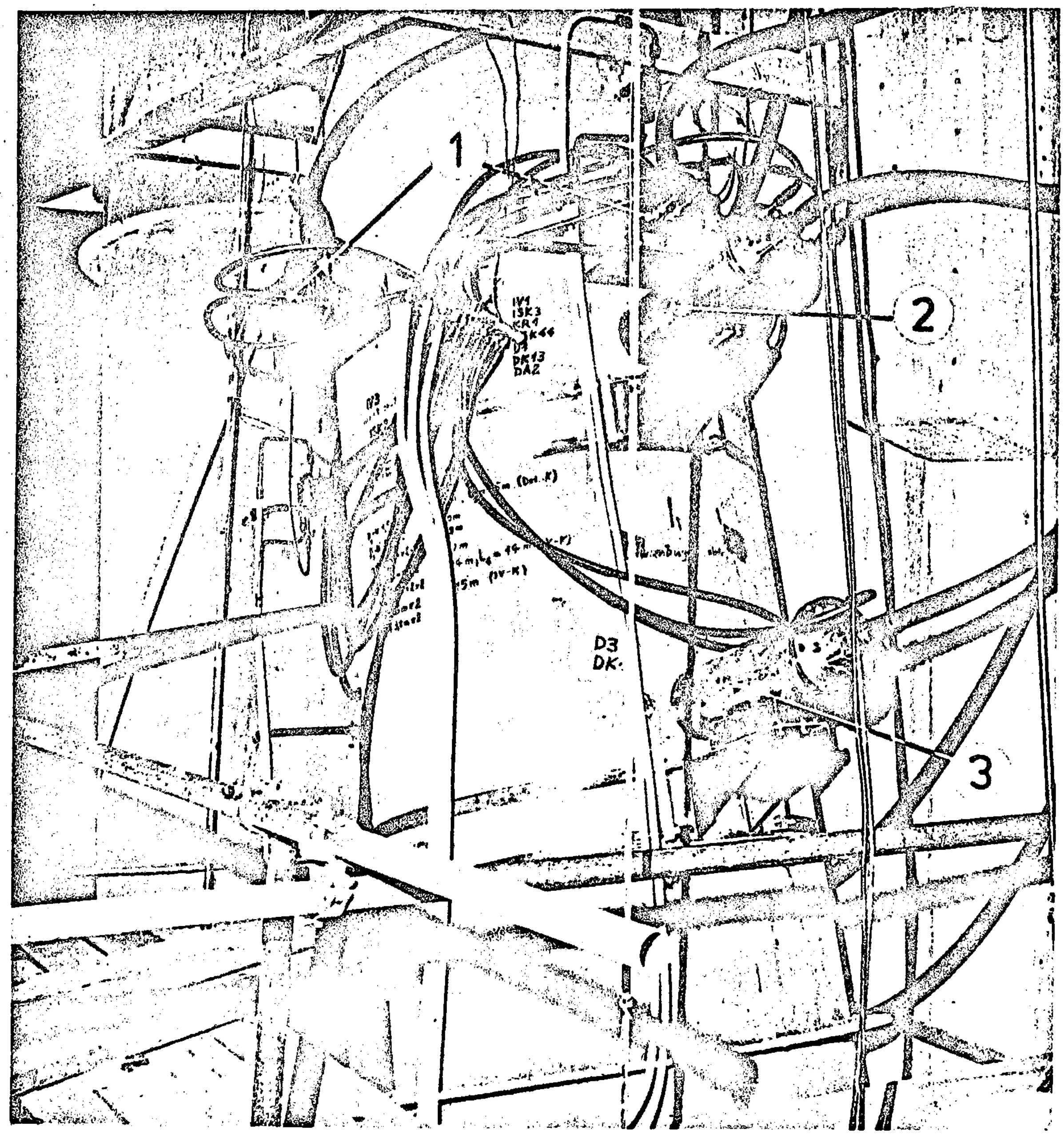


Fig. 1: Velocity measuring device at one of the four blowdown channels of the MARVIKEN-reactor

1 injection valves
2 γ-detector (1st cross section)
3 γ-detector (2nd cross section)

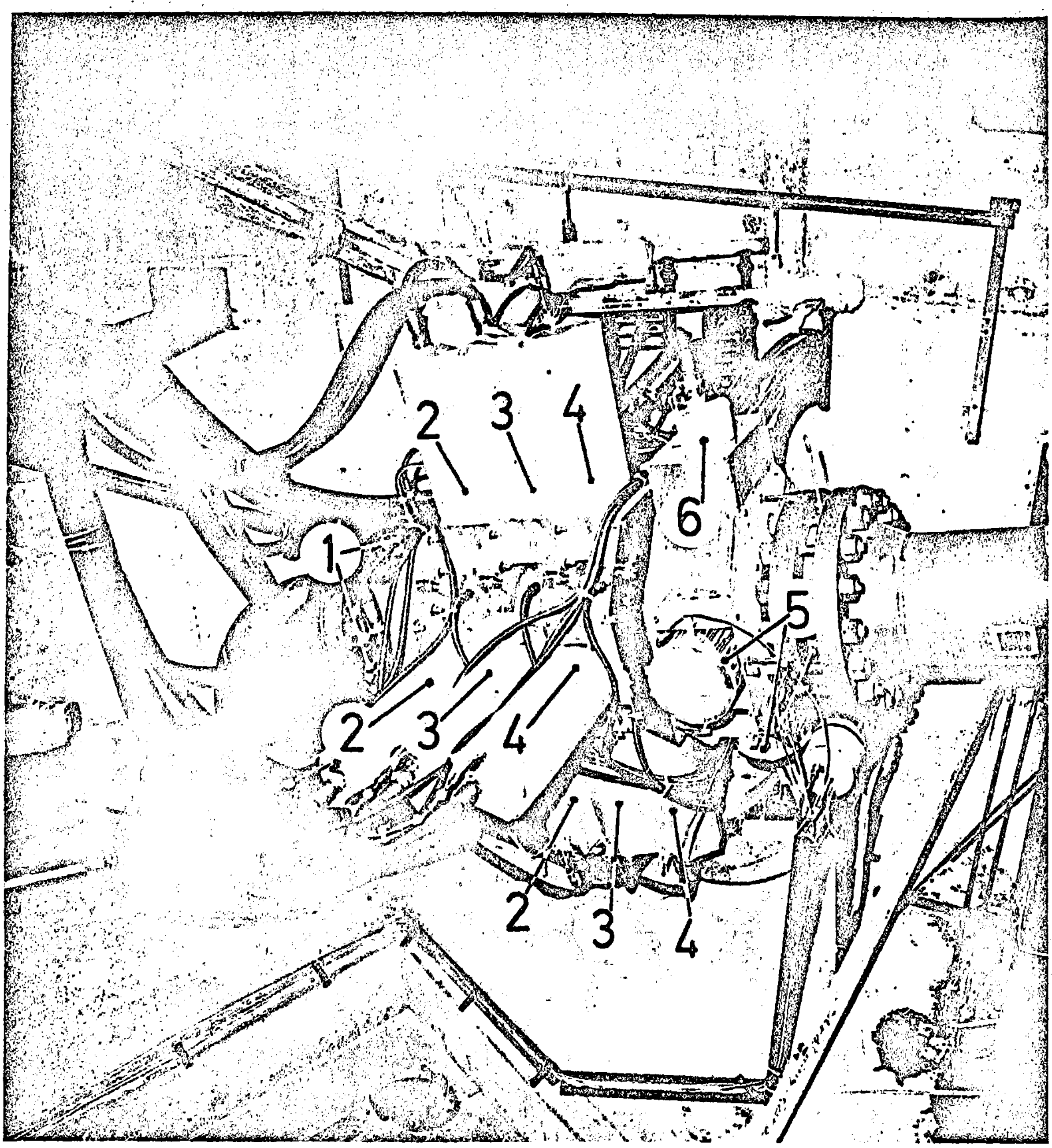


Fig. 2: Radionuclide mass flow measuring device at the discharge pipe of the MARVIKEN-reactor

1 injection valves
2 γ-detector at the 1st cross section (velocity)
3 γ-detector at the 2nd cross section (velocity)
4 γ-detector at the 3rd cross section (velocity)
5 γ-source (densitometer)
6 γ-detector (densitometer)

Classification: 1.1.2	
Title 1 (Original Language): Weiterentwicklung eines Drag-body für die Massenstrommessung bei Blowdown-Untersuchungen im Forschungsvorhaben RS 109 (RS 147 - I.1.1., Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Battelle-Institut e.V., Ffm.
Title 2 (English): Improvement of a Drag Body for the Mass Flow Measurement in Blowdown Experiments of the Research Project RS 109	Project Leader: G. Hampel
Initiated (Date): September 15, 1974 Status: Continuing Completed (Date): June 30, 1977 Last Updating (Date): December 31, 1976	

1. General Aim

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

2. Particular Objectives

3. Research Program

3.1. Theoretical investigations of the dynamic behavior 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 measuring signal.

3.7. Experimental investigation of the behavior of the drag body in steady 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 program "DRAGBODY", which is based on the Finite-Element Method.

5. Progress to Date

Ad 3.3: One special drag body for bidirectional measurements was developed for application in the loop ND 45, in the lower plenum of the test rig RS 109, and in the test loop RS 145. The drag bodies were manufactured, calibrated and submitted to a cold pressure test and a temperature test. The frequency curves were calculated.

Ad 3.5: As it is not possible to establish by experiment the dependence of the drag coefficient on the flow velocity and the density of a two-phase mixture within the operating range of the drag body because there are no reference methods, an attempt was made at determining the dependence theoretically: It was assumed that a two-dimensional steady homogeneous flow exists and that the compressibility of this flow formally obeys the pressure density relation for ideal gases. Another assumption was that discontinuity surfaces form at the edges of the drag plate (normal to the flow). The flow velocity must not anywhere exceed the velocity of sound.

6. Results

Ad 3.3: The static calibration under environmental conditions showed a linearity of $\pm$ 1 %. The hysteres is in reversed deflection direction is negligible. The high-temperature test showed a zero drift of the drag body (drift of the amplifier included) of about 0.024 %/^{o}C.

Ad 3.5: If the equation characterising the two-phase mixture is inserted in Bernoulli's equation, an equation results which implicitly indicates the relation between the density components of a two-phase flow and its velocity. This equation does, however, not give an explicit representation of the functional relationship between mean density ρ and velocity, because it is transcendental with respect to the variable ρ and thus cannot be solved analytically with respect to ρ. A functional approximation by means of expansion in a series is incompatible with the dimensioning of the equation. The effect of the deviations due to the approximation on the final result can hardly be estimated with a reasonable amount of effort.

In order to avoid uncertainties of measurement, it must therefore be ensured that the velocity of flow at the measuring point, i.e. at the drag body, remains sufficiently far below the velocity of sound.

7. Next Steps

The investigations covered by Section 3.7 will be performed in the test rig RS 145. A report on the results of the investigations will be written.

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

1. General aim

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

2. Particular objectives

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

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

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

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

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

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

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

4. Project status

4.1 Progress to date

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

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

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

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

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

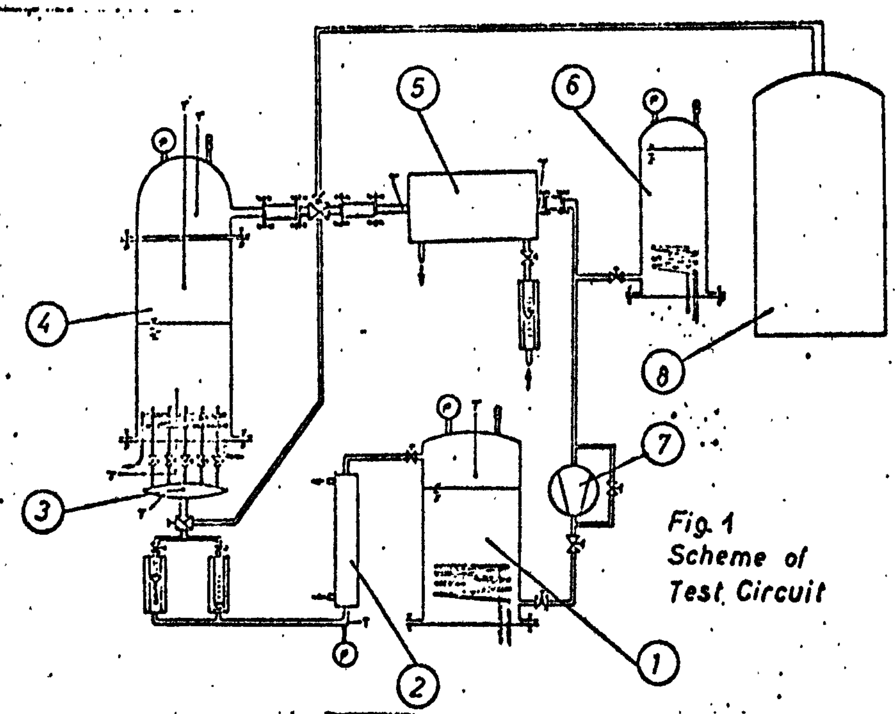

Fig. 1
Scheme of
Test Circuit

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

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

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

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

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

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

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

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

4.2 Results

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

5. Next steps

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

6. Relation with other projects

RS 48, RS 48/1

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

RS 163

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

RS 147

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

7. Reference documents

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

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

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

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

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/5/ F.J. Moody: Maximum flowrate of a single component two-phase mixture; Journal of Heat Transfer, Feb. 1965, Transactions of the ASME

/6/ Quarterly reports in the series IRS-Forschungsberichte

8. Degree of availability

The IRS-Forschungsberichte are available by IRS.

Classification: 1.1.2	
Title 1 (Original Language): Untersuchungen über das Verhalten von Hauptkühlmittelpumpen bei Kühlmittelverluststörfällen Phase A (RS 111 - I.1.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Investigations of the Behaviour of Main Coolant Pumps under MCA Conditions (Phase A)	Project Leader: Dr. Riedle
Initiated (Date): 1. 9. 74 — Completed (Date): 31. 12. 77 Status: Continuing — Last Updating (Date): 31. 12. 76	

1. General Aim

During blowdown the flowrate through the core and the temperature of the fuel rods depends on the behaviour of the main coolant pumps. The behaviour of the pumps under two phase flow conditions will be studied, in order to improve the theoretical and experimental knowledge and to develop and improve models on two-phase behaviour used in the safety analysis.

2. Particular Objectives

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

3. Research Program and 4. Experimental Facilities

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

A testloop at C-E will be rebuilt in order to measure the two-phase pump characteristic for the interesting parameter variations of pressure, flowrate and vapour content. With the two model pumps about 390 stationary points will be run and recorded in the two-phase region. 10 transient tests, where a loop blowdown will verify the applicability of the steady state data to transient LOCA calculations, will be carried out.

5. Progress to Date

During the start-up of the testloop without testpump the heat losses on the instrumentation tubes,the hydraulic stability, the influence of slug-flow conditions and the control of the loop were investigated. The maximum mass flow in the loop and the time, to reach steady state conditions were determined.

The instrumentation was completed. The data evaluation was improved. The code CEFLASH-4A was adapted to the testloop for the calculation of blowdown process data.

First tests were run with the C-E-model pumps in order to determine the friction. During March 1976 a problem with the speed controller and a sudden oil loss were observed. The defect parts of the pump were sent back to the manufacturer.

Inspite of the defect pump several checks were carried out for the testloop control, instrumentation, thermohydraulic behaviour and the case of the pump backstreaming.

New difficulties arose when the pump was repaired with the pump bearings, the oil lubrication was not sufficient under variable load. The pump casing had to be remodelled. As a consequence of the new pump defect several other components were demaged and had to be repaired.

The start-up tests were evaluated. The test program was modified because for some values of the mass flow rate and void flow-instabilities by slug-flow had been observed.

For the ASTRÖ-model pumps several technical problems were discussed. The main components were ordered.

6. Results

Vibrations in the testloop were damped by different suspensions. The forces in the blowdown section were reduced about a factor of two by constructional modifications.

The defect of the Byron-Jackson pump was caused by a broken oil slinger ring. During the disassembly in addition a broken bolt was discovered, other bolts were damaged by corrosion. After the defective parts had been exchanged. 15 steady state and transient tests were carried out after which other material problems occurred. The axial bearing of the pump was overloaded. A new bearing had to be designed.

7. Next Steps

The test with the Byron-Jackson pump will be continued, the tests are planned to be finished in July 1977. After this the experiments with the ASTRÖ-pump will be conducted.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Classification: 1.1.2	
Title 1 (Original Language): Einfluß der DWR-Umwälzschleifen auf den Blowdown LOBI-Projekt (RS-1o9-I.1.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT-Bonn ORGANIZATION: C.E.C. EURATOM J.R.C. Ispra
Title 2 (English): Loop Blowdown Investigations (LOBI)-Project: Influence of PWR primary loops on blowdown	Project Leader: Dipl.-Ing. W. Riebold
Initiated (Date): January 1974 Completed (Date): Nov. 1977 (contract) Status: Continuing Last Updating (Date): Dec. 1976	

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 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-1o9/143-73-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 6o 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 broken "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 210° and 270° 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 85oo 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 RS 147 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 $o,15^{o}$ C at $3oo^{o}$ 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 $o,1^{o}$ C and $o,5^{o}$ 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

Classification: 1.1.2.	
Title 1 (Original Language): GKSS-Pumpenuntersuchungen (RS 144 - I.1.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GKSS, Geesthacht
Title 2 (english): GKSS-Pumptests for the Primary Pumps of RS 109	Project Leader: Katsaounis
Initiated (Date): 1.11.1974 Status: continuing Completed (Date): 1977/78 Last Updating (Date): December 1975	

1. General aim

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

2. Particular objectives

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

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

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

3.1 Experimental facilities

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

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

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

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

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

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

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

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

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

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

3.2 Research program

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

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

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

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

where N indices nominal working conditions. The quality x is varied in the limits $0 \leq x \leq 1$.
A test-series with sufficiently subcooled water is planned.

4.1 Progress to date

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

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

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

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

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

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

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

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

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

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

4.2 Essential Results

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

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

2) approach to velocity of sound,

3) limitation of power available,

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

Concerning:

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

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

5. Next steps

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

6. Relation with other projects

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

7. Reference documents

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

8. Degree of availability

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

Classification: 1.1.2	
Title 1 (Original Language): Untersuchung des thermohydraulischen Ungleichgewichts (RS 77 - I.1.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: CEC, Ispra
Title 2 (english): Investigation of the Thermodynamic Non-Equilibrium	Project Leader: G. Fritz, W. Riebold
Initiated (Date): 1.12.1972 Status: finished	Completed (Date): 31.12.1975 Last Updating (Date): December 1975

1. General aim

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

2. Particular objectives

The deviation from thermodynamic equilibrium between the phases caused by:

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

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

3. Experimental facilities and program

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

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

4. Project status

The experiments are finished.

5. Next steps

Completation of the final report.

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

Classification: 1.1.2	
Title 1 (Original Language): Untersuchung der stationären und instationären kritischen Heizflächenbelastung an Vielstabbündeln von Druck- und Siedewasserreaktoren mit Frigen als Modellflüssigkeit. (RS 64 - II. 1.3 Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GKSS / A
Title 2 (English): Investigation of the Steady State and Transient Critical Heat Flux of Multi-Rod Bundles for PWR and BWR with Freon as Model Fluid	**Project Leader:** Katsaounis
Initiated (Date): 11.72 **Status:** Continuing **Completed (Date):** 30.6.78 **Last Updating (Date):** December 1976	

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 objectives

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 transients 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

The experiments of part I and II are carried out in a so called steady state burnout facility (Fig. 1). For the tests during pressure transients (blow down) the loop from fig. 1 is reconstructed according to the facility for RS 37 research program (Fig. 2). The extension and the conformity of the freon loop to the facility for RS 37 is necessary for the equivalence of the experimental results between freon and water.
The calculation of the characteristics of the loop for blow down experiments was made using the computing program COVACU.
The computer program COVACU is the modification of the computer program BLDGWH /‾1_7. The program COVACU is written in FORTRAN IV, and describes the dynamic behavior of a blow down testing loop für pressure transients.

The program COVACU allows to calculate the time-dependent cross-section of the valves in the feed line, and the blow off lines so that inside the testing loop (fuel bundle) the pressure curve of the reactor is obtained with a practicable accuracy. The behavior of the control system is taken into consideration.

The present version of the program COVACU allows investigations for a testing loop with water only. When the program COVACU is modified, investigations for a testing loop with Freon-12 can be performed.

Short description of the freon loop for the blow-down experiments:

The loop is operating at steady state condition before the blow down procedure starts. During this steady state condition the initial parameters of the blow down phase such as pressure, heat flux, massflow, and enthalpy at the inlet of the test section are obtained in the loop. At beginning of the blow down phase the quick - acting valves HSA 1 to HSA 3 will shut down, and thus the closed cycle of the loop is separated from the test section. During the transition time from steady state to the blow down condition the regulating valve AR will open, and the loop controlled system hold the loop under steady state condition again. The fluid flows now from the storage battery tanks DS1 to DS3 to the test section. Finally real blow down condition is initated by opening and regulating the control valve AS/SA5.

5. Progress to Date

The project is more than one year behind schedule. The main reason for the delay is caused by difficulties with the electric insulation of the original reactor spacer for the test section. The experiments were shut down at the beginning of this year, because the electric insulation of the test section was again defect. A development of a new insulation-method was necessary.

The loop for the blow down experiments is laid out with consideration of the new requirements (the experiments have to be carried out at a 5 x 5 rod bundle, reactor like, and regulated according to the actual calculated reactor - blow down curves), and it's new construction will finished at the beginning of next year.

6. Results

Experiments of Parts I and II:

The experiments using the PWR-test section were interrupted at March of this year, because the insulation of the test section was defect again. It was started immediately to develop a new insulation method, which is based on the insulation of the heated rods. The previous insulation of the original reactor grids was unsuccessful. The grid configuration is very disadvantageous for an electric insulation.

The electric insulation of the heated rods must fulfill the following conditions:

- a very good and safe electric insulation,
- small thickness ($\delta < 0.2$ mm) and very low thermal insulation. The temperatuse gradient behind the insulation must be very small.
- in the region of the rods insulation (i.e.grid) the power supply must be quite less or equal to the power supply of the remaining rod outside of the insulated region.
- The insulation must not essentially disturb the flow in the grid region.

All these requirements are fulfilled by a 0,1 - 0,15 mm thick insulation of the heated rods with "Thermoguss+Epoeytresin". The test of this insulation using original grids in an annular test section was successful. 165 burnout messurments were made at very high burnout - heatfluxes without any mentionable failure on the insulation.
All 44 heated rods of the PWR-test section are completely insulated (about 350 positions) and at the beginning of next year the experiments will coutinue.
Some of the evaluated experimental burnout results are presented in reference /‾2‾/. Fig. 3 shows a comparison between experimental and calculated burnout values. Some of the measured mixing values are evaluated from KWU using the computing program "Thermohydraulik" modified for Freon 12.
The BWR-test section for the following experiments is now also ready. Only it's insulation with the new insulation method stands out.

Experimental facilities of Part III:

As mentioned before the loop is laid out. Following details are completed:

- Calculation of the loop chracteristics /‾3_7 using the computing program COVACU. The calculations are made for water.

- The facility is laid out incl. all components.
- The regulating and control system of the loops is laid out too.
- The instrumentation of the loop is just ready.

For the estimation of the costs all components of the planed facility are inquired.

7. Next steps

a) Finishing the experiments type I and II with the PWR-test section

b) Carry-out the experiments type I and II with the BWR-test-section

c) Finishing the calculation of the loop characteristics using the computing program COVACU modified for freon 12 lay-out and ordering the components for the blow-down loop.

8. Relations with other projects

see report IRS-F24
RS 164
RS 176

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9. References

[1] KWU - Erlangen: Emmerling
BLDGWH - Computer code for simulation of a blow down experiment,
user's manual not available.

[2] GKSS 76/E/44 H. Fulfs, A. Katsaounis, C.v.Minden (English)
Burnout Experiments with 6 x 6, 8 x 8 and 7 x 7 Rod Bundle Test Sections using Freon as Model Fluid.

[3] GKSS 76/I/41 R. Dietrich (German)
Grundlegende Untersuchung zur Auslegung des Steuerungs- und Regelungskonzeptes für die erweiterte Frigen-Versuchsanlage zur Durchführung von Burnout-Untersuchungen (RS 64 - Part III).

[4] GKSS 76/I/11 H. Fulfs (German)
MESS - Ein Rechenprogramm zur Auswertung und Darstellung digitalisierter Meßwerte.

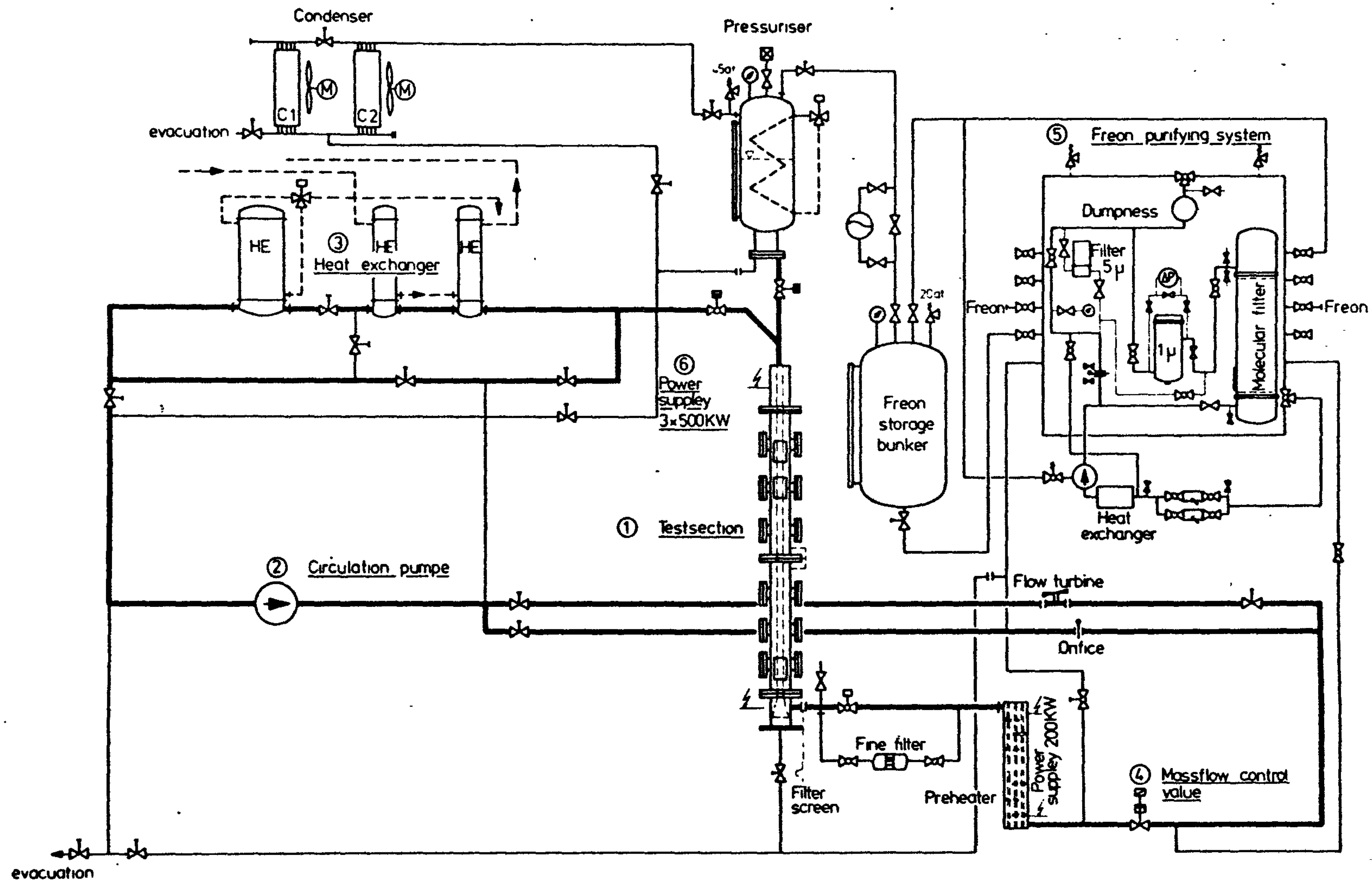
[5] GKSS 72 05 AT - C-08 H. Fulfs (German)
Ergebnisse der Durchmischungsversuche mit Vorwiderstand (1. Abschnitt) PWR / Part I / RS 64.

[6] GKSS 72 05 AT - B - 04 A. Katsaounis (German)
Statusbericht RS 64.

[7] IRS - Research reports : IRS - F - 30 - 34.

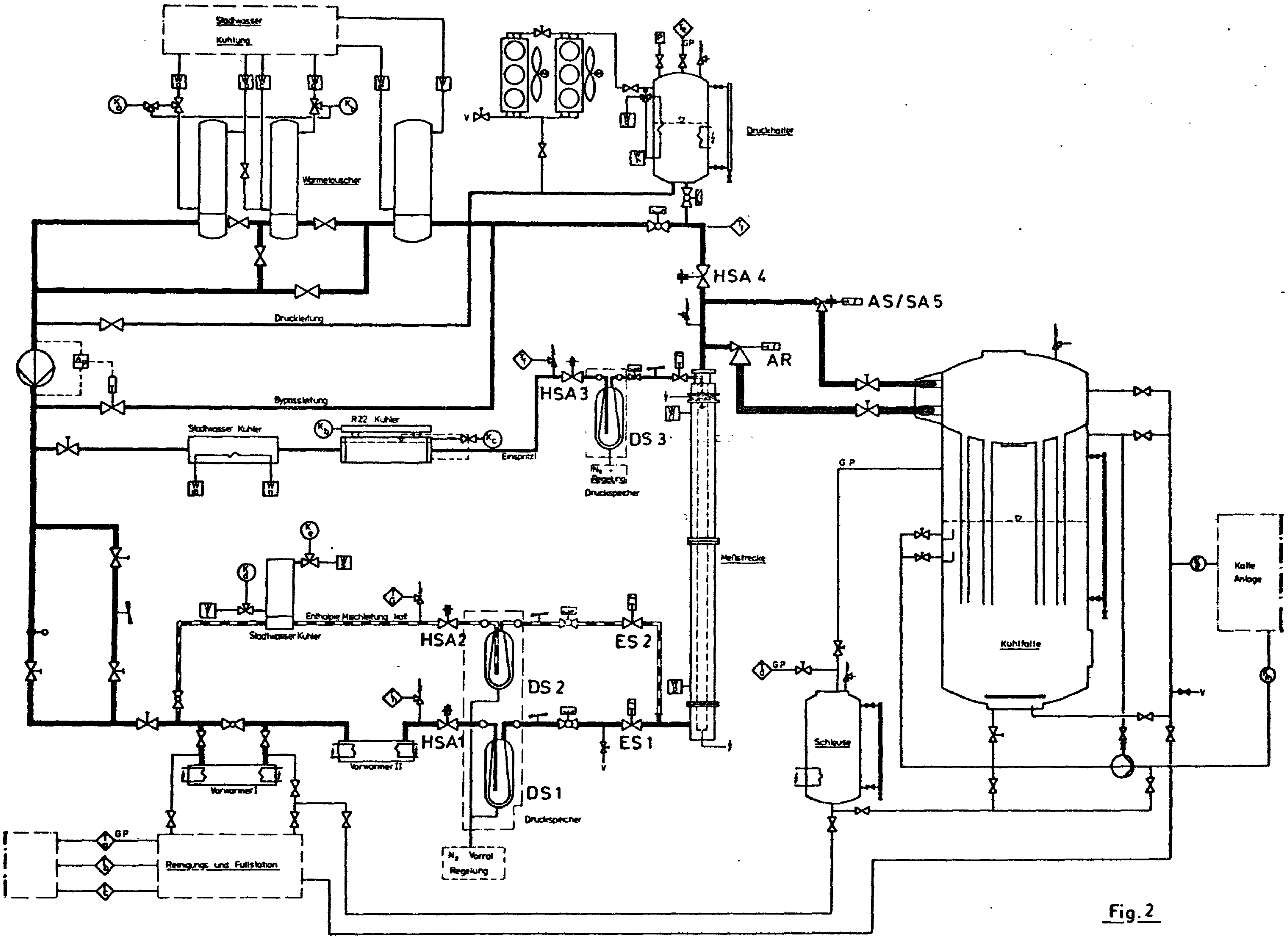
10. Degree of availability

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



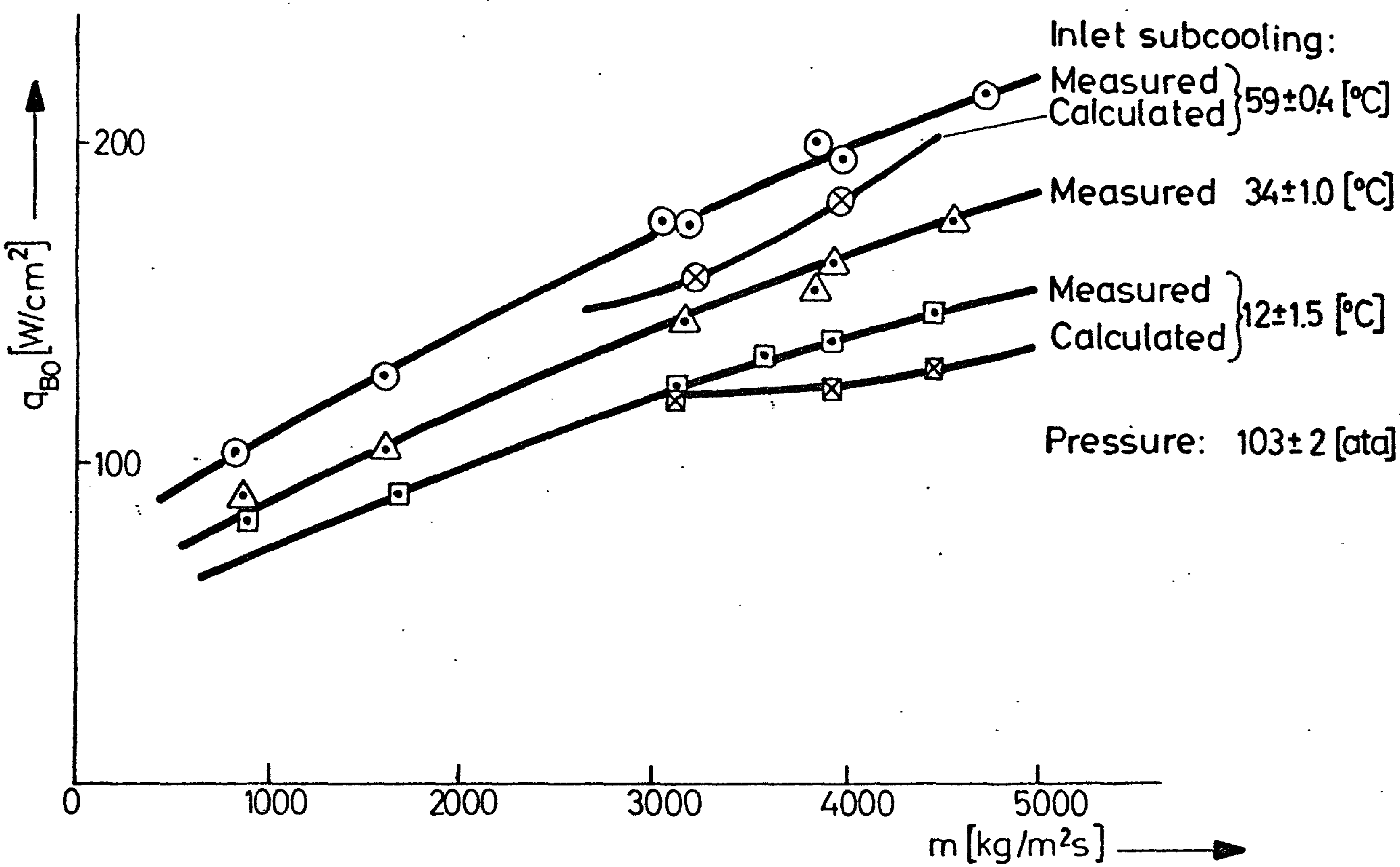
GKSS Fig 1

Freon Loop for Steady - State and for Massflow, Power Transient Experiment



Stadtwasser Kühlung
Wärmetauscher
Druckhalter
HSA 4
AS/SA 5
AR
Druckleitung
Bypassleitung
Stadtwasser Kühler
R22 Kühler
Einspritzl
HSA 3
DS 3
Druckspeicher
Meßstrecke
Kühlfalle
Kälte Anlage
Enthalpie Mischleitung kalt
Stadtwasser Kühler
HSA 2
ES 2
DS 2
HSA 1
ES 1
DS 1
Druckspeicher
Vorwärmer II
Vorwärmer I
Schleuse
Reinigungs und Füllstation
N_2 Vorrat Regelung
GP

Fig. 2



GKSS Fig.3 : Comparison Between Measured and Calculated Values vs. Massflowrate

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Classification: 1.1.2	
Title 1 (Original Language): Stationäre DNB-Messungen in Frigen mit komplexer Abstandshalter-geometrie (RS-176-II.1.3, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GKSS/A
Title 2 (English): Steady state DNB Measurements in Freon with complex Spacer Geometry	**Project Leader:** Fulfs/Katsaounis
Initiated (Date): 1. 9. 1975 **Status:** Continuing **Completed (Date):** 31. 12. 1978 **Last Updating (Date):** December 1976	

1. General aim

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

2. Particular objectives

The program is divided in the following three parts:

I. cold flow pressure drop measurements,

II. mixing experiments with power, measurements of subchannel exit temperature,

III. Critical heat flux measurements.

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

3. 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.

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4. Experimental facilities

The geometry of the test section is a square array of 25 rods with a pitch of $14.3 \cdot 10^{-3}$ m. 16 of them are electrically-heated with a relative power of 1.0, and 6 with a relative power of 1.2, while 3 are unheated. All heated rods are instrumented with thermocouples near the exit end of the heated length, which is 2.985 m long. The outer diameter of the heated rods is $10.75 \cdot 10^{-3}$ m, that of unheated rods $1.372 \cdot 10^{-3}$ m. Wall spacing of the heated rods is $4.544 \cdot 10^{-3}$ m, grid spacing is 0.534 m. 4 pressure taps give information about inlet and outlet pressure, and pressure drop across spacers.

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

5. Progress to Date

The construction of the test section is finished. The fabrication of the main parts of the test section is also completed. The construction for the modification of the electric components is finished and the transformer (700 kW) is ordered.

6. Results

There is not any essential delay in schedule, but as the tests themselves will be run in between RS 64. Any delay of RS 64 will shift the schedule of RS 176. Some new theoretical derivations about the scaling laws water/freon [1] are made.

7. Next Steps

Delivery of the 700-kW-Transformer at the end of the first quarter of next year.

Instrumentation and assembly of the test section.

The experiments can be started three months after the end of the runs RS 64 part I and II.

8. Relation with other Projects

See annual report 1976.

9. References

[1] GKSS 76/I/34 G. Rinne (German)
Beschreibung einer Datei für Meßwerte der kritischen Heizflächenbelastung und der zugehörigen Rechenprogramme.

[2] IRS-Research reports: IRS-F-30, 31, 33

10. Degree of Availability of the Reports

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

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

1. General Aim and 2. Particular Objectives

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

3. Research Program

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

4. Experimental Facilities

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

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

5. Progress to Date

The experiments in water have been completed.
The parameter regions of the tests were:

System pressure:	68 - 160	bar
Mass flow density:	598 - 3688	kg/m^2s
Initial temperature:	135 - 336	°C
Max. bundle power:	5,1	MW

The experiments are to be divided into three categories:

- pressure drop experiments (only at 70 bar; without heating)
- mixing experiments (only at 160 bar; with heating but no burnout)
- burnout experiments (at 70, 100, 140 and 160 bar; with heating until burnout)

The experiments in freon will be carried out in 1978 at GKSS under contract RS 176.

6. Results

The experimental data of the test series 1 were recalculated by THERMOHYDRAULIC. The agreement of the critical heat density was very good at 160 bar with the calculated W-3-R-Grid correlation. The use of this correlation for the core design of KWU-reactors was justified by these experiments. The maximum critical bundle power (5,1 MW) was reached at a mass flow density of 3600 kg/m^2s.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

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

RS 176 Freon tests with the same bundle geometry

9. References

Dr. Ulrych, P. Suchy:
Stationäre DNB-Messungen an Brennstabbündeln mit komplexer Abstandshalter-Geometrie in Wasser und Frigen
Teilvorhaben 1: Versuche im Wasserkreislauf
Abschlußbericht BMFT RS 164
Kraftwerk Union Aktiengesellschaft (Aug. 1976)

10. Degree of Availability

The report is company confidential.

Classification: 1.1.2	
Title 1 (Original Language): Entwicklung einer Massenstromdichte-Meßmethode für transiente Zweiphasenströmungszustände mittels der magnetischen Kernspinresonanz (RS 188 - I.1.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: RWTH Aachen
Title 2 (English): Development of a mass flow density measuring method for testing transient two-phase-flow states by use of Nuclear Magnetic Resonance	Project Leader: Prof.Dr. R. Kosfeld
Initiated (Date): 1.4.76 Status: Continuing Completed (Date): 31.3.77 Last Updating (Date): 31.12.76	

1. General aim:

Development of a measuring method based on NMR, for determining the quasi-stationary and transient mass-flow density up to about $10^3 g/cm^2 s$ in two-phase flows (Water/Vapor) with a time resolution up to the millisecondrange and pressures up to 160 bar, temperatures up to 325°C and flow-velocities up to 10^4 cm/s. This goal should be attained in three consecutive steps; experiments in the first step should deal with temperatures up to about 150°C, pressures up to 5 bar and flow velocities up to 10m/s.

2. Particular objective:

The efficiency of an NMR Method for Determining the mass flow in two phase flows up to the experimental conditions encountered in refill- and blowdown-experiments should be ascertained.

3. Research Program:

3.1 Calculation of theoretical outflow curves.
3.2 Construction of the glass set up for flow tests, including starting of apparatus and test measurements.
3.3 NMR-flow measurements.
3.4 Testing of the mounting of the measuring coil in a steel tube.
3.5 Setup of the apparatus for T_1, T_2, D_s-measurements.

4. Experimental facilities, computer codes:

Programs were written for model calculations of NMR signals of flowing media.
Other programs were developped for constructing and adjusting the measuring coil. All programs were written in BASEX with the exeption of some Assembler Routines. Asimple, open setup for flow experiments was built, in order to check the model calculations.

5. Progress to date:

The expected signals of various flow models were computed in advance. The weight function ofthe high frequency measuring coil, as a function of its geometric parameters, had to be set up as 'Equipment Function'. In order to check the computed 'Equipment Function' water filled ampules were drawn through the measuring apparatus. Later on, first measurements were performed on laminar and turbulent flows and compared with the model computations. Flow measurements in a polarizing field with various gradients were performed in order to evaluate the influence of the quality of the polarizing magnet. Numerical computations for the construction and adaption of the measuring coil were performed.
The apparatus for measuring the relaxation time constants T_1 and T_2 and the self diffusion coefficient D_s has been assembled, except for the pressure arming. Sample preparations for the T_1, T_2, D_s measurements have been started.

6. Results:

Measurements performed on laminar flows were compared with the model calculations. A very good agreement between model and experiment could be seen.
The apparatus for T_1, T_2, D_s measurements was adapted to high frequency. A retransformation algorithm is being developed at the present moment.

7. Next steps:

- Flow measurements on laminar and turbulent flows
- Testing of the mounting of the measuring coil in a steel tube
- T_1, T_2, D_s-measurements
- Development of the retransformation algorithm.

8. Relation with other projects:

Similar objectives are pursued by the following research programs:
- RS 135 Signal correlation
- RS 146 Radiotracer measuring procedure
- RS 147 Drag-Body development
- True mass flow meter

9. References:

- Becker Farrar: Pulse and Fourier Transform NMR (English) Academic Press New York 1971
- Abragam: The Principles of Nuclear Magnetism (English) Oxford At The Clarendon Press 1961
- Bergmann et al.: Study of the Development of a Mass Flow Measuring Method using NMR for transient Two Phase Flow States (German) KWU-Reactor Technique, October 1975 Report
- Puhr-Westerheide: Model Computations of NMR Signals of One Phase Flows (German) Internal Report (July 1976)
- Puhr-Westerheide: An Algorithm for the Numerical Computation of the Mass Coefficient dm/dv over the Velocity by using the NMR Signal generated in Flowing Media (German) Internal Report (Dezember 1976)
- Puhr-Westerheide: NMR Methods applied to Flowing Media (German) Internal Report (November 1976)

Classification: 1.1.2	
Title 1 (Original Language): Dichtemessung in Zweiphasenströmung (Wasser/Dampf) mittels Ultraschallsonden (RS 225 - I.1.1, Jahresbericht A 1976)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: IKT, TU Berlin
Title 2 (English): Density-measurement in a two-phase-flow (water/steam) by means of ultrasonic detectors	Project Leader: Prof. U. Wesser
Initiated (Date): Oct. 1, 1976 Status: continuing Completed (Date): Sept. 31, 1978 Last Updating (Date): Dec. 31, 1976	

1. General Aim

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

2. Particular Objectives

The aim of this project is the development of a measuring method for the average density of a fluid, based on the damping of ultrasonic waves in metallic rods by the surrounding medium.

3. Research Program

3.1 Theoretical work

The aim of the theoretical investigation is to design a physical model and to solve it by means of a digital computer. One or more geometries for detectors are to be determined with the results taking into account the physical and technical parameters.

3.2 Pre-Experiments

Pre-experiments serve to ratify and correct the theory as well as to optimize the geometry of the detectors. Detectors with different dimensions are to be tested in water and water/air mixture. In addition thus to be investigated if this method can be used to measure the fluid level

in a tank.
With these pre-experiments the influence of both the transmitter and receiver is to be investigated.

3.3 Development of Electronical Network

The aim of the development is to get a simple electronical network with high accuracy. This also encludes a short response time and a suitable preparation of the results.

3.4 Tests under stationary and transient Conditions

These tests will be carried out under realistic conditions in two-phase-flow (water/steam). For these tests installations are existing (e.g. Project RS 135) or projected. Changing over to higher mass flows and transients there are considered tests at other locations, for example at Karlsruhe (RS 145) and/or Ispra (RS 109).

3.5 Constructive design

At the end of the applicated intervall a prototype is to be at disposal, which not only can be used under laboratory conditions.

4. Experimental Facilities, Computer Codes

-

5. Progress to Date

5.1 Theoretical work

The theoretical work consisted mainly of literature studying. Furthermore, a physical model and its mathematical background were developed.

5.2 Pre-Experiments

Pre-experiments with simple arrangements were carried out to get information of the coupling between transmitter or receiver and detector.

6. Results

6.1 Theoretical work

The literature studying points out that there are investigations concerning to the behaviour of ultrasonic delay-lines, the propagation of ultrasonic waves in a helical wire without damping and the pyhsical process of the propagation in curved rods. All these publications deal with the propagation of pulses while our transmitter uses a continuous signal.

A physical model is designed and there is also a mathematical description in a simplified form.

6.2 Pre-Experiments

The pre-experiments confirm the dependence of the damping of ultrasonic waves in curved lines on the density of the surrounding medium.

7. Next Steps

7.1 Theoretical Work

The solution of the mathematical description will be improved. The theory is to be enlarged and modified for level measuring.

7.2 Pre-Experiments

Further pre-experiments will be performed and tested on their accordance to the theory. The detectors will be improved. Afterwards experiments will be performed in a water/steam mixture.

7.3 Development of Electronical Network

The first step is to analyse the domain of the results. Then the appropriate network is to be found.

8. Relation with other Projects

It is intended to use this density measurement method under nuclear conditions in a two-phase-flow at GfK-Karlsruhe (RS 145) and at Ispra (RS 109). This measurement method will be tested in project RS 135 (TU-Berlin) too.

9. References

-

10. Degree of Availability of the Reports

-

Classification: 1.1.2	
Title 1 (Original Language): Theoretische Untersuchungen im Zusammenhang mit Druckabsenkungsvorgängen im Primärsystem (Blowdown, Kernnotkühlung-Hochdruckphase) (ATT 085A - I.1.1 - I.1.2, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (english): Simulation of Reactor System Blowdown and High Pressure Emergency Core Cooling	Project Leader: Dr. H. Karwat
Initiated (Date): Completed (Date): Status: continuing Last Updating (Date): December 1975	

1. General Aim

The general aim is to develop computer codes to predict thermal hydraulic response of the primary system of water cooled reactors during the blowdown phase of a loss-of-coolant accident (LOCA) with consideration of high pressure emergency core cooling.

2. Particular Objectives

- Improvement of the code BRUCH-D
- Development of the code DRUFAN
- Development of the code DAPSY
- Performance of theoretical work accompanying related experimental research projects.

3. Experimental Facilities and Program

Project Status

4.1 Progress to Date

The code BRUCH-D has been improved. Two new versions, BRUCH-D-04 /1/ and BRUCH-D-05 /2/ have been finished. In BRUCH-D-04 a better simulation of the heat transfer in the core region is realized by a more detailed assignment of axial fuel rod sections and local fluid conditions. In BRUCH-D-05 extended sets of heat transfer and critical heat flux correlations are incorporated. Also a subroutine smoothing the specific core mass flows, which are used for determining the heat transfer and critical heat flux, has been inserted. A study of the influence of the most important thermal hydraulic parameters on heat transfer and critical heat flux has been conducted. The results in graphical representation permit a quick estimation of the heat transfer coefficient and the critical heat flux for given values of thermal hydraulic parameters /3/.

Further development of the code DRUFAN has been continued. DRUFAN is based on a multinode model taking into account thermodynamic nonequilibrium within the nodes during evaporation and condensation processes. Full flexibility in the arrangement of nodes and flow pathes restricted only by the core storage capacity of the computer permits an appropriate geometrical representation of the primary system of a PWR, as well as the representation of a pressurized experimental fluid dynamic system in the required detail. The techniques used for determination of critical discharge rates - i.e. the use of the instationary Bernoulli equation for the initial phase of the transient, and the use of the Moody model for critical discharge - have been found unsatisfactory in the case of a outlet nozzle represented by several nodes. Therefore a critical discharge model for one and two phase fluid has been developed. The model is based on the stationary one-dimensional description of the flow (1-D Model) with consideration of thermodynamic non-equilibrium phenomena. The critical mass flow rate is controlled by Mach 1 at the discharge orifice.

In the field of "Fluid and Structural Dynamic Coupling" the first step has been the coupling of the fluiddynamic code DAPSY to a finite element structuraldynamic code based on a special two dimensional axisymmetric shell element representation. DAPSY has been developed for the investigation of pressure wave propagation phenomena, in which the fluiddynamic systems are represented by a multi-dimensional network of 1-D flow channels. For the coupling the code DAPSY had to be slightly modified and expanded.

4.2 Essential Results

DRUFAN code verification has been undertaken by the simulation of the depressurization experiment of Edwards and O'Brien (1) on which the USAEC-Standard Problem 1 (2) is based. The calculations have been performed using both the Moody model and the new 1-D model to determine the critical discharge rate. Good agreement between experimental and calculated pressure histories could be attained by applying the 1-D model. For the calculation with the Moody model a discharge coefficient CD = 0.5 has been found necessary.The undershoot in pressure below the saturation pressure, observed in the experiment at the closed end of the test pipe, caused by thermal non-equilibrium between water and steam phase during evaporation, could be simulated with the code DRUFAN satisfactorily /4/. The consideration of thermodynamic non-equilibrium phenomena in the code not only yielded a better prediction of the pressure transients, it also avoids the unrealistic pressure oscillations, which have been a typical multinode model problem.

The code has also been applied to analyze one of the first RS 77-depressurization experiments. The experiments have been carried out to investigate, in particular, thermodynamic non-equilibrium phenomena. With the code DRUFAN the experimental results, characterized by the occurence of non-equilibrium conditions, could be simulated without difficulty.

The RS 16/2 depressurization experiment DWR-5 (3) has been simulated as w[illegible] the code DAPSY, assuming a rigid core barrel wall, as by the coupled fluid and structural dynamic program, assuming an elastic wall. Better agreement between experimental and calculated results have been obtained with the coupled fluid and structural dynamic program.

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In connection with the specifications of the HDR blowdown experiments /5/ several runs with the codes DAPSY and TSHOCK have been performed /6,7/. DAPSY has been applied to problems concerning pressure wave propagation phenomena, TSHOCK for estimation of the thermal stresses in the HDR pressure vessel during the depressurization period.

(1) A.R. Edwards, T.P. O'Brien
Studies of Phenomena Connected with the Depressurization of Water Reactors
J. of the British Nucl. Energy Soc. April 1970

(2) R.W. Garner
Comparative Analyses of Standard Problems
Standard Problem 1
(Straight Pipe Depressurization Experiments)
Aerojet Nuclear Company
Interim Report I-212-74-5.1 October 1973

(3) Battelle-Institut, Versuchsergebnisse aus dem Vorhaben RS 16/B, Techn. Bericht RS 16/B, August 1975

5. Next Steps

At present a comparison of BRUCH-D-05 with RELAP-4 is in preparation. In particular, the influence of the different methods of simulation of the transient main coolant pump behaviour on the blowdown will be studied.

A detailed description of the current version of DRUFAN, which is applicable to analyze the initial blowdown phase of a PWR-LOCA, will be prepared, together with the optimization of the computer code. The verification of the code will be further advanced by simulation of appropriate experiments. In particular the verification of the correlation determining the mass transfer between water and steam phases during evaporation or condensation will be stressed. By insertion of additional subroutines (e.g. fuel rod model, pump model) the program DRUFAN will be extended in such a way that total blowdown can be analyzed.

The fluiddynamic code DAPSY will be coupled to an existing finite element structural dynamic code which is applicable for the representation of any structural system.

6. Relation To Other Projects

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7. Reference Documents

/1/
K.J. Liesch, G. Raemhild
BRUCH-D-04 - Ein Rechenprogramm zur Analyse der fluid- und thermodynamischen Vorgänge im Primärkreis von Druckwasserreaktoren bei schweren Kühlmittelverlustunfällen (4. erweiterte Version). Programmbeschreibung
MRR-P-15, Januar 1975

/2/
K.J. Liesch
BRUCH-D-05 - Ein Rechenprogramm zur Analyse der fluid- und thermodynamischen Vorgänge im Primärkreis von Druckwasserreaktoren bei schweren Kühlmittelverlustunfällen (5. erweiterte Version)
MRR-P-20, Dezember 1975

/3/
K.J. Liesch, G. Raemhild, K. Hofmann
Zur Bestimmung des Wärmeübergangs und der kritischen Heizflächenbelastung im Hinblick auf besondere Verhältnisse in den Kühlkanälen eines DWR bei schweren Kühlmittelverlustunfällen
MRR 150, Juni 1975

/4/
K. Wolfert
USAEC Standard Problem 1
Calculations with the code DRUFAN
presented at CSNI-AD HOC GROUP ON EMERGENCY CORE COOLING, Bethesda, Maryland, USA, September 1975

/5/
T. Grillenberger, B. Österle, K. Wolfert
Beiträge des LRA zu speziellen Kapiteln des PNS-Arbeitsberichtes Nr. 47/75 "Detailspezifikation der sicherheitstechnischen Untersuchungen am HDR-Blowdown-Versuch". Interner Bericht
MRR-I-50, Juni 1975

/6/
Beiträge zur Detailspezifikation der Blowdown-Versuche am HDR. Interner Bericht
MRR-I-40, April 1975

/7/
T. Grillenberger
TSHOCK - Ein Rechenprogramm zur Ermittlung thermischer Beanspruchung von Behälterwänden aufgrund instationärer eindimensionaler Wärmeleitvorgänge, Programmbeschreibung. Interner Bericht
MRR-I-52, September 1975

8. Degree of Availability

Documents are available through
Laboratorium für Reaktorregelung und Anlagensicherung
D-8046 Garching
Federal Republik of Germany

Reports MRR-I are confidential and therefore normally not available.

Classification: 1.1.2	
Title 1 (Original Language): Untersuchungen zu Druckabsenkungsvorgängen (Blowdown) und zur Hochdruckphase der Kernnotkühlung ATT 085 - I.1.1 - Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMI **ORGANIZATION:** LRA, Garching
Title 2 (English): Simulation of Reactor System Blowdown and High Pressure Emergency Core Cooling	**Project Leader:** Dr. H. Karwat K. Wolfert
Initiated (Date): **Completed (Date):** **Status:** continuing **Last Updating (Date):** December 1976	

1. General Aim

The general aim is to develop computer codes to predict thermal hydraulic response of the primary system of water cooled reactors during the blowdown phase of a loss-of-coolant accident (LOCA) with consideration of high pressure emergency core cooling.

2. Particular Objectives

- Improvement of the code BRUCH-D
- Development of the code DRUFAN
- Development of the code DAPSY
- Performance of theoretical work accompanying related experimental research projects.

3. Experimental Facilities and Program

Not relevant.

4. Project Status

4.1 Progress to Date

The blowdown code BRUCH-D has been improved. In the version BRUCH-D-05 a better simulation of the recirculation pump behaviour has been provided by a model based on single-phase homogeneous curves for head and torque. The degrated load on torque with respect to two-phase flow conditions are considered. Additional effects on pump coast-down, such as a break of the coupling between motor and pump, fly-wheel release or a pump rotor lockage, are taken into account. An extended version BRUCH-D-06 has been completed. In this version a module simulating the heat removal of the primary system structures and a procedure for approximate determination of the water level in the pressure vessel are incorporate /1/.

The development of the code DRUFAN - a code to describe thermal hydraulic transients of two-phase fluids during a loss-of-coolant accident - has been continued. DRUFAN is based on the lumped-parameter technique with consideration of thermodynamic nonequilibrium within the defined control volumes. The first DRUFAN version for analysis of the initial blowdown phase has been finished /2, 3, 4/.

The first version of the code DAPSY - a code for the investigation of pressure wave propagation processes - has been completed. With DAPSY, multi-dimensional geometries may be represented by networks of one-dimensional flow paths. Thermodynamic nonequilibrium phenomena are taken into account /5, 6/. The development of a code for the coupled treatment of fluid and structure dynamics has been continued. The principle effect of structural response on the fluid behaviour has been studied by simulation of a BWR feed water pipe with quick closing check valves /7, 8/.

In the framework of "OECD Ad-hoc Working Group on Emergency Core Cooling" co-operation the OECD/CSNI Standard Problem No. 6 has been defined and the data describing the experiment have been compiled /9/. Standard Problem No. 6 will be based on the experiment SWR-2R performed at the Battelle Institut, Frankfurt am Main, simulating water level rise processes (1).

4.2 Essential Results

DRUFAN code verification has been continued by the simulation of the Battelle RS 16/2 experiment DWR-1R (2). On the basis of this experiment especially, the one-dimensional model incorporated in DRUFAN to evaluate discharge rates has been tested in particular. By utilisation of an appropriate relation describing the increased mass transfer between liquid and vapour phases at high Re-numbers, a good agreement between calculation and experimental data could be achieved. The calculations have been presented at the OECD/NEA Specialists' Meeting on Transient Two-Phase Flow in Toronto, August 1976. The simulation of ECCS injection processes has been checked. Even at conditions extremely unfavourable with respect to the analytical simulation (injection into a very small control volume, high injection rate), the processes could be simulated without problem. The typical pressure oscillations, generally computed by codes based on thermodynamic equilibrium models, did not appear.

The Battelle blowdown experiment RS 16/2 DWR2 (3), an experiment with an orifice plate at the outlet, has been simulated by DAPSY. Good agreement between experimental and calculated results has been realized /10/.

5. Next Steps

Further work on DRUFAN and DAPSY verification, based e.g. on the simulation of Battelle RS 16 or LOFT blowdown experiments, is scheduled. DRUFAN will be extended for small leak simulations.

The code describing the coupled fluid-structure dynamics will be tested by simulating a multi-dimensional problem.

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(1) Battelle-Institut Frankfurt/M., Technischer Bericht BF-RSOO16B-32-7, Juli 1976

(2) Battelle-Institut Frankfurt/M., Bericht zum Forschungsvorhaben "Information zu den Referenzversuchen", Kurzfassung der Ergebnisse, Mai 1974

(3) Battelle-Institut Frankfurt/M., Bericht zum Forschungsvorhaben RS 16/2, Versuchsergebnisse vom Druckentlastungsvorgang im Druckbehälter mit flexiblen DWR-Einbauten, Versuchswiederholung DWR 2, Sept. 1975

6. Relation to Other Projects

See above.

7. Reference Documents

/1/
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

/2/
K. Wolfert
Die Berücksichtigung thermodynamischer Nichtgleichgewichtszustände bei der Simulation von Kühlmittelverluststörfällen
Proceedings, Reaktortagung, Düsseldorf, März/April 1976

/3/
K. Wolfert
The Simulation of Blowdown Processes With Consideration of Thermodynamic Nonequilibrium Phenomena
OECD/NEA Specialists' Meeting on Transient Two-Phase Flow, Toronto, August 1976

/4/
K. Wolfert, M.J. Burwell, D. Enix
DRUFAN - A Code Based on a Multinode Model Simulating Blowdown Processes with Consideration of Thermodynamic Nonequilibrium Phenomena. Preliminary Code Description. Internal Report
MRR-I-81, December 1976

/5/
T. G[illegible]enberger
The Computer Code DAPSY for the Calculation of Pressure Wave Propagation in the Primary Coolant System of Light Water Reactors. Internal Report
MRR-I-66, April 1976

/6/
T. Grillenberger
DAPSY - Ein Rechenprogramm für die Druckwellenausbreitung im Reaktorkühlkreislauf. Programmbeschreibung
MRR-P-24, Oktober 1976

/7/
T. Grillenberger
Erste Ergebnisse von Testrechnungen mit einem Programm zur gekoppelten Berechnung fluid- und strukturdynamischer Vorgänge; fluiddynamischer Teil
Proceedings, Reaktortagung, Düsseldorf, März/April 1976

/8/
T. Österle
Erste Ergebnisse von Testrechnungen mit einem Programm zur gekoppelten Berechnung fluid- und strukturdynamischer Vorgänge; strukturdynamischer Teil
Proceedings, Reaktortagung, Düsseldorf, März/April 1976

/9/
F. Steinhoff
OECD Standard Problem No. 6
Determination of Water Level and Phase Separation Effects During the Initial Blowdown Phase. Version 15.11.1976
Second Workshop for CSNI and LOCA Standard Problems, Paris, December 1976

/10/
F. Bruckmeyer, T. Grillenberger
Nachrechnungen zu dem Versuch DWR2 des Forschungsvorhabens RS 16/2, Interner Bericht
MRR-I-78, Dezember 1976

8. Degree of Availability

Documents are available through
Gesellschaft für Reaktorsicherheit mbH
D-8046 Garching, Forschungsgelände, Federal Republic of Germany

Reports MRR-I are confidential and therefore normally not available.

Classification: 1.1.2

Title 1 (Original Language):	COUNTRY:
Beteiligung des LRA am LOFT-Programm der USNRC (RS 198 - I.1.1, Jahresbericht A 76)	BRD
	SPONSOR: BMFT
	ORGANIZATION: LRA Garching
Title 2 (English): Participation by the LRA in the LOFT Project of the USNRC	Project Leader: Dr. Karwat
Initiated (Date): July 1976	Completed (Date): December 1978
Status: Continuing	Last Updating (Date): December 1976

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 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 program of LRA includes two main points:

- Pre- and post-calculation of the LOFT experiments
- Participation in the analysis and the evaluation of technical con-

clusions from the results of the LOFT project.

4. Experimental Facilities, Codes

The codes DAPSY /1/ and DRUFAN /2/ are used mainly to calculate pressure wave propagation phenomena. The codes DRUFAN and BRUCH-D-06 /3/ are to be applied to simulate the whole decompression including the injection process until the beginning of the reflood phase.

5. Progress to Date

Input date sets for the codes DAPSY, DRUFAN and BRUCH-D-06 have been completed and are listed in /4/. Special problems are involved with the simulation of the fluiddynamic behaviour of the Quick Opening Blowdown Valves (QOBV) which is supposed to simulate the real break opening mechanism of a pipe. The analysis of first results shows the necessity of a more detailed geometrical and fluidmechanical simulation of the QOBV in order to obtain sufficient agreement between calculated and measured results.

6. Results

First results of the post-calculation of the L1-2 test, computed with DAPSY, are reported in /4/ and are to be presented at the Reaktortagung 1977.

7. Next Steps

A continuous refinement of the input data should lead to an adequate simulation of the LOFT facility, future work will be the pre-calculation of the L1-4 test.

8. Relation with Other Projects

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9. References

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

/2/
K. Wolfert
The Simulation of Blowdown Processes with Consideration of Thermodynamic Nonequilibrium Phenomena
Paper presented at the Specialists' Meeting on Transient Two-Phase-Flow, OECD/NEA, Toronto (Canada), August 3rd and 4th, 1976

/3/
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

/4/
K.J. Liesch, K. Hofmann, F.J. Ringer
Application of the DAPSY Code on LOFT Nonnuclear LOC Experiment L1-2. Internal Report
MRR-I-79, December 1976

10. Degree of Availability of the Reports

Documents are available through
Gesellschaft für Reaktorsicherheit mbH
D-8046 Garching, Forschungsgelände
Federal Republic of Germany

Reports MRR-I are confidential and therefore normally not available.

140-26-22/4160-01 Classification : 1-1-2 213

Titre	Pays :
Thermohydraulique du LOCA. Etude de la phase dépressurisation d'un réacteur à eau pressurisée : Programme OMEGA	FRANCE
	Organisme directeur : CEA
Titre (anglais) LOCA thermohydraulics. P.W.R. Blowdown Studies : OMEGA Projects.	Organisme exécuteur : CEA/DTCE STT (GRENOBLE)
	Responsable : M. COURTAUD (STT)
Date de démarrage : 01/01/72 Date prévue d'achèvement 31/12/78 Etat actuel : en cours Dernière mise à jour : 21/01/77	Scientifiques : R. RICQUE J.C. ROUSSEAU

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.

Objectifs particuliers :

Développer des modèles physiques pour l'interprétation des expériences : élaboration de corrélation pour RELAP 4, validation de modèles physiques pour les codes 2ème génération.

Installations expérimentales et programme :

- Boucle OMEGA : Pression 170 bars, débit max. 20 kg/sec, 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.

  2) Assemblage de 36 barreaux combustibles type 17 X 17 de 3,65 m de long avec un flux cosinus ; taille de brèche : 4 cm2 (amont, et aux deux extrémités.
- CANON : Dépressurisation d'une section tubulaire de 100 mm de diamètre et 4 m de long, à partir d'une pression de 30 bars.
- SUPER-CANON : Même expérience pour une pression initiale de 150 bars.
- DEDIF : Développement d'instrumentation pour écoulement diphasique.

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Etat de l'étude :

1) Avancement à ce jour :

Campagne d'essais de dépressurisation OMEGA sur section tubulaire terminée. Dépressurisation sur CANON 30 bars terminée.

2) Résultats essentiels :

Mise au point de mesure de taux de vide par neutronographie. Test du code BERTHA sur les dépressurisations CANON. Etalonnage des débitmètres pour écoulement diphasique pour des pressions supérieures à 40 bars et établissement de la procédure de dépouillement des indications de ces débitmètres lors de la dépressurisation.

Prochaines étapes :

Dépressurisation d'une grappe de 36 barreaux. Dépressurisation sur CANON 150 bars.

Documents de référence :

- "Dépressurisation d'une capacité en double phase - installation CANON", B.RIEGEL, A.MARECHAL, JC.ROUSSEAU Note DTCE-STT 490.
- "Void Fraction Measurements during Blowdown by Neutron Absorption or Scattering", JC.ROUSSEAU, J.GERNY, B.RIEGEL - Meeting OCDE, TORONTO, august 3-4 1976.
- "CODE BERTHA-Application aux expériences CANON", JC.ROUSSEAU, G.BOUDSOCQ, A.MARECHAL, B.RIEGEL, M.SCHALL - Note DTCE-STT 491.

140-26-21/4160-04

Classification : 1-1-2 215

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Titre	Pays :
Thermohydraulique du LOCA. Etude des débits critiques en double phase : Programmes MOBY-DICK et SUPER MOBY-DICK.	FRANCE Organisme directeur : CEA-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 : M.COURTAUD (STT)
Date de démarrage : 01/01/72 Date prévue d'achèvement :31/12/80 Etat actuel : en cours Dernière mise à jour : 21/01/77	Scientifiques : HOUDAILLER (EdF-SEPTEN) J.C ROUSSEAU M.LAURO

Objectif général :

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

Objectifs particuliers :

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

Installations expérimentales et programme :

MOBY-DICK : Boucle dans laquelle est réalisée un mélange double phase par autovaporisation. Plusieurs géomètries 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, $X < 2\%$). Il est prévu de modifier la boucle 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.

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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.
Reconstruction de la boucle : La puissance passe de 360 kW à 800 kW et le condenseur a été changé.
SUPER MOBY-DICK : Stade de l'avant-projet.

2) Résultats essentiels :

Valeurs de débits critiques dans la gamme 1 à 7 bars, $X < 1\%$.
Mise en évidence de l'importance du déséquilibre thermodynamique.

Prochaines étapes :

MOBY-DICK : Expérience à réaliser sur une section d'essai de 14 mm de diamètre terminé par un divergent de 7 degrés, $P < 10$ bars, $X < 12\%$. Expériences réalisées sur une section d'essai de 14 mm de diamètre avec un élargissement brusque de 120 mm, $P < 10$ bars, $X < 12\%$.
SUPER MOBY-DICK : Boucle opérationnelle prévue pour septembre 1978.

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.

Documents de référence :

"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.

140-31-10

Classification : 1-1-2

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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) J. AZAM (EDF)
Date de démarrage : 1/01/78 Date prévue d'achèvement : 1/07/79 Etat actuel : Construction Dernière mise à jour : 15/04/77	Scientifiques : M. REOCREUX (CEA) G. HOUDAYER (EDF)

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 des installations expérimentales. Les sections de fuite étudiées seront:200, 300 et 500 mm de diamètre.

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 s'est tenue à MARVIKEN en avril 1976 . La durée des essais est de 18 mois .

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L'accord concernant les essais sera signé d'ici le 15 mai 1977.

Cet accord interesse :

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

Etat de l'étude :

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/TAP
Initiated octobre 1973	Completed Decembre 1977	Scientists H. BLANC-FERAUD H. SUREAU
Status	Last updating : 20.01.75	

I - GENERAL AIM

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

II - PARTICULAR OBJECTIVES

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

III - EXPERIMENTAL FACILITIES AND PROGRAMME

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

a) Essais sur maquettes "froides", Plate-forme d'essais Pompes Turbines de CHATOU. La maquette de type 93 D d'échelle géométrique 1/3,2 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 JUEV 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/[illegible]
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.

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IV - PROJECT STATUS

4.1 - Progress to date

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

4.2 - Essential Results

V - NEXT STEPS

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

VI - RELATION WITH OTHER PROJECTS

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

. Code CLYSTERE.

VII - REFERENCE DOCUMENTS

Néant.

VIII - DEGREE OF AVAILABILITY

Rapport d'essais d'accès libre.

CLASSIFICATION
1.1.2

TITLE 1	MELANGE DANS LA CUVE D'UN PAWR DES ECOULEMENTS PROVENANT DES DIVERSES BOUCLES.	COUNTRY FRANCE SPONSOR E.D.F./SEPTEN ORGANIZATION E.D.F.
TITLE 2	MIXING OF THE FLOWS OF THE DIFFERENT LOOPS ENTERING THE REACTOR VESSEL.	Project Leader E.D.F./DER/LNH
Initiated Mai 1974	Completed mai 1975	Scientists M. PUGNET M. LARRIEAU
Status Etude en cours	Last updating : 20.01.75	

I - GENERAL AIM

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

II - PARTICULAR OBJECTIVES

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

III - EXPERIMENTAL FACILITIES AND PROGRAMME

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

IV - PROJECT STATUS

Etude en cours.

V - NEXT STEPS

Fin des essais en janvier 1975.

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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)

1. General aim

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

2. Particular objectives

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

3. Experimental facilities and programme

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

4. Project status

4.1 Progress to date

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

5. 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. Kjellen (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)	Classification
Transitori termoidraulici in reattori a tubi in pressione durante lo svuotamento	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 UTM (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.

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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 I976 Date completed June I977 Last updating March I977	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 I977.

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) Instability phenomena related to steam relief through S.R.V.	Country: ITALY 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) PROGRAMMA P.I.P.E.R.: esperienze di blow-down in presenza di strutture interne.	Classification 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

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
1.1.2.

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.

10) Personnel : 3 men/year

11) 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	Completed : 31.12.1975	
Status : progressing	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 computer programma ter bestudering van uitstroming	COUNTRY: NETHERLANDS. SPONSOR: ECN ORGANIZATION: ECN
TITLE: (ENGLISH LANGUAGE): CHARME: A computer program to study blowdown	PROJECTLEADER: Speelman, J.E.
INITIATED: April 1976 / LAST UPDATING: April 1977 STATUS: progressing / COMPLETED: 1980	SCIENTISTS: Putten, A.P.W.M. van der Bogaard, J.P.A. van den Koning, H

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 function 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.

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: -

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Netherlands Energy Research Foundation (ECN)	CLASSIFICATION 1.1.1. & 1.1.2.
TITLE: Mechanisch gedrag van het reactorbinnenwerk tijdens grote ongelukssituaties	COUNTRY: 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.

contd...

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.

<u>Reference Documents</u>

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)	<u>Project Leader</u> Prof. W. B. Hall
Initiated October 1975 Status progressing	<u>Scientists</u> J. T. Turner G. P. Ioannu

1,2. General Aims and Particular Objectives

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

3. Experimental facilities

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

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

4. Project Status

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

It is anticipated that detailed experimental work will commence shortly.

contd.

5. Next steps

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

6. Relation with other projects

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

7. Reference documents None

8. Degree of availability

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

7. Reference documents

None

8. Degree of availability

On application to the NII when available.

1. Budget

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

2. Personnel

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

3. Additional information

Time schedule: Stage 1 planned completion Oct 1976.

Stage 2 Construction during 1976.

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

1. 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

Classification: 1.1.4	
Title 1 (Original Language): Bestimmung der Nachzerfallswärme von 235-U im Zeitbereich 10 - 1000 sek (PNS 4234 - I.1.5, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: -- ORGANIZATION: GfK Karlsruhe Projekt Nukleare Sicherheit
Title 2 (English): Decay Heat Measurement of 235-U in the time-period from 10 to 1000 seconds	Project Leader: K. Baumung/INR
Initiated (Date): Sept. 1974 Status: Continuing	Completed (Date): 1977 Last Updating (Date): December 31st, 1976

1. General Aim

LOCA-Analysis

2. Particular Objectives

Providing decay heat data for LOCA calculations and experiments.

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 providing a good cooling of the samples during irradiation was constructed. It is connected to a computer-controlled adiabatic calorimeter. The energy loss due to γ-escape is recorded by a Moxon-Rae-type detector.

Monte-Carlo-codes for the detailed calculation of the burn-up profiles, the γ-escape-characteristics and heat source distributions of the fuel samples were provided

5. Progress to Date

The experimental facilities were constructed, the shielding from γ- and delayed neutron radiation was designed and measures taken against emission of radioactivity, especially of J-131.

6. Results

7. Next steps

The next steps will be the measurement of the heat capacities of the samples and the irradiation experiments at the FR2 reactor.

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

1. General aim

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

2. Particular objectives

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

3. Experimental facilities and program

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

4. Project status

4.1. Progress to date

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

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

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

4.2. Essential results

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

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

5. Next steps

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

6. Relation with other projects

This experiment yields basic data for the LOCA analysis.

7. Reference documents

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

8. Degree of availability

Unrestricted distribution.

Title 1 (Original language)	Classification
Burn-out in reverse flow	I.I.4
Title 2 (English)	Country ITALY Sponsor CNEN Organisation CNEN
Date initiated I975 Date completed I977 Last updating March I977	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.I) freon I2	L = 200 cm D = 4.75 mm
2.2) water	L = 400 cm D = II.5 mm

3 - Experimental facilities

A S.S.water loop (pressure up to I60 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)	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 1.2

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.	

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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

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Blowdown, Refill and Reflood
FLECHT-SET

7. REFERENCE DOCUMENTS

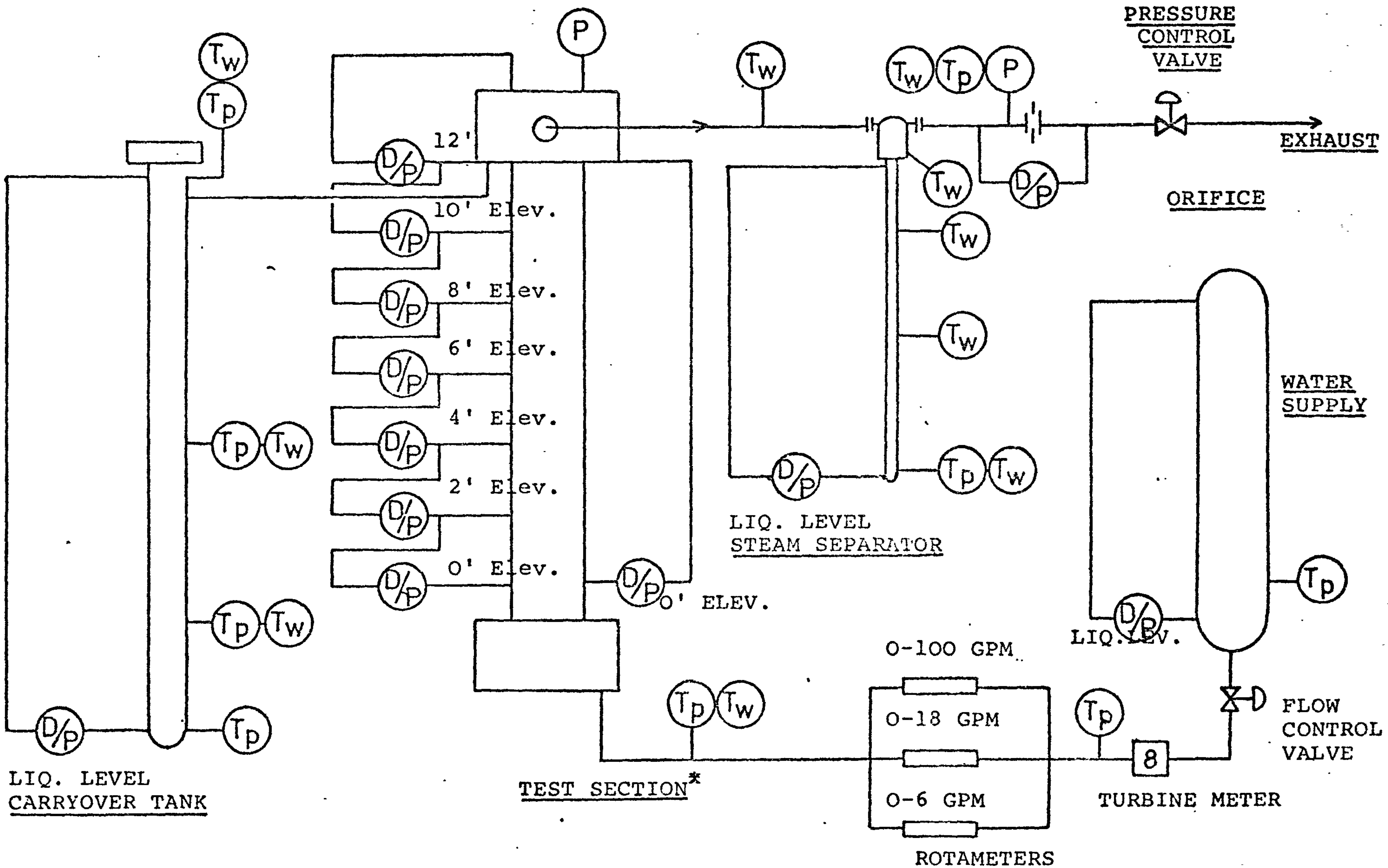
a) WCAP-7665 - PWR FLECHT Final Report, April, 1971.

b) WCAP-7931 - PWR FLECHT Final Report Supplement, October 1972.

8. DEGREE OF AVAILABILITY

Available upon request.

FLECHT[illegible] FLOODING RATE TEST CONFIGU[illegible]N



* ALL INSTRUMENTATION IS NOT SHOWN

Classification 1.2	
Title 1 FLECHT SET Full Length Emercency Cooling Heat Transfer Systems Effect Tests.	COUNTRY Belgium (USA) SPONSOR ORGANIZATION : Westinghouse Nuclear Europe
Title 2	PROJECT LEADER SCIENTISTS :
Initiated (date) 7/30/74 Completed : Status : Last updating	

FLECHT : FLECHT-SET

(Full Length Emergency Cooling Heat Transfer System Effects Tests)

1. GENERAL AIM

Following a primary system loss-of-coolant accident, the system would rapidly depressurize. The loss of coolant may partially or wholly uncover the reactor core. The Emergency Core Cooling System is provided to rapidly reflood the reactor vessel under such conditions, and ensures that any damage to the core does not lead to any unacceptable consequences either in the plant or off-site.

The original FLECHT series of tests were designed as separate effects type tests to investigate the reflood heat transfer history of hot fuel rods in the core during the reflood phase of a LOCA. The reports of this series of tests are given in References 1-4.

2. PARTICULAR OBJECTIVES

FLECHT-SET is a continuation of the FLECHT bottom flooding test except that the effects of the system volumes, resistances, elevations and other heat inputs are modeled to obtain the system feedback on the flooding rate and heat transfer. The program will consist of two phases. Each phase is intended to simulate a 4 loop PWR with various degrees of sophistication. Details on each are included in subsequent sections.

3. EXPERIMENTAL FACILITIES AND PROGRAM

Experimental facility is illustrated by figure 1 and is described in references 5 and 6.

The program is divided in 2 steps :

- PHASE A consisting of scoping tests (1 loop no steam generator)

- PHASE B including a more complete systems effect simulation (2 loop steam generator simulation).

4. PROJECT STATUS

4.1. Programs to-date

Phase A consisted of a set of early scoping tests employing a simplified 1 loop system simulation without a steam generator (long lead item). The simplification (1 loop representing 4 loops) is considered necessary in order to measure flood rate and particularly the test section effluent two phase flow rate. Without the steam generator producing single phase flow at its exit, this is not measurable with standard orifice measuring techniques. Hence a simple system devised which separates, collects, and measures test section liquid effluent, then heats the remaining steam to saturation or above, thereby allowing a meaningful single phase orifice flow measurement. The liquid carryover is separated and collected at a measured rate (at the steam generator location) prior to passing through the largest flow resistance of the loop. A high quality mixture ($x > .95$) then enters a 24 ft. length of heated pipe where any remaining liquid is vaporized prior to passing through the loop orifice. Since the flow through the calibrated orifice is single phase, the flow rate can be determined by measuring the pressure drop and upstream temperature and pressure. A total effluent flow rate and quality can be calculated from the collection rate of liquid and the flow rate through the orifice.

The test in this configuration are complete and a data/analysis report has been issued (reference 5). The general result found from these tests was that the variable flow into the test assembly, caused by the system response during reflooding, yielded higher heat transfer than that which would be calculated using the FLECHT heat transfer correlation and the calculated flooding rate.

Phase B is intended to be a more complete systems effect simulation of a PWR 4 loop plant and 1 broken loop and 3 unbroken loops, including steam generator heat addition and elevation effects. Since the steam generators superheat the test section effluent, meaningful.orifice flow measurements can be made downstream of the steam generators using the loop orifice. The FLECHT-SET phase B loop drawing is given in Figure 1. The system is described in detail in reference 6.

A total of 35 phase B tests have been completed including facility shakedown tests and repeat tests. Of these tests, 20 will be reported in a data report and will be separately analyzed in a data evaluation report.

4.2. Essential Results

The same general trends observed in Phase A were also observed in Phase B ; the variable bundle flooding rate resulted in higher heat transfer than that calculated by the FLECHT correlation.

Several questions have been raised on the scaling logic used to design the FLECHT-SET facility. The AEC critically reviewed the facility and has issued a task force report on the facility. In general, they either agreed with the design or suggested modifications which would make the scaling logic more exact. The AEC was particularly concerned about the observed large oscillations which occurred at the beginning of reflood. The Phase A data indicated that the large oscillations were caused by the rapid heat release from the test section housing. Since the rate of heat release could not be controlled from the housing, (although the time integral of the heat release could be controlled), the majority of the Phase B tests were conducted with the housing heated to the fluid saturation temperature such that the housing heat release was minimized.

5. NEXT STEPS

With the issuance of the new ECCS criteria, the AEC has re-evaluated its reflooding heat transfer requirements and has requested that the systems effects tests stop and that the FLECHT-SET facility be converted into a forced flooding heat transfer facility such that specific reflood heat transfer questions identified in the new criteria could be examined. The FLECHT-SET testing has stopped and the facility is being converted to a forced flooding mode of operation and tests in this configuration are scheduled to begin in December 1974.

6. RELATION WITH OTHER PROJECTS

This program was in the line of other ECCS programs on the post blowdown phenomena like FLECHT, STEAM WATER MIXING...

7. REFERENCE DOCUMENTS

1. J.O. Cermak, A.S. Kitzes, F.F. Cadek, R.H. Leyse, and D.P. Dominicis, "PWR Full Length Emergency Core Heat Transfer (FLECHT) Group 1 Test Report", WCAP-7435, January 1970.

2. F.F. Cadek, D.P. Dominicis, and R.H. Leyse, "PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group II Test Report", WCAP-7544, September 1970

3. F.F. Cadek, D.P. Dominicis, and R.H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report", WCAP-7665, May 1971.

4. F.F. Cadek, D.P. Dominicis, H.C. Yeh and R.H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report Supplement", WCAP-7931, September 1972.

5. J.A. Blaisdell, L.E. Hochreiter, J.P. Waring, "PWR FLECHT-SET Phase A Report", WCAP-8238, December 1973.

6. W.F. Cleary, et, al., "FLECHT-SET Phase B System Design Description", WCAP 8410, 1974.

8. Degree of availability

Available upon request.

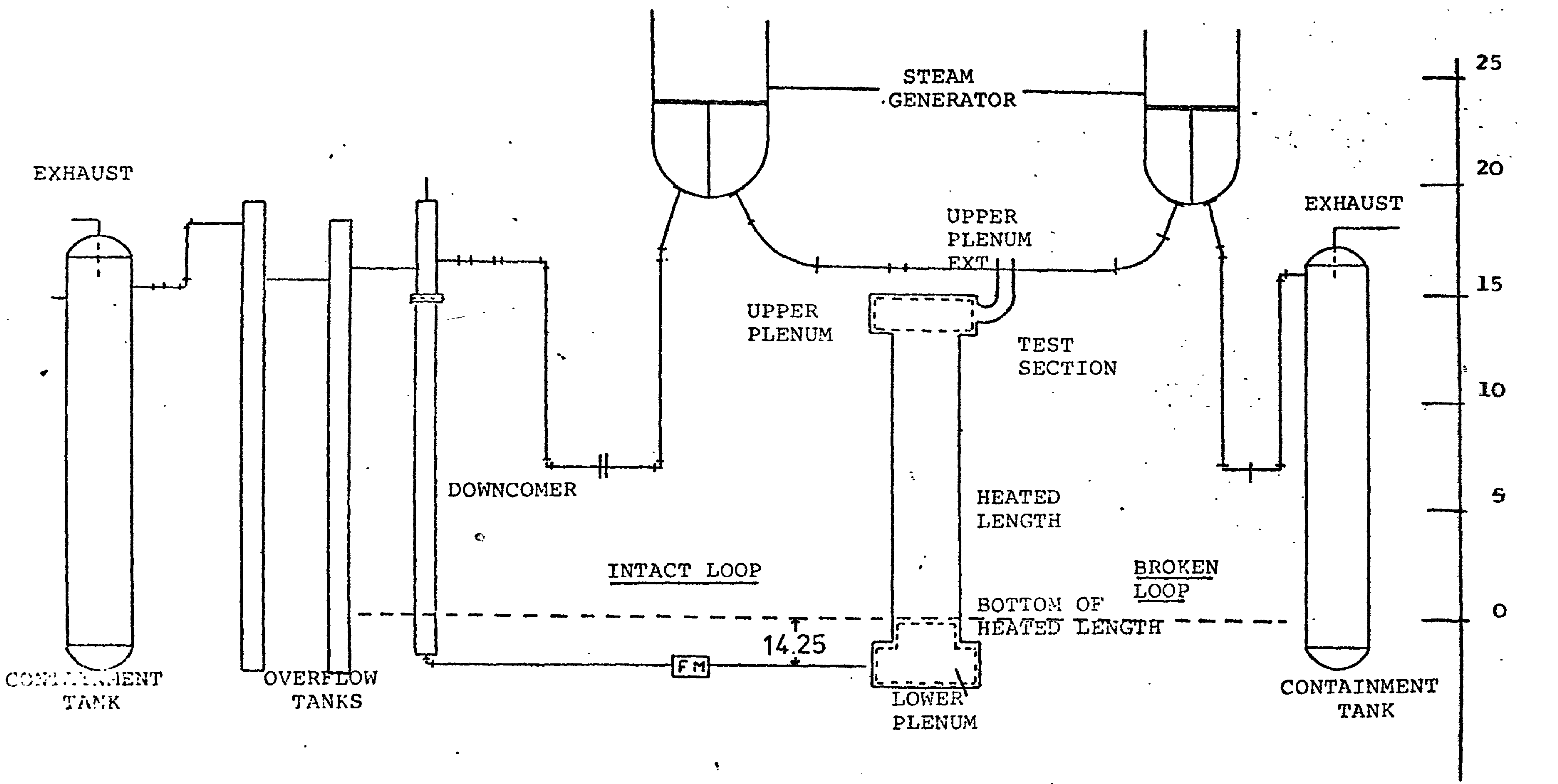


FIGURE 1

FLECHT-SET PHASE B

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Classification 1.2	
Title 1 Steam Water Mixing Tests.	COUNTRY Belgium (USA)
	SPONSOR
	ORGANIZATION : Westinghouse Nuclear Europe.
Title 2	PROJECT LEADER SCIENTISTS
Initiated Completed Status Last Updating	

1. GENERAL AIM

During a LOCA, the effects of venting steam with cold water accumulator and safety pump injection are not quantified. In order to calculate the steam flows vented through the cold leg, the effective resistances must be determined experimentally.

2. PARTICULAR OBJECTIVES

The AEC interim criteria states in part :

1. "No steam flow shall be permitted in intact loops during the time period that accumulators are injecting".

2. "All effects of cold injection water, in either a hot or cold leg, in steam flow (and ΔP) should be included in the calculation".

The intent of the steam/water mixing program is to relax these overly conservative design criteria by obtaining pressure drop data during cold water injection for use in blowdown and reflood codes.

3. EXPERIMENTAL FACILITIES AND PROGRAM

Tests were conducted at approximate conditions expected to exist during and after blowdown. Table 1 presents a list of the important parameters and their ranges.

The test sections represent scaled segments (length to diameter ratio is constant) of the piping between the reactor coolant pump and the reactor vessel. The full PWR primary coolant loop resistance is also simulated.

Surge tanks at either end allow a constant pressure drop to be set across the loop, representing a fixed downcomer head. The steam flow resulting from this fixed driving force was measured. A typical test setup is pictured in Figure 1.

The effect of scale was studied to extend the test results to a full scale PWR. Tests have been run at 1/14 and 1/3 scale. Tests were also run with and without the full length cold leg extension pictured in Figure 1 for the 1/3 scale test section.

Instrumentation included density measurement by a low energy X-ray attenuation technique, as well as temperatures, pressures and pressure drops.

This work was performed by Westinghouse at the Canadian Westinghouse Laboratories in Hamilton, Ontario, Canada. This program has been submitted to EPRI (Electrical Power Research Institute) for cooperative funding.

4. PROJECT STATUS

Progress to-date and essential results.

A series of tests have been completed at 1/14 scale with injection angles of 90°, 60° and 45° in both the accumulator and SIS phase of reflood. Test section pressure drops in the accumulator range can be predicted reasonably well with a simple model based on one-dimensional momentum considerations. For 90°, the effect of accumulator injection is to decrease test loop steam venting capability by 5 to 30% from the no-injection case. For 45° injection, the steam venting capability is increased due to the pumping action of the angled injection. For the SIS range of flow rates,

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cold leg injection has a very minor effect on overall loop resistance.

The 1/14 scale simulated blowdown tests have been performed and the pressure drop data was found to agree reasonably well with the one-dimensional momentum prediction. Density measurements indicated that the two-phase flow was nearly homogeneous during the higher pressure blowdown tests.

Density measurements have also been obtained for both the high ECC flow (accumulator) and low ECC flow (pumped injection) portion of the reflood transient. The pressure oscillations which were observed on the pressure transducers was found to be caused by oscillating flow. The oscillating flow behaviour was observed on both the density and thermocouple readings. The 1/14 scale report shall be issued shortly.

The 1/3 scale tests and data analysis is complete and the report is presently being published. The 1/3 scale tests showed similar behavior but more scatter as compared with the 1/14 scale data. The same model which was used to represent the steady cold leg pressure drop data for the 1/14 scale tests will also represent the 1/3 scale data if the upper bound limit is increased to 1 psia. Scale effects were observed in the 1/3 scale tests, however, they can be included in the 1 psia upper bound on the data.

5. NEXT STEPS

EPRI has indicated that they would require additional testing, these requirements are now being determined.

6. RELATION WITH OTHER PROJECTS

This program is related to all other ECCS programs that aim to a better understanding of the post blowdown transient such as FLECHT, FLECHT-SET ...

TABLE 1

COLD LEG STEAM/WATER MIXING TESTS

Parameter	Range
System Pressure	45 to 20 psia
Cold Leg Steam Velocity	50 to 400 ft/sec
Cold Leg Steam Quality	60% - 300 psia to 550°F - 20 psia
Water Injection Velocity	1 to 90 ft/sec
Accumulator Water Temp.	80 to 150°F
Water Injection Angle	45° 60° 90°

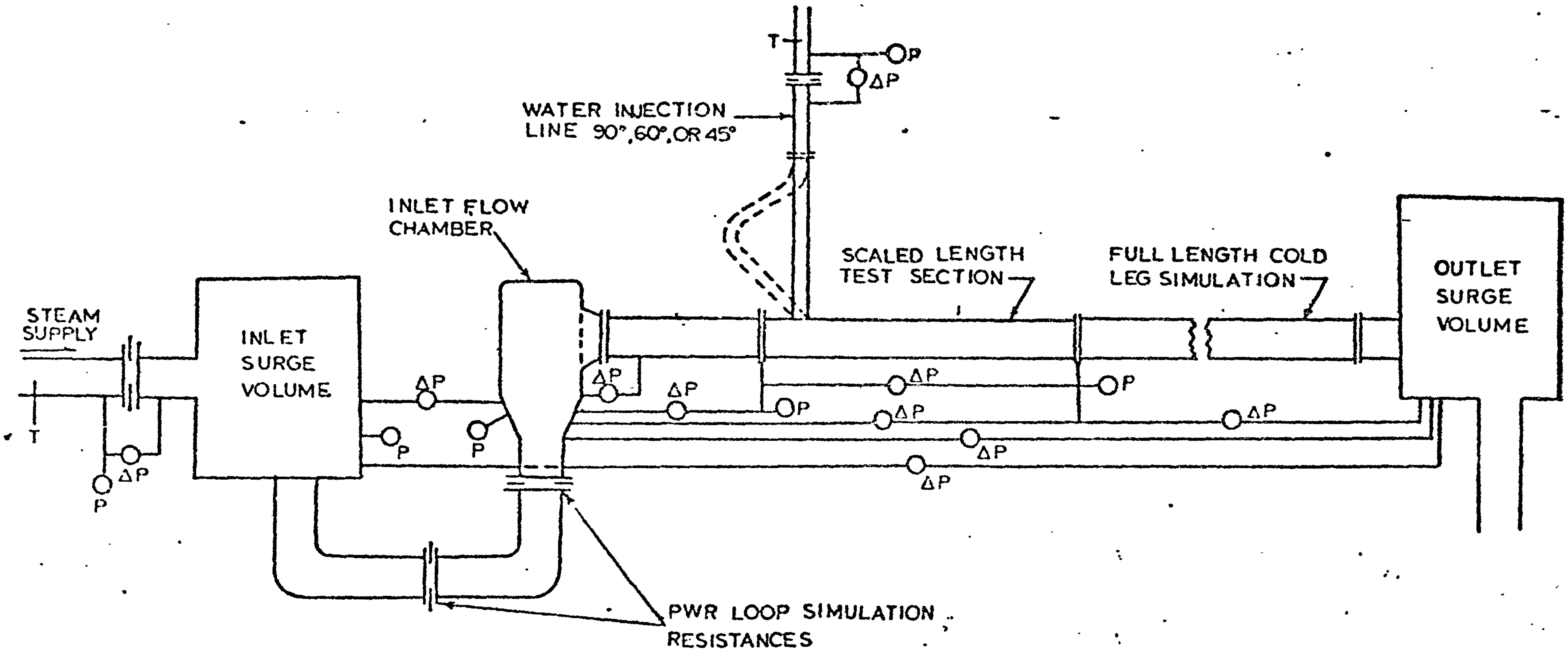

FIGURE 1
STEAM-WATER MIXING TEST CONFIGURATION
SHOWING PRESSURE AND FLOW INSTRUMENTATION

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

General Aim and Particular Objectives

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

Experimental Facilities and Program

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

Project Status/Progress to Date

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

Project Status/Essential Results

The calculated heat transfer coefficients were in good agreement both from the difference of the measured tube and saturation temperature of the fluid and the values from the steam convection, as long as the steam was not overheated, that means the axial

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measuring position must not be too far away from the swell level.

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

Next Steps

Work on this project has been completed.

Relation with Other Projects

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

Reference Documents/Degree of Availability

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

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

Classification: 1.2	
Title 1 (Original Language): Notkühlprogramm - Niederdruckversuche Wiederauffüllversuche mit Berücksichtigung der Primärkreisläufe (RS 0036 B - I.1.2, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** KWU, Erlangen
Title 2 (English): Emergency Core Cooling Program-Refilling Experiments with Simulation of the Circulation Loop	**Project Leader:** Ruthrof
Initiated (Date): 1. 1. 73 **Completed (Date):** 31. 5. 77 **Status:** Continuing **Last Updating (Date):** 31. 12. 76	

1. General Aim

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

2. Particular Objectives

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

3. Research Program

The test program includes the following parametric variations:

- Max. initial clad temperature: 500 to 800 °C
- System pressure: 1 - 6 (40) bars
- Max. decay heat flux: 4 to 8 W/cm^2
- Time function of decay heat: const, ANS standard
- Reflood rates: 6 - 60 cm/sec
- Split of reflood rates top/bottom: 0/1, 1/1, 2/1, 1/2
- Time function of injection rates: Const, accumulator characteristic
- Break size: 0,25 to 2 F (double ended guillotine)
- Break location: hot leg, cold leg, between SG- and pump

- Simulated pump resistance: locked rotor, free rotor
- Residual water: 0 to lower grid plate
- Loop seals: 0 to lower grid plate
- Loop seals: 0 to full
- Loop wall temperatures: 150 to 300 °C

4. Experimental Facilities

In order to investigate the reflood phase in a PWR including the feedback of the complete primary system, a test facility is being build. Beside a 340 rod-testbundle it includes three scaled down primary loops with full height steam generator simulation.

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 data channels at 1 Hz scanning rate.

5. Progress to Date

The test facility was built and the test bundle was installed. The instrumentation was completed and calibrated. Thermocouples were mounted. The volumina of the facility were measured. First tests were run with 25 % and 50 % bundle power. The initial system pressure was 1,4 bar, the wall temperature of the rods 400 °C. The following shake down tests IBS-2, 3 and 4 were run at higher power. The flooding rate was about 5,5 cm/s.
The values for the hot- and cold injection were adjusted.

After every test a preliminary evaluation was started, the results from the 300 instrumented points were recorded. The program for the drum-plotter was completed, the measured values were transformed to physical values and plotted automatically.

The subroutine for the control of the water injection and the switching of the values was programmed and tested.

6. Results

The test results showed that some modifications were necessary in the test loop. Difficulties arose from the heating of the model steam generator and the safety valves, which opened at 52 bar instead of 60 bar. The function and the interaction of all components were studied successfully.

7. Next Steps

- Preliminary tests to determine the condensation efficiency in the upper plenum and the cold leg injection point.
- Start with tests with a hot and cold leg break
- Programming of the mass and energy balances

8. Relation with Other Projects

RS 36 C	Emergency Core Cooling Program Low Pressure Experiments. BWR-Second Cluster
RS 37 C	Emergency Core Cooling Program PWR Post DNB Experiments with a bundle of 25 rods

9. References

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10. Degree of Availability

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

General Aim

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

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

Particular Objectives

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

Experimental Facilities and Program

The research program RS 36 includes the 3 steps:

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

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

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

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

Project Status/Progress to Date
Experimental investigations of the series "A" were carried out, encircling the following parameters:

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

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

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

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

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

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

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

About 20 % of the sprayed water wetted the containment and not the bundle. The water level was kept constant about 300 mm beneath the heater rods. Two tests were carried out with joint spraying and flooding.

Project Status/Essential Results

The test series "A" and "B" are completed. The results are plotted and stored.

The result of the tests on $c_p \cdot \rho$ was lower than expected theoretically (~ 5 %).

The test series "C" are completed and evaluated. During spraying the power was decreased about 40 % when a temperature of 925 °C was reached. The result was, that the heat transfer during spraying is better at higher pressure load compared with 1 bar.

Next Steps

Additional tests as proposed by the SK "Notkühlung" : comparative tests, evaluation of heat transfer coefficients from the test results, investigation of parameter influences.

Relation with Other Projects

see RS 36.

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Reference Documents/Degree of Availability

E. Hicken, K. Riedle
"Bubble rise velocity and heat transfer in a vertical rod bundle"
Paper presented at the Meeting of the "European Two-Phase Flow Group"
at Haifa, June 2 - 6, 1975

H.-P. Gaul, E. Hicken
H. Loser, K. Riedle
K. Ruthrof, J. Sarkar
"Wärmeübergang an stagnierendes Fluid in zwei SWR-Bündeln
unterschiedlicher Leistung"
Paper presented at Gemeinsame Fachtagung der Fachgruppen "Reaktor-
sicherheit" und "Thermo- und Fluiddynamik" der Kerntechnischen
Gesellschaft. 28. - 30. Januar 1975

K. Riedle
"German reflood heat transfer program"
Presented at the 3rd Water Reactor Safety Research Meeting
Gaithersburg, Md. Sept. 29, 1975

Classification: 1.2	
Title 1 (Original Language): Theoretische Untersuchungen zur Niederdruck- und Wiederauffüllphase der Kernnotkühlung (ATT 085 A - I.1.2 , Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (English): Low Pressure and Refilling Phase of Emergency Core Cooling	Project Leader: Dr. H. Karwat Dr. A.B. Wahba
Initiated (Date): Completed (Date): Status: continuing Last Updating (Date): December 1976	

1. General Aim

The general aim of this project is the development of computer codes to predict the thermal and hydraulic response of water cooled reactors during the refilling phase of a loss-of-coolant accident (LOCA).

2. Particular Objectives

One main objective is to develop a code which is able to simulate efficiently the flooding of the core and the various physical phenomena involved in this process.

Besides this development the application of existing codes is continued in order to indicate which particular problem will need special consideration in the new code.

From the analytical verification of related experiments additional information is expected to support the analytical work.

3. Experimental Facilities and Program

Not relevant.

4. Project Status

4.1 Progress to Date

Applications of the US Code RELAP4 are completed /1/. The behaviour of a simulated pressurized water reactor at the end of blowdown and during cold water injection was studied. The two versions RELAP4-002/8 and RELAP4-003/85 were used.

The parameter study using the refill program WAK-1 is completed. The hydrodynamic results were used to estimate the thermal behaviour of an average pin during the refill and reflooding process. For this estimation the US fuel element thermal analysis program TOODEE2 (1) was used. The core reflooding rate and the steam production rate above the water level was obtained from WAK-1, the temperature response of an average pin from TOODEE2 through activation of the appropriate heat transfer correlation.

The preliminary version of a refill and reflood program FLUT has been developed. The version FLUT-BV (basis version) simulates mainly the processes taking place within the pressure vessel. It consists of a fuel rod model, a quench model and a hydraulic model. The fuel rod model is one-dimensional in radial direction and includes up to 100 axial segments. For the decay heat the ANS standard curve plus 20 % is used. Different flow regimes are considered for the heat transfer model. The motion of the quenchfront is determined using the analytical equation from Duffy and Porthouse. The formation of an upper quenchfront through the water injection in the upper plenum is considered. The motion is con-

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(1) G.N. Lauber
TOODEE2. A two-dimensional time dependent fuel element thermal analysis program
NRC, May 1975

trolled by using the Yamanouchi correlation. In the hydraulic model a quasi-steady state approach on the basis of the energy, momentum and mass balances is used. Separation between swell and collapsed level in core is activated using the Wilson correlation for bubble rise. Carry-over criteria are considered in the dispersed flow reqime. Time averaqed values for the U-tube fluid motion between the core and downcomer are used.

In order to analyze the rewetting experiments performed in connection with the BMFT Research Projects RS 62 and RS 184, program modifications were made. The time-dependent local heat transfer coefficients can be predicted from the measured temperature using the program INSTHTC /2/. This model contains a numerical solution for the nonlinear inverse heat conduction problem of estimating surface temperature and heat flux utilizing a measured temperature history inside a heat conducting solid.

4.2 Essential Results

A basic version of the reflooding program FLUT has been developed which is in the stage of final testing. First results of calculations performed with this code showed the necessity of further improvements with respect to dynamic effects and to boundary conditions prevailing in relevant experimental facilities which serve for code verification.

5. Next Steps

A detailed description of the FLUT-BV will be prepared together with small improvements of the computer code. Modifications have to be made in order to use the code for analysis in connection with the RS 36 refill and reflood experiments.

An improvement of the fuel rod model through the use of an implicit integration will be done to save computation time. A quench model will be developed together with a heat rod model which can be verified by simulating the experiments carried out in connection with RS 184. The hydrodynamic model of FLUT-BV will be improved through simulation of the dynamic U-tube oscillations. Additional subroutines for a heat exchanger model, a pump model and models for additional effects in the upper and lower plenum will be included in FLUT. The improved version of FLUT will be verified using the loop experiments in connection with RS 36 B.

6. Relation with Other Projects

Our code development and verification has strong relations to the refilling and reflooding experiments performed by KWU Erlangen (RS 36, RS 62, and RS 184).

7. Reference Documents

/1/
K.J. Liesch, V. Teschendorff, A.B. Wahba
Erfahrungsbericht über die Anwendung von RELAP5-003/85. Interner Bericht MRR-I-70, August 1976

/2/
A.B. Wahba
Berechnung von zeitabhängigen Wärmeübergangszahlen während der Wiederbenetzung von hoch aufgeheizten Rohren
MRR 167, November 1976

/3/
A.B. Wahba, A. Berning
Quenching Phenomenon in Water Reactor Emergency Core Cooling
Trans. Am. Nucl. Soc. 23, p. 285, 1976

/4/
Quarterly reports in the series IRS-Forschungsberichte

8. Degree of Availability

Documents are available through
Gesellschaft für Reaktorsicherheit mbH
D-8046 Garching, Forschungsgelände

Federal Republic of Germany

Reports MRR-I are confidential and therefore normally not available.

Classification: 1.2	
Title 1 (Original Language): Untersuchungen zur Hydraulik des Flutvorgangs und zu bisher noch unberücksichtigten Einflußgrößen beim Wiederbenetzen (RS 184 - I.1.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU Erlangen
Title 2 (English): Investigations on the influence of the hydraulic during reflooding	Project Leader: H. Hein
Initiated (Date): 1. 10. 75 Status: Continuing Completed (Date): 30. 9. 77 Last Updating (Date): 31. 12. 1976	

1. General Aim

In order to improve the reflood model, the hydraulic effects during reflooding will bei studied in detail.

2. Particular Objectives

A detailed knowledge of the hydraulics in a channel during the reflooding and the resulting flow pattern shall improve the calculation of heat transfer in the unwetted area. With the help of this experiment criteria will be worked out for the transition of vapour - to fog flow and from fog flow - to film boiling in order to get more information of the extension of different heat transfer regions.

The coupling of the rewetting model based on heat conduction in the wall with the hydraulics of the channel is a presupposition for a general applicability of theoretical calculations.

3. Research Program

For three different hydraulic diameters rewetting experiments will be carried out variing the parameters initial wall temperature, inlet subcooling, supply velocity and system pressure. Also the ratio stored heat to the water content within the channel will be varied.

With the annular test section a comparision will be made for the advancing of the rewetting front for Zircaloy and stainless steel cannings.

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 testrig used for the program RS 62 will be modified for getting more detailed information on the hydraulics during the reflood period. Also for these experiments constant inlet conditions will be focused.

For special tests an annular test section with a quarz-glass tube for the outer wall will be used.

5. Progress to Date

With the modified annular test section preliminary tests have been run in order to compare the advancing of the rewetting front between Zry-4 and stainless steel claddings. The reflooding tests were recalculated with a computer code.

Special investigations were started on the begin of the rewetting of the end of a hot rod. The time between submerging and rewetting was determined for the parameters

- initial temperature
- temperature of the reflood water
- reflood velocity

The reflooding tests were continued with tubes of 16,8 mm internal diameter (the first test were run with tubes of 13,8 mm). The velocity was varied between two and 6 cm/s, the initial temperature of the coolant between 35°C and 144°C and the heat flux density between 3 and 6 W/cm^2. The pressure was constant at 4,5 bar, the initial tube temperature was 600°C.

The reason for the appearance of new rewetting fronts was investigated with a 3m-tube, containing spikes on the internal surface.

In order to describe the hydraulics of the reflooding the thermodynamics non-equilibrium has to be known. For measuring the superheat of the steam, tests were carried out on a steam exhaust equipment which prevents disturbations of the measured steam temperature by dragged water droples.

A literature study was started for the description of the heat transfer during reflooding in the non-wetted-region.

6. Results

The rewetting front on a Zry-4 tube compared to a steel tube moves nearly a factor of two faster. As the heat conductivity is not very different, this effect can only be explained by the different heat capacities of the two materials. After reaching a fixed height, the rewetting velocity is nearly the same for both materials. Some tests were repeated with a copper-tube.

Recalculations of the rewetting tests showed good agreement between experimental and theoretical results for the circaloy-, steel- and copper-tube.

The time between submerging and rewetting was markedly lower when the water temperature and the initial tube temperature were low. The reflooding velocity, however had nearly no influence on the delay time.

New rewetting fronts appear when the heat removal of a spike is so large, that the rewetting temperature at that point is reached before the rewetting front has reached the perturbation zone.

7. Next Steps

The technique for the measurement of superheated steam will be improved.

A theoretical method will be developed for the calculation of the local steam content.

8. Relation with Other Projects

RS 62	Experiments for the Establishment of a Theory of the Rewetting of Highly Heated-up Rods by Pipe Experiments
RS 36	Emergency Core Cooling Program - Low Pressure Experiments

9. References

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10. Degree of Availability

Classification: 1.2	
Title 1 (Original Language): Notkühlprogramm - Niederdruckversuche SWR - 2. Doppelbündel (RS 0036 C - I.1.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Emergency Core Cooling Program - Low Pressure Experiments. BWR- Second Cluster	Project Leader: H. Hein
Initiated (Date): 1. 8. 74 — Completed (Date): 31. 12. 76 Status: Completed — Last Updating (Date): 31. 12. 76	

1. General Aim

The project belongs to tests, which were carried out under RS 36. Investigations were carried out on the bahaviour of a BWR-core during emergency cooling after MCA.

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

2. Particular Objectives

Informations about the rewetting velocity on the shroud and fuel rods. Temperature behaviour of the cladding at different water levels with and without steam injection.

Long term cooling of the rods with spray cooling.

3. Research Program and 4. Experimental Facilities

The research program RS 36 included the 3 steps:

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

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

Spacer geometry, heated length (3500 mm), pitch (18,75 mm), rod diameter (14,3 mm) inlet or outlet grid plate are designed identical to those of KWU BWR's.

The axial power distribution is approximated by a chopped cosine.

5. Progress to Date

15 additional tests were run, which were desired by the SK-Notkühlung. The parameters were:

Initial temperatures:	400, 600 and 800°C
System pressure:	5 and 10 bar
Power:	350/264, 300/200 and 200/150 kW.

During two tests the rod factor of the central rod A 18 was about 26 % higher

Flood velocity	≙ 1,5 and 3,5 cm/s
Spray mass	= 0,68 and 1,02 m^3/h per bundle

With one bundle four tests were run which served for comparison with the FLECHT-Program. The parameters were:

Initial temperature:	516 - 867°C
System pressure:	1 bar
Power:	100 and 200 kW
Spray mass:	0,556 m^3/h.

In order to calculate the heat transfer coefficients a program was written which allows the evaluation of the test series on the "Siemens 4004" computer.

After the double-bundle test the test section was dismantled, bundle "B" was completely disassembled.

The experimental data were evaluated.

6. Results

The tests showed that the influence of the initial temperature during flooding and spraying on the heat transfer coefficients is very low. The cooling however is better with higher coolant mass. Spray-cooling is sufficient, but during flooding the temperatures are quenched much faster.

The spray-tests showed that the wetting velocity and the heat transfer coefficient in the non-wetted region increase with higher pressures.

During the test runs 4 rods failed.

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 37 C Emergency Core Cooling Program
PWR Post DNB Experiments with a Bundle of 25 Fuel Rods

9. References

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10. Degree of Availability

Classification: 1.2

Title 1 Level Swell Reflood Heat Transfer	Country BRD (U.S.A) Sponsor Babcock and Wilcox Proprietary
Title 2	Organisation BBR Mannheim
Initiated: Jan. 1975 Completed: Status: Last updating: Dec. 1975	Project leaders Dr. B. E. Bingham R. T. Bailey

1. General aim

To determine reflood characteristics of a B & W "vent valve" plant and to evaluate the effectiveness of level swell cooling when the water inventory is insufficient to cover the core entirely with a two-phase froth.

2. Particular Objectives

3. Experimental Facility and Programme

In 1975, reflood and level swell tests were conducted with an electrically heated 56 tube full length bundle with an axial power profile peaked at the ten foot elevation.
Level swell experiments were conducted in two modes:

1) a constant water inventory is maintained by establishing a make-up flow equal to the steaming rate;
2) the water is allowed to boil off, progressively uncovering the bundle.

The ranges of system parameters covered by the investigation are:

Average Linear Power, kw/ft	0.1-0.3
Pressure, psia	20 - 180
Inlet Subcooling, $(h_f - h)/\frac{Btu}{lb}$	0 - 175

Thirty forced flooding experiments were conducted using this facility covering the following ranges of system parameters:

Peak Linear Power, kw/ft	0.5-1.0
Flooding Rate, in/sec	1.0-3.0
Pressure, psia	25-60
Inlet Subcooling, ^{o}F	0-140
Maximum Initial Temperature, ^{o}F	800-1400

4. <u>Project Status</u>

4.1 Progress to date

The experimental phase has been completed. Evaluation of the data is in progress.

4.2 Essential results

Level Swell: For a given core water inventory the swell level increases with power density and inlet enthalpy and decreases with pressure. Within the range of experimental variables, the Wilson bubble rise model predicts reasonably accurate void distributions. The computer code FOAM2, used to calculate the swell level and steaming rates during the quiscent period of small break LOCA's, has been shown to be an accurate formulation of the phenomenon.

Reflood: still under review.

5. <u>Next Steps</u>

During 1976, the level swell/reflood program will be continued using a single tube. System parameters affecting the reflood and level swell phenomenon that will be investigated are:

1) Hydraulic diameter of flow channel
2) Power profile (uniform, symmetric, inlet and exit peaks)
3) Chemical additives

6. Relation With Other Projects

7. Reference Documents

8. Degree of Availability

Proprietary
(It is intended to publicly release the level swell results during the third quarter of 1976).

Classification 1.2

Title 1	COUNTRY Denmark SPONSOR Risø ORGANIZATION Risø
Title 2 NORHAV - RHC a core heat-up computer program	Project leader: Jens G. Munthe Andersen
Initiated: November 1971 Completed: 1976 Status: progressing Last updating:	Scientists: Jens G. Munthe 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

   Available on exchange basis.

Titre	Pays : FRANCE
Thermohydraulique du LOCA. Etude des interactions mécaniques et thermo-dynamiques dans l'injection de secours d'un réacteur PWR : Programmes EPIS I et II.	Organisme directeur : CEA - EdF/SEPTEN
Titre (anglais) LOCA thermohydraulics. Steam-water mixing studies for PWR : EPIS I and II projects	Organisme exécuteur : CEA/DTCE (Saclay) Responsable : T.ELIAUX (SEEN)
Date de démarrage : 01/01/75 Date prévue d'achèvement : 31/12/78 Etat actuel : en cours Dernière mise à jour : 21/01/77	Scientifiques :

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.

Objectifs particuliers :

Developper des modèles physiques pour interpréter les expériences.

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).

Etat de l'étude :

1) Avancement à ce jour :

Essais EPIS 1 terminés. Installation EPIS 2 en construction.

2) Résultats essentiels :

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.

.../...

<u>Documents de référence</u> :

"Programme d'études des interactions mécaniques et thermodynamiques entre l'écoulement principal de vapeur et l'eau des injections de secours d'un réacteur PWR", Rapport SEEN-RT 76-014.

Classification : 1-2

140-26-23/4160.02

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Titre	Pays :
Thermohydraulique du LOCA : Etude expérimentale du refroidissement de secours des réacteurs à eau : Programme ERSEC.	FRANCE
	Organisme directeur : CEA
Titre (anglais) LOCA Thermohydraulics : Experimental investigation of water reactors safety injection : ERSEC project	Organisme exécuteur : CEA/DTCE - STT (GRENOBLE)
	Responsable : M. COURTAUD
Date de démarrage : 01/01/72 Date prévue d'achèvement : 31/12/78 Etat actuel : en cours Dernière mise à jour : 21/01/77	Scientifiques : R. DERUAZ

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.

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.

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.
Interprétation en cours des expériences : programme du front de trempe avec le modèle PSCHITT et corrélation d'échange en aval avec le modèle FLIRA.

2) Résultats essentiels :

Développement de modèles physiques représentant le rayonnement en aval du front de trempe.

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Prochaines étapes :

Essais de renoyage à débit constant sur tubes de différentes longueurs chauffantes.

Documents de référence :

"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.
"Refroidissement de secours des réacteurs à eau légère - essais de renoyage d'une grappe 5 X 5, géométrie 15 X 15", R.DERUAZ, P.CLEMENT, M.LAMBERT, P.PIC - Note DTCE-STT 509.
"Etude bibliographique des principaux modèles de remouillage utilisés dans l'étude du refroidissement de secours des réacteurs à eau - choix d'un modèle", P.CLEMENT - Note DTCE-STT 507.

Classification 1.2

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

1. But général

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

2. Objectifs particuliers

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

3. Installations expérimentales et programmes

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4. Etat du projet

a) code CERES

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

b) code FLIRA

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

5. Prochaines étapes

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

Interprétation des essais ERSEC

Couplage avec CERES

.../...

6. Relation avec d'autres projets

Essais sur la boucle ERSEC
Essais sur la boucle PHEBUS

7. Documents de références

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

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

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

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

  par LANGE et MEGNIN

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

Classification 1.2

Title 1 (original language)	COUNTRY : FRANCE
Etude expérimentale du refroidissement de secours des réacteurs à eau.	SPONSOR : C.E.A.
	ORGANIZATION C.E.A.
Title 2 (english) Expérimental study of the water reactor safety cooling	Project leader CEA/DTCE/STT M. DERUAZ Scientists :
Initiated (date) Completed : (date)	
Status : progressing Last updating (date) Janvier 1975	

1. But général

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

2. Objectif particulier

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

3. Installations expérimentales

Boucle ERSEC II - principales caractéristiques

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

4. Etat du projet

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

Interprétation:

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

.../...

5. Prochaines étapes

Essais :

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

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

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

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

6. Interprétation

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

Application au cas des grappes ....Courant 76

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

6. Relation avec d'autres projets

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

7. Documents de référence

Heat transfer during the reflooding of a tubular section

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

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

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

1. OBJECTIVES

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

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

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

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

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

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

2. PROJECT STATUS

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

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

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

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)	Country : ITALY Sponsor : CNEN Organisation : CISE
Bottom flooding ECCS	
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 thermo-hydraulic 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.	$\boxed{1.1.1}$ 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
1.2

Title 1	Country : JRC
Transient boiling heat transfer in emergency core cooling conditions	Sponsor : CEC
	Organization: JRC ISPRA Establishment
Initiated : 1974 Completed : December 1976	Project leader: E. Burck
Status : progressing Last updating : March 1975	

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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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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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

1.

Classification
1.1.
(1.2.)

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

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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

	Classification 1.2	
Title 1 PERFORMANCE OF SPRAY COOLING		COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION AEE WINFRITH
Title 2		Project Leader
Initiated 1968 Status :	Completed : 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 mde to optimise (speed up) quenching.

Reference Documents

Internal documents.

CLASSIFICATION 1.2

Title 1: P.W.R. Refill Studies Title 2: -

Initiated: 1st November 1975 Completed: -

Status: Progressing Last Updated: -

Country: United Kingdom Sponsor: UK - NII

Organisation: Strathclyde University

Project Leaders : H C Simpson, D H Rooney

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1. General Aims:

To simulate the refill process in a P.W.R. downcomer and study its effectiveness.

2. Particular Objectives:

To produce a theoretical model, or correlation, defining the refill process enabling the limiting conditions to be predicted.

3. Experimental Facilities and Programme:

Work to be carried out in three phases. Phase 1 is developed annulus with tangential water injection. Phase 2 is developed annulus with normal water injection. Phase 3 is 1/10 scale model of P.W.R. downcomer. All test sections transparent, fluids steam and water, pressures just above atmospheric. Measurements to be taken include steam and water flowrates, pressures, temperature distributions. Cine photography to capture liquid bridging effects.

4. Project Status:

Phase 1 data being collected.

5. Next Steps:

Production of theoretical model

6. Relation with Other Project:

Similar in some respects to the Wallis work at Creare.

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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: 1.3	
Title 1 (Original Language): Verhalten von Zry-4-Hüllrohren unter den bei Kühlmittelverluststörfällen auftretenden Beanspruchungen (RS 107-I.1.3, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Behaviour of Zry-4-Canning Tubes under Loss- of-Coolant-Accident Conditions	Project Leader: H.-J. Romeiser
Initiated (Date): 1. 8. 73 — Completed (Date): 30. 6. 76 Status: Completed — Last Updating (Date): 31. 12. 76	

1. General Aim and 2. Particular Objectives

Objective of this task was the investigation of the behaviour of fuel rod canning tubes with respect to loss-of-coolant-accident (LOCA) conditions. The investigations concern to internal pressure burst tests of fuel rod specimen at elevated transient temperatures to determine the diameter increase, burst-pressure, and the burst-temperature.

3. Research Program and 4. Test Facilities

The program is divided into three parts:

a) The influence of different parameters on diameter increase and burst-rupture will be investigated by a parameter study.
The following parameters will be regarded:

- heating rate
- maximum temperature T_{max}
- internal pressure
- material condition (oxidized, H_2 content)
- test-atmosphere (air, inertgas, steam)

Tests will be run with direct resistivity-heating of fuel rod specimen filled with alumina-pellets and a distinct helium pre-pressure. Temperature-time-correlation will be simplified.

b) The influence of the increasing gap between the fuel and the cladding during the heat-up phase on the diameter increase will be studied. Therefore special specimen must be developed with internal heaters and high heat capacity.

c) To find the correlation between the conservative tests above and a realistic excursion of a LOCA, tests will be run with approximated temperature-time-correlation.

5. Progress to Date

Isothermal and isobaric creep tests were carried out at 800°C in vapour and He-atmosphere under the following conditions:

Specimens:	Zry-4 canning rods with PWR dimensions, filled with UO_2 pellets or Al_2O_3 tubes with a central steel rod
Length:	210 mm
Internal Pressure:	5 - 80 bar
Temperature:	900 °C.

The specimens were heated up in steam or in air (about 20 min). When the temperature level was reached, the internal pressure was adapted. The diameter was measured several times.

6. Results

The tests with UO_2 pellets in tubes at 10 and 20 bar showed that the time to burst was prolonged a factor 2 when vapour was used of Helium atmosphere.

The results were:

Internal pressure in bar	Burst time sec	Burst strain %	Strain rate s^{-1}
5	not bursted	8	$6{,}7 \cdot 10^{-6}$
10	4300	25	$5{,}7 \cdot 10^{-5}$
20	145	16	$5{,}3 \cdot 10^{-4}$

The maximum circumferential strain in steam atmosphere was about 40 % - 50 % lower compared with tests in Helium. The rupture time of the specimens was in vapour at an internal pressure of 50 bar greater, at 65 % equal and at 80 bar lower compared with Helium.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

RS 177: Preliminary empirical description of the fuel rod behaviour during LOCA

9. References

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10. Degree of Availability

Classification: 1.3	
Title 1 (Original Language): Vorläufige empirische Beschreibung des Verhaltens von Brennstäben bei hypothetischen Kühlmittelverluststörfällen (RS 177 - I.1.3, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Preliminary Empirical Description of the Fuel Rod Behaviour during LOCA	**Project Leader:** Dr. Wunderlich
Initiated (Date): 1. 9. 75 **Completed (Date):** 31. 12. 76 **Status:** Completed **Last Updating (Date):** 31. 12. 76	

1. General Aim

Deformations of fuel rod tubes during LOCA were to be described empirically. The experimental data of the ballooning and fracture tests gained from RS 107 were to use for the calibration of a material law, which describes analytically the process of fuel rod ballooning during LOCA.

2. Particular Objectives

For ballooning the creep law derived by Norton was to improve because constant Norton parameters are not applicable in the total range of stress and temperatures during a LOCA.

3. Research Program

- development of the empirical material law
- adjustment of the parameters to the data of the RS 107 burst test
- discussion of the influences of indirect heating, oxidation and hydriding on burst-stress and -strain.

4. Experimental Facilities

No experimental facilities were necessary.

5. Progress to Date

The experimental results of ~100 controlled temperature transient burst tests and ~120 creep rupture tests of the R + D project RS 107 are used as data base. All tests were conducted with direct resistance heated KWU Zircaloy-tubes. The range of the experimental parameters was

test time	3 - 94	s
holding time at elevated temperature	0 - 90	s
temperatures	925 - 1650	K
heating rates	50 - 274	K/s
differential pressure	5 - 180	bar
initial stress	3 - 112	N/mm^2
Material	Zircaloy OD 10.75 mm WT 0.725 mm.	

The measured burst-stresses σ_B and -strains ε_B were used to calibrate the parameter A, Q, n of the following creep law

$$\dot{\varepsilon} = A \cdot \sigma^n \cdot \exp(-Q/RT) / (1 - (\sigma/\sigma_B)^2) \qquad (1)$$

The fit results in two sets of parameters referring to the transient burst tests and creep rupture tests respectively.

Controlled temperature transient burst tests:

$$A = 1.2\,(N/mm^2)^{-3.5}\,s^{-1} \qquad (2)$$

$$n = 3.5$$

$Q_\alpha = 2.17 \times 10^5$ J/mole $\qquad Q_\beta = 97218.5 + 125.4\,(T-1253)$ J/mole

$Q_{\alpha\beta}$ linearly averaged

Creep rupture tests:

$$A = 5 \times 10^{2}\ (N/mm^{2})^{-2.8}\ s^{-1} \qquad (3)$$

$$n = 2.8$$

$$Q_{\alpha} = Q_{\alpha+\beta} = Q_{\beta} = 2.03 \times 10^{5}\ \mathrm{J/mole}$$

6. Results

The numerical integration of the strain rate $\dot{\varepsilon}$ in equ. (1), (2), (3) shows a steep increase of the calculated strain with infinite values at rupture. Therefore the measured and calculated failure temperatures are compared, showing a fairly good agreement between theory and experiment. The R + D project RS 177 is completed. Nevertheless, it is necessary to extend the analysis to tests with

- low heating rates
- prior to rupture strains
- internal heated, hydrided, oxidized tubes.

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

B. Brzoska, G. Cheliotis, A. Kunick, G. Senski
Ein neues Modell zur Beschreibung des Dehnungsverhaltens von Zry-Hüllrohren während hypothetischer Kühlmittelverluststörfälle
Vortrag Reaktortagung 1977

10 Degree of Availability

Compact available April 1977.

Classification: 1.3

<table>
<tr><td colspan="2">Title 1 (Original Language):

Parameteruntersuchungen über die Beeinflussung der Hüllrohre durch Nachbarstäbe beim Kühlmittelverlust-störfall
(RS 185 - I.1.3, Jahresbericht A 76)</td><td>COUNTRY:
BRD

SPONSOR:
BMFT

ORGANIZATION:
KWU, Erlangen</td></tr>
<tr><td colspan="2">Title 2 (English):

Investigations on the Influence of Neighbouring Fuel Rods During LOCA.</td><td rowspan="3">Project Leader:

Dr. Weidinger</td></tr>
<tr><td>Initiated (Date):
1. 10. 75</td><td>Completed (Date):
31. 7. 77</td></tr>
<tr><td>Status:

Continuing</td><td>Last Updating (Date):

31. 12. 1976</td></tr>
</table>

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 forces 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

Two fuel rod specimen surrounded by six dummies, made of compact rod material, arranged in a 3 x 4 - 4 configuration. The influence of the ballooning and bursting on the neighbouring rods will be investigated with cooling (gas) and without cooling for the following cases:

a) tests with equal temperatures:
The arrangement shall gurantee, that besides ballooning the lateral displacement of the rods can be investigated. Of special interest are the ballooning of one specimen in a non-disturbed surrounding, the ballooning of a specimen towards another deformed specimen in an non-disturbed surrounding and tests with two ballooning rods.

b) tests with different thermal load:
The specimens are heated internally, resulting in higher temperatures compared with the surround dummies. Planned are tests with one internally heated specimen, two specimen with equal temperatures, and two specimen with different temperatures.

The tests will be run under argon-atmosphere. The fuel rods will contain helium. This concept will be improved when the first results are available.

4. <u>Experimental Facilities</u>

The test apparatus consists of the following equipments:

a) Five-zone-oven with control equipment for the surrounding temperature of the samples.

b) Inside heating (heating transformer with control equipment) to adjust the higher temperature of the samples I or II.

c) Pressure apparatus to adjust the inside pressure.

d) Measuring and recording equipment to control and record the data. Sample arrangement consisting of 2 active and 6 passive samples (dummies).

5. Progress to Date

The test arrangement for isothermal investigations was built-up and assembled. Specimens have been installed. Some preliminary isothermal tests under Helium- and Air-atmosphere have been carried out with two neighbouring Zry-4 rods of 400 mm length. The inner pressure (isobar) was 80 bar, the temperature (isothermal) was 800°C.

6. Results

The first results demonstrate the feasibility of the measuring technique:

- touching and rupture time
- temperature by thermocouples during ballooning and oxidation.

7. Next Steps

The methods for recording the touch-time have to be improved.

A new concept of internal heating of the dummy rods and tubes will be developed.

8. Relation with Other Projects

RS 107 Behvaiour of Zry-4 Canning Tubes Under Loss-of-Coolant-Accident Conditions

9. References

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10. Degree of Availability

Classification: 1.3	
Title 1 (Original Language): Parameteruntersuchungen über die Beeinflussung der Hüllrohre durch Nachbarstäbe beim Kühlmittelverluststörfall (RS 185 - I.1.3, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Investigations on the Influence of Neighbouring Fuel Rods During LOCA	Project Leader: Romeiser
Initiated (Date): 1. 10. 75 Completed (Date): 31. 7. 77 Status: Continuing Last Updating (Date): 31. 12. 1975	

General Aim

The influence of neighbouring fuel rods on the ballooning and burst behaviour of a single fuel rod during LOCA and the mechanical and thermal forces are to be investigated.

Particular Objectives

The experimental tests shall give information, whether the results, obtained for a single fuel rod (parameter: differential pressure, temperature, holding time) are changed by neighbouring fuel rods. Expecially it will be investigated, what happens when the neighbouring rods are touched: change of diameter increase, time until burst and influence of burst on neighbouring rods.

Experimental Facilities

The test apparatus consists of the following equipments: five-zone-oven with control equipment for the surrounding temperature of the samples.
Inside heating (heating transformer with control equipment) to adjust the higher temperature of the samples I or II.
Pressure apparatus to adjust the inside pressure.
Measuring and recording equipment to control and record the data.
Sample arrangement consisting of 2 active and 6 passive samples.

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Research Program

Two fuel rod specimen are surrounded by six dummies, made of compact rod material, arranged in a 3 x 4 - 4 configuration. The influence of the ballooning and bursting on the neighbouring rods will be investigated with cooling (gas) and without cooling for the following cases:

a) tests with equal temperatures:
The arrangement shall guarantee, that besides ballooning the lateral displacement of the rods can be investigated. Of special interest are the ballooning of one specimen in a non-disturbed surrounding, the ballooning of a specimen towards another deformed specimen in a non-disturbed surrounding and tests with two ballooning rods.

b) tests with different thermal load:
The specimen are heated internally, resulting in higher temperatures compared with the surrounding dummies. Planned are tests with one internally heated specimen, two specimen with equal temperatures, and two specimen with different temperatures

The tests will be run under argon-atmosphere, the fuel rods will contain helium. This concept will be improved when the first results are available.

Project Status/Progress to Date/Essential Results

The test arrangement for isothermal investigations was designed and constructed. Some components of the equipment are built and assembled. The material for the specimen and dummies have been procured. The tests have not yet started, therefore no results are available.

Next Steps

The test apparatus will be completed, the specimen and other components will be installed. Some preliminary isothermal tests with and without argon cooling will be performed.

Classification: 1.3	
Title 1 (Original Language): Notkühlung von LWR: Theoretische und experimentelle Untersuchungen zum Brennstabverhalten beim Kühlmittelverlustunfall und ATWS und zur Auswirkung von Brennstabschäden auf die Wirksamkeit der Kernnotkühlung (PNS 4230 - I.1.3., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Theoretical and Experimental Investigations of LWR-Fuel Rod Behaviour during LOCA and ATWS	Project Leader: A. Fiege (coordination)
Initiated (Date): 1973 — Completed (Date): 1979 Status: continuing — Last Updating (Date): December 1975	

General Aim

The aim of this project is the development of verified analytical models for the response of LWR fuel rods to LOCA and ATWS conditions and the reliable description of failure mechanisms and their feedback to the effectivness of ECCS in a fuel behavior code system SSYST (PNS 4231).

The detailed quantitative understanding incorporated in the fuel behavior code must be verified by representative experiments.

The basic philosophy of the experimental Program of the Projekt Nukleare Sicherheit (PNS) is to investigate the failure mechanisms of zircaloy-clad fuel rods as a function of the main parameters (as differential pressure, temperature and material properties) systematically with a broad spectrum of out-of-pile experiments and to confirm these out-of-pile experiments with special in-pile investigations.

The main tasks of this program are:
- Investigations of the material properties of zircaloy at high temperatures (PNS 4235.1, 4235.2, 4235.3)
- Out-of-pile loop experiments under reactor typical coolant conditions in different accident phases (PNS 4236, 4238, 4239)
- In-pile experiments in the steam contamination (DK)-loop of the FR2 reactor (PNS 4237.1, 4237.2)

In addition, a smaller experiment is performed in the FR2 reactor to provide better data of the 235 U decay heat in the first 10 to 1000 sec of a LOCA (PNS 4234)

These different projects mentioned before are described in seperate reports.

Classification: 1.3	
Title 1 (Original Language): Theoretische Untersuchungen zum Brennelementverhalten bei Störfalltransienten (PNS 4231 - I.1.3, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: ORGANIZATION: GfK Karlsruhe Projekt Nukleare Sicherheit
Title 2 (English): Theoretical Investigations of Fuel Behavior under Transient Conditions	**Project Leader:** R.Meyder/GfK-IRE H. Unger/IKE Stuttg
Initiated (Date): 1973 **Completed (Date):** 1980 **Status:** Continuing **Last Updating (Date):** December 1976	

1. General Aim

The aim of this project is the development of a code system (SSYST) in order to model and calculate the behavior 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 of the effectiveness of emergency core cooling is a major aim of the completed and 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 fuel and clad, pressure in coolant and fuel rod as well as the thermo- and fluiddynamic conditions in the coolant channel and the primary coolant system of a LWR.

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 rod bundles.
- Investigation of geometrical bundle configurations for longterm coolability.

4. Project Status

4.1 Progress to date

The progress up to 1976 has been reported in PNS semiannual reports. The progress achieved in 1976 can be summarized as follows:

- SSYST-Mod 1 is in use for test rigs /1/, PWR /2/ and for comparision with FRAP-T /3/
- SSYST-Mod 1 is documented in /4/.
- The work for improved models for SSYST-Mod 2 is well underway. It covers the topics: subchannel-thermohydraulics, rod mechanics and probabilistic methods.
- To describe the long term behaviour of fuel rods, a version of FRAP-S has been implemented on a CDC 6600, it is also available on an IBM 370/168.

4.2 Essential Results

Within the framework of the German-American cooperation in the reactor safety related field of the program development in order to model the fuel pin behaviour, the ANC-FRAP "Standard Problem No. 1" was calculated using SSYST. Those calculations allowed an integral test of all important modules of SSYST-Mod 1 and a comparison of the results with those of FRAP-T2 calculations

The comparisons showed good results in general. Deviations in the gap heat transfer coefficient during the transient were caused by differences in the calculational models. Calculations of the clad deformation are in good agreement with the results of ballooning experiments performed by Emmerich /ASTM-STP-458(1969)/.

Improved data for Zry-cladding oxidation were deduced from experiments done at ORNL and PNS 4235.2. The temperature dependance of diffusion coefficient for oxygen in ZrO_2 could not be modelled with one Arrheniusterm only; it is felt that in the temperature range $1170K \leq T \leq 1570K$ at least two Arrheniusterms are necessary.

For the temperature range of $870K \leq T \leq 1020$ K parameters for Nortons creep law were deduced from tensile tests in PNS 4235.1. These data fit quite well with burst experiments in PNS 4238.

LOCA calculations for fresh reactor rods show that axial pressure differences are small.

The discussion on statistical methods to determine for example the probability-density function for peak cladding temperature showed that response surface method should be preferred against the nonlinear error propagation method.

By means of a special numerical treatment of the wetting-region (large temperature gradients in axial direction!) during the reflood phase a reduction in computer time and -space could be achieved.

In order to avoid numerical instabilities under certain circumstances, the treatment of the heat conductance in the fuel pin and the thermohydraulics of the coolant has been revised and combined in a new module (ZETHYD).

The documentation of SSYST-Mod 1 has been performed in several steps. There are reports available /4 - 6/ for both, the user who is interested in routine-type calculations as well as for the specialist who has to service the program system.

5. Next Steps

The steps for the near future are:

- Incorporation of a steady state code into SSYST.
- Investigation of the propagation problem.
- Continuous development of modules for SSYST-Mod 2.
- Verification of the code-system with integral experiments.

6. Relation with other Projects

This project (PNS 4231) is part of the major project PNS 4230 (emergency core cooling of LWR) and strongly connected to other activities within PNS 4230.

For the present project, experimental information about the mechanical properties of Zry at elevated temperatures is provided by PNS 4235.1, information on high-temperature Zircaloy-steam reaction by PNS 4235.2, and on the chemical interaction between Zry and fuel by PNS 4235.3. Within the presently described project (PNS 4231) theoretical calculations assisting the conception and analysis of experiments (PNS 4236-4239) are performed by using special module sequences of SSYST.

7. Reference Documents

/1/ R. Meyder, S. Raff, W. Sengpiel
Sensitivity study on some parameters of internal pressure of PWR fuel rods during blowdown of a LOCA using SSYST-Mod 1.
CSNI Specialist Meeting, Spåtind, Norway, 13-16. Sept. 1976

/2/ R. Schützle in
R. Meyder, H. Unger
Beiträge zum 2. Halbjahresbericht 1976 des Projektes Nukleare Sicherheit des Kernforschungszentrums Karlsruhe (KFK in preparation)

/3/ W. Gulden et al.
Synopis of the basic models implemented in fuel rod behaviour codes FRAP-T and SSYST and some comparative calculations.
CSNI Specialist Meeting, Spåtind, Norway, 13-16. Sept. 1976

/4/ W. Gulden et al.
Dokumentation SSYST-Mod 1. Ein Programmsystem zur Berechnung des Brennstabverhaltens bei einem Kühlmittelverluststörfall (KFK in preparation)

/5/ Gulden, W.
Eingabebeschreibung für die Moduln des Programmsystems SSYST-1.
Bericht des Instituts für Kernenergetik der Universität Stuttgart (in preparation)

/6/ Brestrich, I.A., R. Rühle
Implementierung des Programmsystems RSYST, Version 1.2, auf der IBM 370/168.
Bericht des Instituts für Kernenergetik der Universität Stuttgart, IKE 4-46 (Juli 1976).

8. Degree of Availability

Unrestricted distrubtion.

Classification: 1.3	
Title 1 (Original Language): Untersuchungen zum mechanischen Verhalten von Zircaloy-Hüllmaterial bei Störfalltransienten (PNS 4235.1 - I.1.3, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** **ORGANIZATION:** GfK, Projekt Nukleare Sicherheit
Title 2 (English): Investigations of the Mechanical Behavior of Zircaloy Cladding Material under Transient Conditions	**Project Leader:** M. Boček, IMF/II
Initiated (Date): April 1972 **Status:** Continuing **Completed (Date):** 1978/79 **Last Updating (Date):** Dec. 1976	

1. General Aim

Investigation of the plastic behavior of Zircaloy-4 during different reactor incidents, especially LOCA-typical temperatures and stress transients.

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 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 of irradiated material.

4. Experimental Facilities

- Tensile testing will be performed in an INSTRON closed-loop machine.
- For burst tests in vacuum tubes will be pressurized in a radiation furnace.
- For integral experiments (steam environment) cladding with internal heaters will be used.

5. Progress to Date

Ad 3.1:

- Investigations on superplastic deformation of Zry-4 tensile specimens (tests in air atmosphere).
- Creep behavior of Zry-4 tensile specimens in low vacuum.

Ad 3.1.2:

- Examination of the influence of temperature excursions into the β-phase region on the plastic behavior of Zry-4 tensile specimens tested in the α-phase region.

Ad 3.1.3:

- Analysis of the influence of ZrO_2-layers upon the strain rate sensitivity index m.
- Examination of the influence of the environment on the change in shape of Zry-4 specimens during a tensile test.

Ad 3.2:

Analysis of a stability criterion for burst tests.

6. Results

Ad 3.1:

- Metallographic techniques as well as texture and scanning microscopic examinations gave evidence that in the $(\alpha+\beta)$-phase region the grain switching is the dominating deformation mechanism.
- Anomalous transient creep is observed in the temperature range from 700 to 1000 ^{o}C for tests performed at low vacuum. The strain rate sensitivity index m determined from load cycling tests are dependent on the temperature and load.

  Within the range of temperatures under consideration and for deformation rates $\cong$ $10^{-6}s^{-1}$, it is obvious that $m \to 1$.

In cases where comparisons were made between m-values determined by different methods the agreement is satisfactory.

Ad 3.1.2:

Temperature excursions performed in vacuum (up to 1200°C) reduce the ductility of Zry-4 tensile specimens tested in the α-phase region (800°C). The results are also dependent on the details of the temperature treatment.

Ad 3.1.3:

- The test specimen is considered as a two element composite cylinder (a thin and brittle ZrO_2 coating firmly covers the ductile metallic filament). Plastic deformation of the specimen can proceed only if cracking of the coating is possible. From considerations it follows that if the actual crack density depends on the strain rate (or stress), the strain rate sensitivity index does not solely reflect the plastic behavior of the filament but also depends on the properties of the coating.
- From a comparison of results of tensile experiments performed in air atmosphere and vacuum resp. it follows that ZrO_2-coatings suppress macroscopic necking.

Ad 3.2:

Relationships were derived between the logarithm of the tangential deformation rate log $\dot{\varepsilon}_\theta$ and the corresponding strain ε_θ, on the one hand, and certain material properties, on the other hand.

From these relationships conclusions can be drawn about the deformation behavior of the pressurized tubing.

Three cases were considered in which the tendency towards instability increases in the sequence: hoop stress σ_θ = const.; hoop load L_θ = const. and internal pressure p = const.

7. Next Steps

1. Examination of creep behavior of Zry-4 tensile specimens under very low loads and very low partial pressure of oxygen (Ad 3.1).
2. Investigation of the influence of ZrO_2-coatings upon the ductility of tensile specimens in the α-phase region after a temperature excursion in the β-phase region (vacuum testing) (Ad. 3.1.2 and 3.1.3).
3. Experimental examination of oxide coatings on the plastic behavior of the composite Zry-4/ZrO_2 (Ad 3.1.3).

8. Relations to other projects

PNS 4235.2
PNS 4235.3
RS 107

9. References

Boček, M.; Faisst, G.; Petersen, C.
Examination of the plastic properties of Zry-4 at elevated temperatures in air atmosphere.
Journal of Nuclear Materials. 62 (1976) S. 26-36.

Boček, M.; Hofmann, P.; Petersen, C.
Superplasticity of Zircaloy-4.
3rd International Conference on Zirconium in the Nuclear Industry, Quebec, Canada, August 10-12, 1976.

Boček, M.; Petersen, C.
High Temperature plastic behaviour of Zry-4 in Air Atmosphere.
Specialists Meeting on the Behaviour of Water Reactor Fuel Elements under Accidental Conditions, Spåtind, Norway, September 13-16, 1976.

1st PNS-Sem.-Annual Report 1976 (German with English abstracts), KFK 2375, Nov. 1976. S. 256-283.

10. Degree of availability

Unrestricted distribution.

Classification: 1.3	
Title 1 (Original Language): Untersuchungen zur Hochtemperatur-Wasserdampf-Oxidation an Zircaloy-Hüllrohrmaterial (PNS 4235.? - I.1.3, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** **ORGANIZATION:** GfK Karlsruhe Projekt Nukleare Sicherheit
Title 2 (English): Investigation of the High Temperature Steam Oxidation of Zircaloy Cladding Tubes	**Project Leader:** S. Leistikow /IMF II
Initiated (Date): 1973 **Completed (Date):** 1978/79 **Status:** Continuing **Last Updating (Date):** December 1976	

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. Experimentel Facilities

   Experimental set-ups for isothermal and temperature-transient oxidation reactions. Facilities for isothermal and temperature-transient stress-rupture testing.

5. Progress to Date

   An induction heating system for program-controlled Loca-similar time-at-temperature steam exposure of tube sections and the experimental set-up for stress-rupture testing of internally pressurized tube capsule in argon and steam were successfully operated. Loca-similar experiments were performed in exposing tube sections to steam oxidation attack at program controlled temperatures and exposure times. Oxygen take-up was measured as function of various blowdown peak temperatures (950 - 1200°C), PWR-Loca shaped time-at-temperature functions (900 - 1300°C), and of various cooling rates.

   Furthermore stress-rupture experiments were performed in argon and steam (800 - 950°C) as second part of comparative tests. By single specimen tests and post-test evaluation creepcurves were established.

6. Results

The exposures under different blowdown-temperature conditions (peaking at 950 - 1200°C) resulted in weight gains of 50 - 200 mg/dm^2. Typical Loca conditions ($\leq$ 900°C, 180 sec) caused weight gains of $\leq$ 100 mg/dm^2, corresponding to calculated < 6,7 μm ZrO_2. Loca similar exposures with holding times at 1000 - 1200°C showed weight gains of 200 - 540 mg/dm^2 which were at all temperatures tested about 33 % lower than the corresponding isothermal ones. The temperature-transient tests were accompanied by SIMTRAN I-Zircaloy-Oxidation-Code calculations, which proved not only agreement in total weight gain, but also in the calculated thicknesses of the ZrO_2- and α-layer. It could be demonstrated that when specimens are exposed during to Loca typical temperature-transient conditions up to 1300°C, followed by controlled cooling (2,2 - 57,5°C/s), consequently surplus weight gains up to 30 % and structural variations in the α'-phase are registered. Their influence on mechanical properties is matter of future testing.

The exposure of preoxidized specimens to Loca typical temperature-transient conditions during 3 min and peaking at 1000 - 1100° resulted in (compared to the exposure in the metallic state) reduced oxygen consumption, increased oxygen uptake was measured at 1200°C.

The tube capsule stress-rupture tests established the initial tangential stress/time-to-rupture relationships for 800 - 1300°C, 2 - 70 at, 20 sec to 40 min in argon. The maximum circumferential elongation was plotted as function of temperature, the time-to-rupture was given as function of initial and true tangential stress.

Comparative tests were performed in argon and steam at 800, 900 and 950°C. They showed that a strengthening effect and a remarkable reduction of ductility was evoked by oxidation already in this time and temperature range. Again, stress-rupture curves were established, but also three creep curves were measured by interruption of single specimen tests at 900°C and their macroscopic and microscopic evaluation. Scanning microscopic evaluation proved that the crack pattern of ruptured specimens varied, with changing time and mechanical load.

At high creep rate a small number of broad cracks, at low creep rate many narrow cracks could be detected. The interdependence of the number and width of cracks with the internal pressure time-to-rupture and maximum elongation, was shown semiquantitatively.

7. Next Steps

Comprehensive documentation of oxidation kinetic results. Temperature-transient exposure of specimens preoxidized at lower temperatures. Stress-rupture testing under isothermal ($\geq$ 1000°C) and temperature-transient conditions.

8. Close relations to the other fuel behaviour programs of PNS/GfK and US-NRC.

9. References

9.1 S. Leistikow, H.v. Berg, D. Jennert, R. Kraft u. G. Schanz
"Untersuchungen zur Hochtemperatur-Wasserdampf-Oxidation von Zircaloy 4-Hüllrohren"
PNS Semi-Annual Report 1976/1.
KFK 2375 (1976) 283 - 314.

9.2 S. Leistikow, H.v. Berg und D. Jennert
"Comparative Studies of Zircaloy 4 High Temperature Steam Oxidation under Isothermal and Temperature Transient Conditions".
CSNS-Spec. Meeting on the Behaviour of Water Reactor Fuel Elements under Accident Conditions, Spåtind, Norway, 13-16.9.76.

10. Degree of Availibility of the Reports.

Unrestricted distribution.

Classification: 1.3	
Title 1 (Original Language): Untersuchungen zum Brennstabverhalten in der Blowdown-Phase eines Kühlmittelverluststörfalles (PNS 4236 - I.1.3, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** **ORGANIZATION:** GfK/Karlsruhe Projekt Nukleare Sicherheit
Title 2 (English): Investigations of the Fuel Rod Behavior during the Blowdown-Phase of a Loss-of-Coolant Accident	**Project Leader:** G. Class, IRE K. Hain, RBT
Initiated (Date): 1972 **Status:** Continuing **Completed (Date):** 1978/79 **Last Updating (Date):** 31.12.1976	

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.

5. Progress to Date

The construction of the system was terminated and the definitive license for operation was granted by the supervising authority. The behavior during operation of single component groups was tested and experiments were performed including the simulation fuel rods with loads applied of up to about 30 % of the maximum load.

A prototype version of the required pressure- and temperatureresistant true-mass-flow-meter was completed. Further special measurement devices were mounted and tested. Work was continued on the recording of measured values and on component control.

Theoretical activities centered around the establishment of the computer program for evaluation. This includes the provision and conversion of data, their examination for inner consistency and, finally, the thermo-hydraulic interpolation of conditions prevailing in the test section on the basis of boundary conditions measured.

6. Results

The test loop reached the nominal operating values on October 18 for the first time. It is fully performing. The test of single component groups yielded satisfactory results. However, delays occurred in the realization of measured values recording and component control and in the fabrication of the true-mass-flow-meter.

A RELAP4 version for the thermohydraulic interpolation in the test section was completed and subjected to testing. Also the parts of the evaluation program were programmed and tested which will be used for the computation of heat transfer coefficients and for data conversion and consistency verification.

7. Next Steps

The true-mass-flow-meters and a safety scram system for rod heating will be installed in the test loop; likewise the unit for the transmission of measured values and the control of components. Subsequently, test operation with the simulation fuel rod can start. It is intended to complete the computer program for evaluation at such date that the first tests recorded by CALAS can be immediately evaluated. Based on preliminary test results, the test parameters can then be specified more accurately.

8. Relation with other Projects

PNS 4231, 4237, 4238, 4239

9. References

1st semiannular Progress-Report PNS 1/1976
2nd semiannular Progress-Report PNS 2/1976

10. Degree of Availability

Unrestricted distribution.

Classification: 1.3	
Title 1 (Original Language): Untersuchungen zum Brennstabversagen in der 2. Aufheizphase eines Kühlmittelverluststörfalles - In-pile-Versuche mit Einzelstäben im DK-Loop des FR2 (PNS 4237 - I.1.3, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: ORGANIZATION: GfK Karlsruhe Projekt Nukleare Sicherheit
Title 2 (English): Investigations on Fuel Rod Failure in the 2nd Heatup Phase of a LOCA In-pile Experiments with Single Rods in the DK Loop of the FR2 Reactor	Project Leader: E. Karb/RBT
Initiated (Date): July 1972 Status: Continuing	Completed (Date): 1980 Last Updating (Date): 31.12.1976

1. General Aim

The in-pile experiments performed in the DK loop* of the FR 2 reactor aim at investigating the influence of the "nuclear parameters" on the mechanisms 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

3.1 PNS 4237.1: It is planned to carry out 42 nuclear tests with fresh and preirradiated specimens. Stages of burnup: 0/ 2,500/ 5,000/ 10,000/ 20,000/ 35,000 MWd/t_U. Range of internal pressures: 45 to 100 bar.

3.2 PNS 4237.2: 25 non-nuclear BSS tests are planned. Range of internal pressures similar to 3.1.

* DK = Dampf-Kontamination = Steam-Contamination

4. Experimental Facilities, Computer Codes

4.1 Test Facilities

4.1.1 Test Loop

Both types of rods are used in the DK loop of the FR2. This loop is operated by 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 separate waste-gas system is available for the fission products es[illegible] from the burst nuclear rod, which decontaminates the wa[illegible] gases from volatile halogens and noble gases; solid and liquid isotopes are retained by filters.

4.1.2 Preirradiation

The test rods are preirradiated in the FR 2 up to the desired target burnup. For each burnup six rods are assembled to form a preirradiation rig similar in its structure to that of an FR 2 fuel element. For safety monitoring and determination of the condition of burnup each rig has been provided with:

- 1 double TC at the D_2O inlet,
- 1 double TC at the D_2O outlet, } ΔT-determination
- 1 vanadium detector allowing to measure the neutron flux density,
- 1 flow rate transducer (turbine type),
- 1 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
- 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, specimen rig assembly and specimen instrumentation are carried out in the FR 2 shielded cell. This requires special auxiliary devices, such as

- coupling and welding device,
- 2 cutting devices,
- testing vessel with equipment for leak testing,
- 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.

In the FR 2 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,
- X-ray testing,
- metallographic examination,
- sample fabrication for the radiochemical burnup determinations.

4.2 Computer Programs

For precalculation and verification, respectively, 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).

5. Progress to Date

5.1 The PIE of the first two rods exposed to transients in October 1975 was meanwhile completed in the Hot Cells; other rods from tests in 1976 are presently examined. First partial results of this year's first specimen (A 2.1) are available.

5.2 A total of five nuclear tests were performed in 1976. Three tests completed the scoping test series. In this test series the internal pressure and the axial power profile were varied. The design data of modern PWR's (45 to 100 bar) were taken as the basis for the range of internal pressures. The internal pressure was adjusted to 25 bar in one of the tests.

5.3 After inserting the third preirradiation rig in the FR2 in March 1976 (anticipated burnup 20,000 MWd/t_U) those three rigs which are to provide the highest of the selected burnup stages (20,000 and two times 35,000 MWd/t_U) are in operation now.

5.4 After the new orifice for steam flow determination had been exactly measured and calibrated, it was exchanged early this year against the old orifice provided in the loop.

5.5 To increase the oxygen offer during the transient in the test section calculations were made about the smallest steam mass flow

- feasible technically,
- guaranteeing a sufficient O_2-offer.

Considerations concerned a capillary allowing to set this mass flow.

5.6 The errors in the data acquisition system were eliminated early this year so that the CALAS system was able to assume complete logging of the data measured during the tests.

5.7 A first fabrication series of thermocouples (TC) was bought. Compared with the prefabrication series only one major modification was specified: Instead of the TC-sheath made of stainless steel Inconel 600 is used because of its better resistance against scaling in water vapor.

5.8 The following problems were studied on four non-pressurized fuel rod simulators (BSS) in six test groups:

- operating behavior of the heater rods,
- heatup behavior using different pellet materials (Al_2O_3, UO_2)
- "true" cladding temperature,
- comparison of thermal power with electrical power,
- influence of reactor power (γ-heat).

5.9 A design concept was elaborated for a BSS version able to undergo ballooning.

5.10 Comparative calculations were performed by IKE Stuttgart

- concerning the heatup behavior of the rod if equipped with Al_2O_3 and UO_2 annular pellets, respectively;
- to verify the test values. For the first time, new values were used for the specific heat c_p of Zry-4 (according to MATPRO) which undergoes abrupt charges above 800 °C.

5.11 To evaluate the test data stored on magnetic tape a computer program was established allowing to print out and plot the measured data as a function of time. The first tests have already been evaluated using this program. The program serves above all to plot the cladding tube temperatures and internal pressure versus the time. The data printed out provide the basis of the quicklook data sheet.

6. Results

6.1 The essential results from PIE of the first ballooned rod are:

- permanent elongation of the rod 5.5 mm $\hat{=}$ 1.1 % of active zone
- length of ballooning about 60 mm,
- maximum circumferential increase at the point of ballooning 65%,

- length of the crack 19 mm,
- maximum crack width 1.3 mm.

The pictures of the whole rod show that there are no further balloonings besides the large ballooning.

The X-ray photos show broken pellets in the zone of ballooning whilst the pellet stack is unaffected in the non-ballooned zone.

It appears from the transverse sections in the zones of maximum and moderate radial elongation that ballooning is almost circular in shape.

6.2 Since the PIE of the test rods have to be performed in the Hot Cells, the results are available after relatively long delays only. Immediately after a test only data can be evaluated which had been recorded during the test. The most important variables are the burst pressure and the burst temperature. After these quantities had been defined, the pertinent results of our tests were compared with out-of-pile data published by other authors. It appears that the in-pile data are within the range of results obtained out of pile.

6.3 There has been no failure so far during irradiation of the rods. Some disturbances occurring at the measuring equipment were eliminated.

6.4 The calibration of the new measuring orifice yielded for the flow number α: Although the tube diameter, nominal width 10, is clearly below the range of validity of 52 mm for standard orifices, the α-value exactly corresponded to the tabulated value of the standard orifice according to DIN 1952, with the existing aperture ratio.

6.5 The steam mass flow ensuring the desired O_2-offer at the specimen was determined to be 0.2 kg/h. In this way, the cooling effect during heatup remains sufficiently low and with respect to the steam velocity at the fuel rod the conditions are similar to that in a LOCA of the PWR. The vapor flow is to be realized with a capillary of about 0.5 mm inside width and 12.5 m length.

6.6 The CALAS data acquisition system has performed in a reliable manner and without major disturbances.

6.7 In TC-fabrication difficulties were encountered during hammering of platinum jackets: The required shear forces of at least 100 N were not achieved. In-depth investigations of the possible influencing factors have shown that only the wall thickness of the TC-sheath exerts an influence on hammering. The best results were obtained with t = 0.13 to 0.15 mm. Therefore, only TC sheaths having this wall thickness will be used in future.

6.8 In all the tests the heater rods have performed without fault. Their principle of design can be adopted for the design of reference rods.

The transient cladding tube temperatures recorded clearly differ in temperature rises between Al_2O_3 and UO_2 annular pellet configurations at the same power. The temperature rise with time is less for the Al_2O_3 rod than for the rod containing UO_2 annular pellets. This is in conformity with the verifications. The determinant variable is the heat storing capacity which takes the following values for the different rod types:

	BSS with Annular Pellet Al_2O_3	UO_2	Nuclear Rod (Solid Pellet) UO_2
$\frac{m_{rod} \cdot c_{rod}}{cm} \left[\frac{J}{K \cdot cm}\right]$	3.25	2.86	2.51

(standard values at 400°C) according to Steiner, IKE Quarterly Report, unpublished

The tests relating to the "true" cladding tube temperature yielded little dependence of the temperature difference between the wall and the surface TC's on the temperature level, but a clear dependence on the specimen power. This allows to establish a correcting value of 75 $\pm$ 35K for the 50 W/cm electric power. The scattering range of $\pm$ 35K is attributable to the external TC's which are more or less closely fixed to the cladding tube.

After examination of the measuring devices for power determination the influence of the TC configuration in the test rig remains as a possible source of error. It is assumed that as a result of inadequate mixing stream filaments of different temperatures are formed so that the ΔT-measurement becomes erroneous. Neither can a leak stream in the head of the test rig be excluded completely, which falsifies the mass flow indication. It has been demonstrated by the BSS 06 calorimeter that error sources arising outside the reactor rig can be excluded. The deviation between the thermal and electrical power amounted not higher than 2,5 W/cm in this case.

The influence of reactor power on transient heatup appeared to be very low. This is confirmed by the results of earlier BSS tests and of calculations made: An estimate based on measured temperature values yielded a power contribution by reactor operation of about 0.5 W/cm.

Obviously, the reference tests have not to be conducted at reactor power.

6.9 The DILO double screw coupling at the head of the test rig is replaced by an explosion welded joint between the Zry-cladding and the stainless steel specimen carrier tube.

6.10 If the c_p-values according to MATPRO are taken into account, the calculated plots give a better approximation to the test curve in the upper temperature range above 800°C.

7. Next Steps

The test series B comprising fresh nuclear rods will be continued. Six tests have still to be made within this series. Also in the future priority will be given to the nuclear rod tests as against the BSS tests.

As a preparation of tests with preirradiated rods testing will start of the Hot-Cell components (coupling, welding, inspection and handling devices).

The preirradiation in the FR2 reactor three rigs containing six rods each will be continued

It is intended to insert, calibrate and test the bypass system ensuring the O_2-offer.

Arrangements have started to provide replacement parts for the pressure tube and immersion tube of the in-pile section of the loop since these components will reach the admissible service life the foreseeable future.

Studies will continue on the deviations in power determination by a special BSS test.

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 using computer programs.

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

9.1 Publications

E. Karb, G. Harbauer, W. Legner, L. Sepold, K. Wagner: "Theoretische und experimentelle Untersuchungen zur Gasströmung in LWR-Brennstäben bei Kühlmittelverluststörfällen", KFK- Bericht 2411, December 1976

10. Degree of Availability of the Reports

Unrestricted distribution

Classification: 1.3	
Title 1 (Original Language): Referenzversuche zu den in-pile-Experimenten PNS 4237.1 mit elektrisch beheizten Brennstabsimulatoren (Einzelstäbe) (PNS 4237.2 - I.1.3., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Reference Tests for PNS 4237.1. In-pile Experiments with Electrically Heated Fuel Rods (Single Rods)	**Project Leader:** B. Räpple
Initiated (Date): 1.1.1973 — **Completed (Date):** 1979 **Status:** continuing — **Last Updating (Date):** December 1975	

1. General Aim

Investigations of the mechanisms and the extent of failure of Zircaloy clad fuel rods during the second heatup phase of an LWR loss-of-coolant accident.

2. Particular Objectives

Provision of experimental data by means of electrically heated fuel rod simulators as a basis of comparison with results of experiments obtained with fuel rods exposed to nuclear heating. (PNS 4237.1)

3.1 Experimental Facilities

The experiments are being performed in the in-pile loop of the FR2 Research Reactor.

3.2 Research Program

Some 25 rods are exposed to transient operating conditions under the test program. The pressure differences, the temperature gradients and the power profiles at the specimen are assimilated to the conditions applicable to nuclear tests and the resulting points of failure are examined and evaluated.

4. Project Status

Present activities relate to testing of the heater rod concept, the test rig and the measuring system.

4.1 Progress to Date

Manufacturing was completed of a first series of 5 heater rods to be used in preliminary in-pile tests. One of the rods was installed in a test rig together with Al_2O_3 ring pellets and subjected to about 30 steady-state and 3 transient tests in a first in-pile program.

4.2 Essential Results

The tests had been essentially a success. Disturbances only occurred in the measurement of wall temperatures, which had a negative effect on axial temperature profile recording at the cladding tube. All the other test targets have been reached. They allowed to gather experience relative to

- the gamma dose rate of the irradiated specimen,

- the necessary setting times (electric heating, reactor, loop),

- possible sources of error at the specimen, the instrumentation, the electric and the measurement systems inclusive of data processing (CALAS),

- the application of the methods allowing to determine the thermal and electric power, respectively,

- processing of measured values in steady-state and non-steady-state operation.

5. Next Steps

It is planned to perform additional preliminary in-pile tests during the first half of 1976 aiming at the following targets:

- Continued testing of the heater rod, the electric and measurement systems; error correction.

- Determination and comparison of the power generated in the heater zone (electric, thermal, nuclear).

- Determination of the gamma heat fraction in the aggregate thermal power.

It is planned to install another three test rigs for this purpose. The prerequisite of these preliminary in-pile tests is the installation in the test rig of a new calibrated orifice.

Work on specimen design and testing will be continued and evaluation will start of the preliminary in-pile tests.

6. Relation with Other Projects

The tests will alternate with the nuclear tests to be performed at the same test facility and in the same test position of FR-2. The results obtained in the preliminary in-pile tests will also be used in the nuclear power determination method under the PNS 4237.1 project.

The main tests planned for a later date are intended to yield reference data on the nuclear test parameters of the same project.

7. Reference documents

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

8. Degree of availability

Unrestricted distribution

Classification: 1.3	
Title 1 (Original Language): Untersuchungen zur Wechselwirkung zwischen aufblähenden Zircaloy-Hüllen und einsetzender Kernnotkühlung (PNS 4238 - I.1.3, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** **ORGANIZATION:** GfK Karlsruhe Projekt Nukleare Sicherheit
Title 2 (English): Investigations on the Interaction between Ballooning Zircaloy Claddings and the Reflooding Emergency Core Cooling Water	**Project Leader:** K. Wiehr/IRB
Initiated (Date): 1973 **Status:** Continuing	**Completed (Date):** 1980 **Last Updating (Date):** 31.12.1976

1. General aim

The aim of this project is to obtain experimental information for development and verification of the SSYST fuel rod behavior code with regard to the heat-up and reflooding phases of a LOCA.

2. Particular objectives

Important features of the experiments are the interaction between the ballooning mechanism and the emergency core cooling and the recording of the time-dependent ballooning process of the zircaloy cladding.

The experiments have the following particular objectives:

- time-dependent ballooning mechanism of single rods
- interaction between ballooning and cooling
- thermal and mechanical effects of the rod-to-rod interaction on failure behavior in rod bundles
- information on failure propagation
- extent and distribution of flow blockage.

3. Research program

The experiments have begun with separate effect tests related to heat transport inside the fuel rod simulator, to inner gas flow in the gap between cladding and pellets and to deformation mechanism of the cladding. With these tests performed on shortened fuel rod simulators and under simplified conditions the corresponding moduli of the SSYST-code will be developed and verified.

The integral tests with a 5x5 rod array are concentrated, for the time being, on single rod tests in a steam atmosphere and under flooding conditions in order to investigate the detailed ballooning mechanism. After sufficient understanding of the single rod failure mechanism has been generated the tests will be extended to bundle tests for investigations of rod-to-rod interaction and failure propagation.

The test parameters will be varied in the following range:

- rod power	24 to 80 W/cm
- axial power profile	stepped profile and cosine-shaped profile
- cladding temperature at beginning of flooding	600 to 900° C
- pressure difference across cladding	20 to 100 bar
- system pressure	1 to 4.5 bar
- flooding rate (cold)	1 to 30 cm/s
- water inlet-temperature	25 to 65 °C

4. Experimental facilities

The experiments will be carried out with newly developed fuel rod simulators with 3.90 m heated length, axial power profile and a total length of 5 m. The fuel rod simulators are assembled to a 5x5 array and installed in the test section of the test rig. The test rig is able to simulate the processes during the refill and reflooding phases of a LOCA. The test rig is a closed loop system with steam and water circuits. The water circuit delivers the cooling water into the test section at different flooding rates and inlet-temperatures. The electrical power control of the fuel rod simulators will be done by an automatic system which can adjust the power according to the decay heat.

X-ray penetration with a camera will be used for observation and recording of the ballooning process. For temperature measurement of the ballooning cladding a special device with a two-colour pyrometer has been developed. The internal gas pressure in the ballooning zone will be measured by means of a capillary with a micro-pressure strain-gauge transducer.

The 130 experimental data are recorded by a fast data acquisition system at a scanning frequency of 10 kHz and ten times per second. A fast data line with terminals and video displays is used to process the measured data by the CALAS-system.

5. Progress to Date

The following tasks have been largely completed:

- Development and testing of the fuel rod simulators
- Characterization of the fuel rod simulators with regard to temperature distribution
- Commissioning of the test rig
- Set-up of separate test rigs for calibrating unconventional measuring devices
- Theoretical studies and calculations on the thermal and gas dynamics behavior of the fuel rod simulators, on the response time characteristic of the capillary pressure measuring system and on the deformation and bending of zircaloy claddings during ballooning
- Experiments on shortened fuel rod simulators to assess separate effects of the ballooning process of zircaloy claddings
- First burst test on a fuel rod simulator with 3,90 m heated length and on axial power profile.

6. Results

For characterization of the fuel rod simulators with regard to the temperature distribution a photographic method in connection with an evaluation of the films by micro-densitometer with respect to different blacknesses has been proved successfully. Further potentials for improving the fuel rod simulators were identified by this method. The heated rods exhibited a good axial temperature distribution with small local temperature deviations. Furthermore it has been shown that such temperature differences on the heater rod will be damped substantially up to the zircaloy cladding of the fuel rod simulator.

The internal design of the fuel rod simulator was fixed by optimization calculations with a non-steady state heat conduction program. A comparison of calculated cladding temperature transients of the fuel rod simulator and a nuclear fuel rod in a quasi-steady state adiabatic heat-up phase followed by different cooling periods exhibited a good agreement.

For experimental verification of the thermal behavior of the fuel rod simulator steady state and non-steady state experiments were performed. All experiments and calculations proved the good thermal simulation quality of the fuel rod simulator.

To obtain information on the significance of the gas flow in the fuel rod simulator from the plena to the ballooning area further theoretical work has been carried out. Such a gas flow through the annulus between pellets and cladding influences the pressure in the balloon and therefore the whole deformation behavior of the cladding. A quasi-compressible, quasi-steady-state gas flow model was derived /8/ and tested by experiments. From these experiments, friction factors were evaluated. In the smooth annuli, the laws for laminar flow and for fully developed turbulent flow in smooth geometries are verified well. It can be seen that for laminar flow the Hagen-number in the friction factor f = Ha/Re is depending systematically from the hydraulic diameter of the annulus, whereas the turbulent flow is independent from the dimensions of the annulus. Pellets hinder the formation of laminar flow. In comparison with laminar flow in smooth geometries, an increase of the friction factors can be stated even at low Reynold numbers.

First test calculations by applying the flow model to simulate ballooning processes show that only small gap widths result in substantial pressure differences whereas large gap widths give only small deviations between balloon and plena pressures. Consequently, the influence of gap widths on the mass flow between plena and balloon is low. A further result of these test calculations is that for gas flow during ballooning the region of low Reynolds numbers is of special interest.

For measuring the internal gas pressure of the fuel rod simulator in the area of the balloon a special measuring system with minimized dimensions, consisting of a capillary and a micro strain gauge pressure transducer with a very small measuring chamber has been developed. For evaluating the pressure signal, basic theoretical work on the response time characteristic of this measuring system has been performed. It has been proved that this response measuring system lends itself well to determine the local ballooning and burst pressure in fuel rod simulators.

Heat-up and burst tests with fuel rod simulators under internal overpressure exhibited circumferential temperature distributions, a clad lifting on the colder side and a subsequent rupture on the hotter side, which did not lift from the pellets. This special deformation pattern of the zircaloy claddings may be produced by anisotropy effects of the zircaloy material and/or by mechanical forces. To clarify the influence of eventual mechanical effects, the forces and momentums induced by different thermal expansions of heater rod and zircaloy cladding and transmitted to the cladding have been calculated. It has been shown that in the inelastic region the cladding gives way to the pairs of momentums built up in the elastic region. This may result in the special deformation pattern described above. It remains to be studied by further experiments and theoretical evaluations if the anisotropy effect has possibly the predominant influence.

Ballooning experiments on shortened, indirectly heated fuel rod simulators brought the following results:

- At an internal rod pressure of 70 bar circumferential burst strains in the order of 36% were found; higher internal rod pressures result in lower strains.
- The values determined for the strain rate and temperature correspond to normal plastic behavior.
- Depending on the development of straining, there may be azimuthal temperature differences on the zircaloy cladding of up to 100 K.
- The zircaloy cladding always bursts at the hottest spot with the highest strain, i.e., the most pronounced weakening of the wall thickness.
- The surface line of the cladding tube on which the point of rupture is located in most cases is a straight line; it practically does not raise above its heat source.
- The opposite colder side of the rod shows a marked lifting of the zircaloy cladding tube without any weakening of the wall thickness.

To clarify the specific deformation shape of the claddings observed at these tests some experiments with directly heated empty tube specimens have been performed. These tests resulted in circumferential burst strains over 100 % and axial shrinkages of the burst zircaloy claddings.

At recent experiments with indirectly heated fuel rod simulators with 325 mm heated length axial shrinkages of the zircaloy claddings were measured as well, after having modified the internal structure in a way, that no more internal constraints occur. Circumferential burst strains of about 30 % and axial shrinkages of about 1,5 mm were determined. These data are well within the lower scatter band of the burst strains measured at previous tests.

The test rig for full-length bundle tests has been commissioned. The first full-length fuel rod simulator with a stepped axial power profile has been subjected to a burst test with subsequent flooding. The resulting burst strain amounted to 27% at a burst pressure of 76 bar and a burst temperature of 808° C. The circumferential strains measured over the total heated length of 3900 mm exhibited an extremely high sensitivity to the different temperatures in the general power steps as well to local thermal and mechanical influences caused by spacers and thermocouples. The cladding around the burst has the same deformation shape known from the previous tests with shortened fuel rod simulators e.g. straight surface line at the side of rupture, lifted oppposite side.

7. Next Steps

Important steps to be taken include the following main aspects:

- separate effect tests on shortened fuel rod simulators to the ballooning process
- single rod tests on full-length fuel rod simulators
- preliminary measurements for bundle tests
- testing and calibrating of several unconventional measuring devices
- theoretical work on evaluation of the experiments and verification of the SSYST modul DRUSPA.

8. Relation with other projects

Other theoretical and experimental investigations on fuel rod behavior during a LOCA performed by GfK (PNS 4231, 4235, 4236, 4237, 4239) and KWU (RS 107) and on transient events in the reactor core and primary circuit in the low pressure phase of a LOCA, performed by KWU (RS 36).

9. References

/1/ 1st PNS-Semi-Annual Report 1974, KFK 2050 (German)

/2/ 2nd PNS-Semi-Annual Report 1974, KFK 2130 (German with English abstracts)

/3/ 1st PNS-Semi-Annual Report 1975, KFK 2195 (German with English abstracts)

/4/ 2nd PNS-Semi-Annual Report 1975,
KFK 2262 (German with English abstracts)

/5/ 1st PNS-Semi-Annual Report 1976,
KFK 2375 (German with English abstracts)

/6/ F. Erbacher, H.J. Neitzel, M. Reimann, K. Wiehr
"Out-of-pile Versuche zum Aufblähvorgang von Zircaloy-Brennstabhüllen in der Niederdruckphase eines Kühlmittelverluststörfalles"
Reaktortagung 1976, Düsseldorf 30. März - 2. April

/7/ F. Erbacher, H.J. Neitzel, M. Reimann, K. Wiehr
"Out-of-Pile Experiments on Ballooning in Zircaloy Fuel Rod Claddings in the Low Pressure Phase of a Loss-of-Coolant Accident"
Paper SNI 8/25, Specialists Meeting on the behavior of water reactor fuel elements under accident conditions, Spatind, Norway, 13th - 16th September 1976

/8/ M. Reimann
"Analytische Untersuchung von Gasströmungen in Ringspalten beim Aufblähvorgang von Zirkaloy-Hüllrohren"
KFK 2280, Mai 1976

10. Degree of Availability of the Reports

Unrestricted distribution.

140-21-31/4112-04

Classification : 1-3 429

Titre Eléments combustibles : Comportement des gaines de zircaloy pendant un accident de perte de caloporteur : programme EDGAR.	Pays : FRANCE
	Organisme directeur : CEA/DSN - EdF
Titre (anglais) Zircaloy cladding behaviour during a loss of coolant accident. EDGAR programme	Organisme exécuteur : CEA/DTECH-SRMA (Saclay)
	Responsable : M.CHAGROT (DSN - SETSSR) P.MORIZE (DTECH)
Date de démarrage : 01/01/74 Date prévue d'achèvement : 31/12/80 Etat actuel : en cours Dernière mise à jour : 01/03/77	Scientifiques : R.ROULLIAY

Objectif général :

Déterminer les déformations et le niveau d'oxydation des gaines de zircaloy soumises à des transitoires de températures et de pression représentatives d'un accident de perte de caloporteur des réacteurs à eau sous pression.

Objectifs particuliers :

1) Effectuer les essais de simulation sur des gaines neuves et sur des gaines irradiées.
2) Réaliser des vérifications des modèles de déformations utilisés sans les codes de calcul existant.
3) Elaborer de nouveaux modèles plus représentatifs de la réalité.

Installations expérimentales et programme :

1) Les essais sur les gaines neuves sont réalisés par le DTECH-SRMA à Saclay. L'appareillage permet de chauffer le tube par effet Joule suivant un programme préétabli. De même, la surpression interne imposée au tube est programmée en fonction du temps. Les essais peuvent être réalisés sous vide, sous vapeur d'eau ou en atmosphère inerte.
2) Un dispositif analogue est en cours d'étude en vue de son installation en laboratoire chaud pour étudier le comportement des gaines préirradiées.

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Etat de l'étude :

1) Avancement à ce jour :

La première phase de l'étude a été achevée. Elle a consisté à s'assurer que les résultats à pression et vitesse de chauffe constante sont en bon accord avec les résultats publiés à l'étranger.

2) Résultats essentiels :

On a pu généraliser les résultats obtenus et tracer :
- Des abaques indiquant le temps pour atteindre une déformation donnée dans des conditions de pression initiale et de vitesse de chauffage donnée,
- Les courbes maitresses de déformation en fonction de la contrainte pour des températures et vitesses de chauffage donnée.
- Ces résultats ont été publiés à la réunion de spécialistes du CSNI à Spatind (Norvège) en septembre 1976.

Prochaines étapes :

La deuxième phase de l'étude consistera en 1977 à mesurer la déformation des gaines lorsqu'elles sont soumises aux transitoires de température et de pression déterminés par les calculs thermohydrauliques, et vérifier la validité des modèles de thermomécanique. La troisième phase consistera en 1977 et 1978 à réaliser des itératifs entre calculs et expériences. Parallèlement en 1977, on poursuivra l'étude et la construction du dispositif d'essai en laboratoire chaud.

Relation avec d'autres études :

Le programme doit servir de support technique au programme PHE[illegible]

Documents de référence :

"Comportement des gaines en zircaloy pendant un accident de r[redacted]i-
dissement d'un réacteur à eau ordinaire - analyse bibliograp[redacted]e -
propositions pour les études CEA", Note DTECH-SRMA/73-533, [redacted] 1973.
"Comportement des gaines en zircaloy pendant un accident de [redacted]oi-
dissement - point des essais au 15 février 1976", Note DTEC[redacted]MA/GMM
(76)2883, avril 1976.
"Comportement du gainage en zircaloy des éléments combustib[redacted] des
réacteurs à eau sous pression pendant un accident de refro[redacted]ssement -
programme EDGAR", CSNI Specialist Meeting, Spatind (Norvèg[redacted] septembre
1976.

Titre	Pays :
Eléments combustibles : Comportement des gaines de zircaloy pendant un accident de perte de caloporteur : programme EDGAR.	FRANCE
	Organisme directeur : CEA/DSN - EdF
Titre (anglais) Zircaloy cladding behaviour during a loss of coolant accident. EDGAR programme	Organisme exécuteur : CEA/DTECH-SRMA (Saclay)
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Date de démarrage : 01/01/74 Date prévue d'achèvement : 31/12/80 Etat actuel : en cours Dernière mise à jour : 01/03/77	Scientifiques : R.ROULLIAY

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Objectifs particuliers :

1) Effectuer les essais de simulation sur des gaines neuves et sur des gaines irradiées.
2) Réaliser des vérifications des modèles de déformations utilisés sans les codes de calcul existant.
3) Elaborer de nouveaux modèles plus représentatifs de la réalité.

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Relation avec d'autres études :

Le programme doit servir de support technique au programme PHEBUS.

Documents de référence :

"Comportement des gaines en zircaloy pendant un accident de refroidissement d'un réacteur à eau ordinaire - analyse bibliographique - propositions pour les études CEA", Note DTECH-SRMA/73-533, avril 1973.

"Comportement des gaines en zircaloy pendant un accident de refroidissement - point des essais au 15 février 1976", Note DTECH-SRMA/GMM (76)2883, avril 1976.

"Comportement du gainage en zircaloy des éléments combustibles des réacteurs à eau sous pression pendant un accident de refroidissement - programme EDGAR", CSNI Specialist Meeting, Spatind (Norvège), septembre 1976.

<table>
<tr><td colspan="2">Titre

Fonctionnement de l'installation PHEBUS.</td><td>Pays :
FRANCE</td></tr>
<tr><td colspan="2"></td><td>Organisme directeur :
CEA/DSN</td></tr>
<tr><td colspan="2" rowspan="2">Titre (anglais)

Operation of the PHEBUS facility.</td><td>Organisme exécuteur :
CEA/DSN-SES</td></tr>
<tr><td>Responsable :
A. TATTEGRAIN (SES)</td></tr>
<tr><td>Date de démarrage : 01/01/76
Etat actuel : Etude en cours</td><td>Date prévue d'achèvement : 31/12/82
Dernière mise à jour : 18/11/76</td><td>Scientifiques :
J. BLAUD
PH. DELCHAMBRE
G. MANET.</td></tr>
</table>

Objectif général :

Cette action couvre les opérations de mise en service, de conduite et de maintenance de l'installation PHEBUS.

Etat de l'étude :

Avancement à ce jour :

Le recrutement du personnel est pratiquement terminé. Sa formation aux problèmes des installations à eau pressurisées a été assurée auprès d'autres installations de type PWR. Les vérifications des matériels en usine et sur place sont en cours.

Prochaines étapes :

Les essais de sous-ensembles doivent commencer en mars 1977.

Relation avec d'autres études :

Certains moyens en personnel et en matériel, en particulier les installations d'acquisition et de traitement de données sont communs aux trois réacteurs PHEBUS, SCARABEE, CABRI.

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.

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.

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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

Classification: 2.1	
Title 1 (Original Language): Auswertung von WASH 1400 bzgl. Energiebilanzen im deutschen Kernschmelzprogramm (RS 72 C - I.1.5, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Evaluation of the WASH 1400 Report with Respect to Coremelting (Energy Balances)	Project Leader: H. Goetzmann H. Hassmann
Initiated (Date): 1. 9. 75 Status: Completed Completed (Date): 31. 3. 76 Last Updating (Date): 31. 12. 76	

1. General Aim

Study of the chapters of WASH-1400 dealing with coremelting problems (Appendix VIII).

2. Particular Objectives

Comparison of models and results in WASH 1400 and RS 72 a, b with respect to the expected course of a hypothetical core melt down accident.

3. Research Program

Study of Appendix V, VII and VIII of the Rasmussen report.

4. Experimental Facilities

No experimental facilities necessary.

5. Progress to Date

The evaluation of App. V, VII and VIII of the Rasmussen Report was completed. The study was completed.

6. Results

The evaluation of the Rasmussen-report showed that the physical phenomena of the hypothetical core melting were in good agreement with the model in RS 72.

Important data, which are to be verified experimentally (e.g. the slumping temperature of the fuel rods) are more realistic in RS 72, which uses experimental results. In WASH-1400 the decay heat was lower, as the analysis was started 1 h after blowdown. A direct comparison of the calculated values is therefore not possible.

Another important difference with respect to the pressure increase in the containment was the fact, that the american power stations use concrete with limestone aggregates, which produces CO_2 after heating. The melting temperature is lower than 1000° C. In the FRG concrete with basalt aggregates is used. Experiments (RS 160) have shown, that concrete with basalt aggregates has better chances at higher temperatures.

7. Next Steps

Work has been completed.

8. Relation with Other Projects

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9. References

Final Report RS 72c.

10. Degree of Availability

The report is available on request from GRS, Köln.

Classification: 2.1	
Title 1 (Original Language): Kernschmelzen: Theoretische Untersuchung der Abschmelzphase (RS 73 - I.1.5; Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Core Meltdown Program Theoretical Investigation of the Different Phases of the Meltdown Process	Project Leader: H. Goetzmann H. Hassmann
Initiated (Date): 1. 10. 72 Completed (Date): 31. 3. 76 Status: Completed Last Updating (Date): 31. 12. 76	

1. General Aim and 2. Particular Objectives

The aim of this investigation was to develop a computer code which describes the core heat-up until failure of the core support structure of the hypothetical meltdown accident.

3. Research Program

For the core heat-up in case of a hypothetical melt down accident the computer code MELSIM has been developed.
The following phases are modeled :

Decrease of the water level in the core region:
Characterized by the evaporation of the residual water in the core by the decay heat generated in the fuel pins still covered by water.

Heat-up of the core: Heat up of the fuel pins not covered by water due to steam cooling, the fission products release from fuel and the chemical reaction of the steam with the fuel cladding. Because slumping of fuel in the core and heat up of the core support structure is considered too, it is possible to calculate the time interval until failure of the core support structure occurs.
Both phases had to be investigated theoretically.

4. Experimental Facilities

No experimental facilities were necessary.

5. Progress to Date and 6. Results

a)	SLUMP:	The experimental results of GfK were used.
b)	NACHZERF:	The NACHZERF-results are compared with those of RIBD. The agreement of the Shure-curve was quite well.
c)	KUMI:	In collaboration with GKSS KUMI was improved. Some corrections were made.
d)	KOCH:	The water level decrease model from RS 72 b was introduced into KOCH.
e)	Test example:	A test example was designed, tested and evaluated. Calculations of the core heat-up phase with MELSIM will be continued in RS 183.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

RS 183 Energy Balances Hypothetical RPV Failure with Respect to Concrete Destruction.

9. References

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10. Degree of Availability

Classification: 2.1	
Title 1 (Original Language): Untersuchungen thermohydraulischer Vorgänge sowie Wärme- und Stoffaustausch in der Coreschmelze (RS 48/1 - I.1.5., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: TU-Hannover
Title 2 (english): Theoretical and Experimental Investigation of the Thermohydraulic Behaviour of a Molten Core at the Bottom of the Reactor Pressure Vessel	Project Leader: Prof. Dr. Mayinger
Initiated (Date): May 1971 Completed (Date): April 1975 Status: finished Last Updating (Date): December 1975	

1. General aim

In this research project the thermohydraulic behaviour of molten reactor core at the bottom of the reactor pressure vessel both theoretically and experimentally had been studied. In the program different geometries and boundary conditions were assumed and calculated. The experiments which had been made for verification the computer codes, had affirmed the numerical treatments.

2. Particular objects

Of particular interest was the determination of the heat flux density from the core debris to the wall of the reactor pressure vessel and at the surface of the molten core. In addition the temperature increase in the debris and the convection behaviour were determined. This studies were not limited to the specific geometry of the reactor pressure vessel, but were also extended to horizontal fluid layers. This had to be done by experiments and numerical investigations.

3.1 Experimental facilities

The experiments were made up with the help of the holographic interferometry using water as modelling fluid. This optical method allows to make visible the whole temperature field in the fluid. This temperature field can be recorded and from it the local heat transfer coefficient can be determined. For these investigations two-dimensional models were used and the examined fluid had to be transparent. The volumetric heat sources were simulated by Joules' heat. More detailed information about this measuring technique can be found in /1/.

3.2 Research program

In the theoretical part of the project the computer code THEKAR was established, which describes the thermohydraulic behaviour and the heat transfer in the molten core in different geometries. This includes a horizontal layer, the rotationally symmetrical case of a cylinder and a semisphere, which corresponds to the real conditions within a LWR-reactor.

The experimental studies had the aim to verify the calculated situations in the rectangular and semicircular geometry, essentially the measurement of the heat transfer coefficient. In addition experimental investigations were performed using models in the shape of a segment of a circle, the height of the models was smaller than the radius of curvation. Using these models the situation should be simulated when the bottom of the reactor vessel is only partially filled up.

4. Projekt status

4.1 Progress to date

The research project run out at April 30th, 1975. The theoretical part had been finished some month before and the experimental work in the last period had been done essentially with the interpretation of the results, which we obtained before. In addition to our previous results we were able to establish some empirical correlations of heat transfer in the geometry of a segment of a circle.

With this experimental results, we could introduce a model for the turbulence into the program-code THEKAR. With this model we were able to compute free convection at very high Rayleigh-numbers, which are expected for the molten core in the case of accident.

The results of this research program are explained in the final report in all details (1).

In addition to the equations we gave in earlier reports, we set up correlations for the heat transfer in a semispherical geometry.

For the upper plane wall we obtained the equation

$$Nu = 0.40 \ Ra'^{0.2} \qquad (1)$$

and for the semispherical bottom the equation

$$Nu = 0.55 \ Ra'^{0.2}$$

These equations are valid for the averaged Nusselt-numbers. Particularly at the bottom local values of the Nusselt-numbers differ from the averaged values with increasing Rayleigh-numbers. The maximum value lies closely to the upper wall in this case.

5. Next steps

The continuation of the work is now done in the project RS 166.

6. Relation with other projects

RS 73: Theoretical Core Meltdown Studies
KWU Erlangen, IKE-Stuttgart 1.10.72 - 31.12.75

7. Reference documents

Quarterly reports in the series: IRS Forschungsberichte (German)
Jan. 1975 - March 1975 IRS-F-25
Apr. 1975 - Jan. 1976 IRS-F-26

/1/ Untersuchung der thermohydraulischen Vorgänge sowie Wärmeaustausch in der Kernschmelze
BMFT-Forschungsvorhaben RS 48/1, Abschlußbericht 1975 (German)

8. Degree of availability

The reports are available at the IRS.

Classification: 2.1	
Title 1 (Original Language): Theoretische und experimentelle Untersuchung des Verhaltens eines geschmolzenen Kerns im Reaktorbehälter und auf dem Betonfundament (RS 166 - I.1.5, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: T.U. Hannover
Title 2 (English): Theoretical and Experimental Investigation of the Thermohydraulic Behaviour of a Molten Core in the Reactor Vessel and on the Concrete of the Basement	**Project Leader:** Prof. Dr.-Ing.F. Mayinger
Initiated (Date): May 1975 **Status:** Continuing **Completed (Date):** Dec. 1978 **Last Updating (Date):** December 1976	

1. General aim

The aim of this research project is to establish computer codes, which describe the thermal interaction of the molten core with various components of the reactor. In order to test the reliability of the codes some experiments with modelling fluids have to be done.

2. Particular objectives

Of particular interest is the heat transfer from the molten core material to the wall of the reactor pressure vessel, while the molten material is flowing into the bottom of the vessel.

Another important point is 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. The heat and mass 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.

3. Research program

In the first step of the research program a computer code will be established, describing the thermohydraulic behaviour of the molten core material at the bottom of the reactor vessel in the melt down phase.

During this period the molten material is continously flowing down into the bottom of the pressure vessel. The code shall be able to handle the variation of physical properties and of the heat source density in the melt with time and space. One result of this code is the local magnitude of the heat flux from the melt to the pressure vessel wall. A number of modelling experiments is provided to verify the code. For the experiments, a model with a semicircular bottom is used to simulate the semispherical pressure vessel bottom. To measure the temperature field the holographic interferometry is used, while the velocity field is recoreded with a Laser-Doppler-Anemometer.

The second point of the research program is the investigation of the heat transfer between the concrete of the reactor foundation and the molten core material laying on it. This heat transfer is mostly influenced by the vapour, escaping from the concrete. A computer code is developed, which is able to predict the thermohydraulic behaviour at the melt concrete interface including steam or gas bubbles formation. Additional, experiments will give informations about the bubble dynamics and the heat transfer from an internal heated fluid to a cold wall, where from gas bubbles arise.

Results of the numerical and experimental work will lead to a model, describing the heat transfer from the melt to the concrete.

4. Experimental facilities and computer codes

To predict the thermohydraulic behaviour and the heat transfer from th melt to the bottom of the reactor pressure vessel with consideration of inflowing material, the code THEKAR K2 is established. This code is an extension of the code THEKAR K, and it is able to handle variable physical properties and heat sources in the melt. The basis of this code is the numerical solution of the conservation laws for mass, momentum and energy by finite differences.

The interaction between the melting concrete and the liquid core material is described by the code BETON. This code is able to handle two separate phases including the effect of their surface tension. The two phases can be molten core and molten concrete or molten core and vapour.

To restrict the computer time on a tolerable quantity, only a relative small area of the boundary between the two phases at the interaction zone is computed, the rest can be regarded as isothermal with good approximation.

To verify the numerical treatments experiments are provided. In the fluid layer with in flowing material, temperature and velocity fields are measured. The temperature fields are recorded by the help of the holographic interferometry. With this measuring technique the temperature field in the whole test chamber can be recorded instantanously without disturbing the thermohydraulic behaviour of the fluid. A Laser-Doppler-Anemometer is used to measure the velocities in the fluid.

To investigate the local heat transfer from a pool of molten material to its boundaries with turbulent convection, measurements were done with the holographic interferometry, too.

For the verification of the code BETON experiments are performed with low melting materials and with gas injection into a pool of internal heated water.

5. Progress to date

The computer codes THEKAR K2 and BETON are mostly completed. The code THEKAR K2 is used to compute the heat flux to the wall of the reactor vessel for a number of inflow rates and different boundary conditions. The code BETON was first used to calculate the behaviour in a system of two liquids with different density.

The experiments with low melting material and the measurements in a fluid layer with inflowing material could be finished in the last year.

Furtheron, the interferometric measurement of the local heat flux at a vertical wall of a pool with internal heated fluid at turbulent free convection was completed. For the investigation of the influence of gas bubbles on the heat transfer to the wall the test chamber is constructed and the experiments are prepared.

6. Results

The thermohydraulic behaviour and the heat transfer to the wall could be determined for the combined free and forced convection in a fluid layer with inflowing material. The onset of a perceptible influence of the inflowing fluid could be described by the dimensionless number Gr/Re^2. The Gr-number related to the free convection in the layer, while the Re-number is related to the inflowing fluid. Two cases were investigated more intensively:

1. The temperature of the inflowing melt is equal to the wall temperature, i.e. in a reactor accident the inflowing material has just melting temperature.
2. The inflowing fluid is hot, i.e. its temperature is like the highest temperature in the fluid layer.

For the onset of the influence of forced convection, the following equations were derivated:

Case 1: $$\left(\frac{Gr}{Re^2}\right)'_{krit} = 20.83 \cdot Ra'^{0.265} \cdot \left(\frac{s}{h}\right)$$

Case 2: $$\left(\frac{Gr}{Re^2}\right)'_{krit} = 0.235 \cdot Ra'^{0.265}$$

In these equations s is the diameter of the inflowing fluid beam while h is the height of the fluid layer.

The figures 1 and 2 show two examples of the distribution of the incoming material in the melt at the bottom of the reactor pressure vessel, computed with the code THEKAR K2. In figure 1, heavier material is flowing into the melt. It penetrates the molten pool and forms a layer at the bottom. In figure 2 lighter material flows into the pool and spreads out at the top of the pool.

An example for the results computed with the code BETON is shown in figure 3. A bubble is just rising from the bottom of the fluid layer and the convection in the fluid initiated by the rising bubble has a great influence on the heat transfer to the bottom. At each side of the bubble a distinct maximum in the local Nu-number can be seen.

An example for the local Nu-numbers measured at a vertical wall of a closed test chamber at turbulent free convection is plotted in figure 4. The local Nu-number has a maximum at the top of the wall were the hot fluid flows out of the core to the wall. A second maximum in the local heat flux exists near the bottom. This maximum results from a recirculation flow at the lower part of the wall. The mean Nu-number for the whole wall can be represented by the equation

$$Nu = 0{,}85 \ Ra'^{0{,}19}$$

for Ra'-numbers up to 4 10^{13}.

7. Next Steps

The results computed with the code BETON together with the results of the gas injection experiments shall lead to a heat transfer model for the heat flux from the molten core to the solid concrete. Therefore at first the experiments must be completed.

6. Relations with other projects

RS 73
Theoretical core melt down studies
TU Erlangen, IKE Stuttgart 1.10.72 - 31.12.75

RS 154
Investigations of the interaction between molten core material and concrete, KWU Erlangen 1.2.75 - 30.9.76

RS 183
Energy balance after a hypothetical reactor pressure vessel failure under consideration of the concrete destruction
KWU Erlangen 1.9.75 - 31.5.77

7. Reference documents

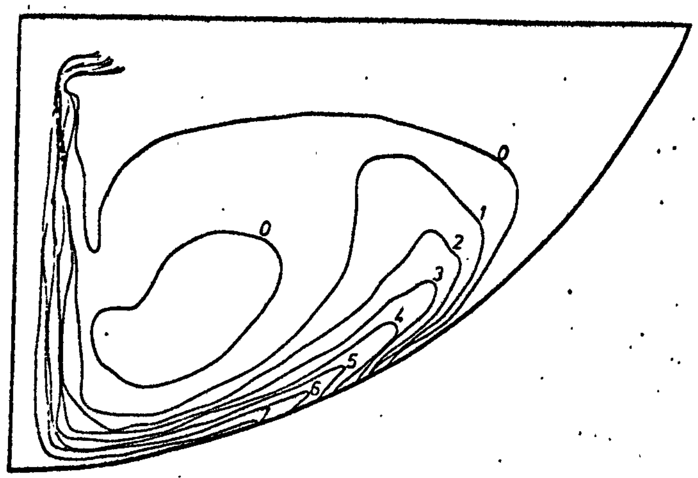
uarterly report in the series: IRS Forschungsberichte (German)

Report period:	Jan 1976 - Mar 1976	IRS F-30
	Apr 1976 - Jun 1976	IRS F-31
	Jul 1976 - Sep 1976	IRS F-32
	Oct 1976 - Dec 1976	IRS F-33

Annular report 1975 (German)

8. Degree of availibility

The reports are available at the GRS, Köln



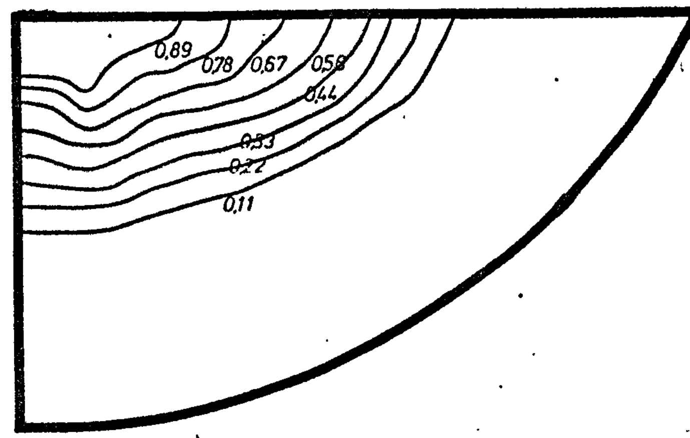
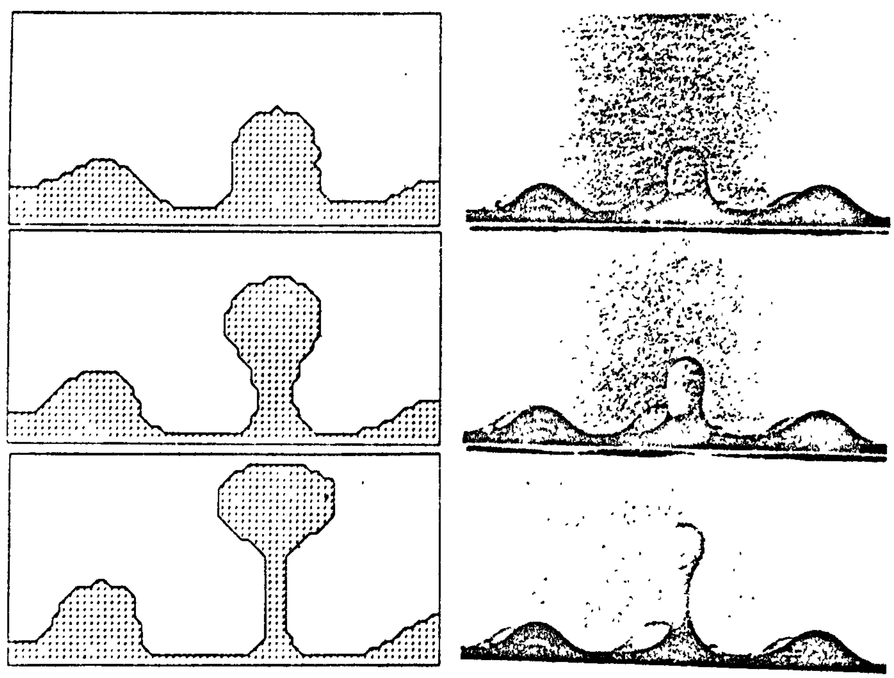


Fig. 1 Concentration distribution with heavier stream

Fig. 2 Concentration distribution with lighter stream



$Ra = 7.2 \cdot 10^{10}$
$Pr = 7$
$We = 2 \cdot 10^{7}$
$\Delta\varrho = 0.087$

$\sigma = 0.0071 \ N/m$
$\Delta\varrho = 87 \ kg/m^3$
$L = 0.05 \ m$

computed

experimental (parafin/water)

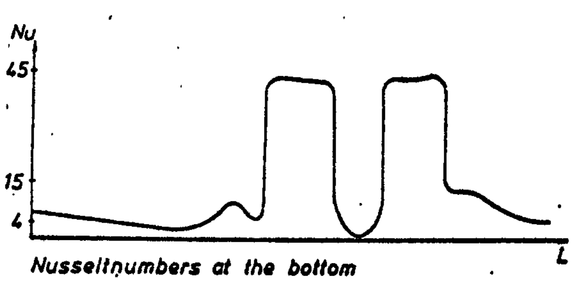


Fig. 3 Rising bubble of lighter material and heat flux distribution at the bottom

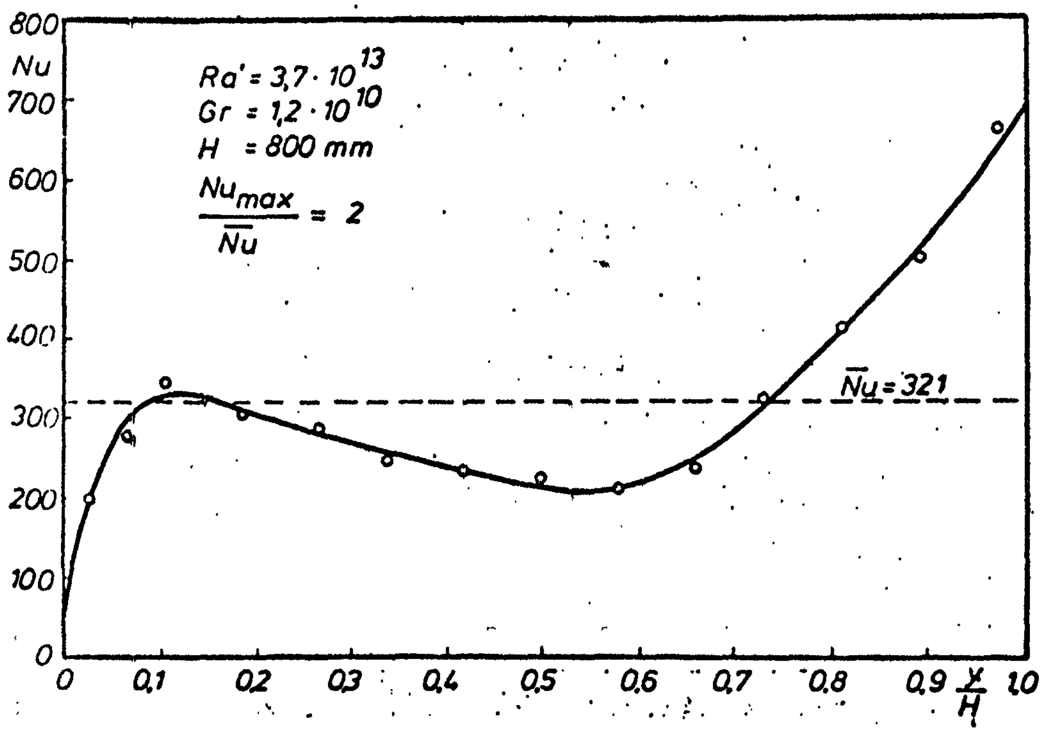


Fig. 4 Local Nusselt numbers at the wall with turbulent free convection in a closed test chamber

Classification: 2.1	
Title 1 (Original Language): Experimentelle Untersuchung der Abschmelzphase von UO_2-Zircaloy-Brennelementen bei versagender Notkühlung (PNS 4241 - I.1.5, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: ORGANIZATION: PNS GfK Karlsruhe
Title 2 (English): Experimental Investigations of the Meltdown Phase of UO_2-Zircaloy Fuel Rods under Conditions of Failure of Emergency Core Cooling	Project Leader: Dr. S. Hagen RBT/IT
Initiated (Date): 1973 Status: Continuing Completed (Date): 1978 Last Updating (Date): December 1976	

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

In 1976 we have mainly done experiments with single rods in steam at different heating rates and experiments with single rods with spacers in helium and steam.

3./4. Research programme and experimental facilities

Details for the programm are given in ealier reports.

In this year we have put into operation two new facilities:

1. A new vessel for fuel rod simulators with heating by central tungsten rods. It allows rods up to 1 m, bundle arrangements up to 7 x 7 rods and flowing steam.
2. A furnace was built, which can heat up rods with solid pellets by radiation.

5. Progress to date

In the reporting period we have done experiments on single rods in steam with different heating rates between 0.75 and 3.5 ^{o}C/sec.
Experiments on the influence of the spacers on the meltdown behaviour were carried out in helium and steam up to temperatures of 1830 ^{o}C.
The oxide layer formed in steam between rod and spacer reduces strongly the interaction. In order to simulate the influence of oxide layers which are formed during normal operation of an LWR, we have heated up to 1800 ^{o}C in He preoxidized rods with layers between 10 and 50 µm.

6. Essential results

The experiments with different heating rates in steam show, that in all cases a uniform smooth oxide layer is formed. The thickness of this oxide layer decreases with heating rate. In the range investigated, it changes from roughly 80 % to 40 % of the original Zircaloy can. At ca. 1900 ^{o}C little holes are formed through which the melt of unoxidized Zircaloy and the UO_2 poured out. Even the thinnest of the oxide layers stayed intact during heat up with the exception of the formation of the small holes.

While we have in He a strong interaction between Inconel and Zircaloy, in steam the attack of the spacer on the can is strongly reduced by the formation of the oxide layer. In He the interaction starts at 1000 ^{o}C. At 1250 ^{o}C it begins to influence also the region outside the rod. The can in the region of the spacer is melted at 1600 ^{o}C and at 1800 ^{o}C also large parts of the can are melted outside the region of the spacer.

In steam we have quite a different behaviour. Up to 1200 ^{o}C no sign of interaction is found. At 1250 ^{o}C the first small changes at the can-spacer interface can be seen. Near 1450 ^{o}C the contact-spring of the spacer is deformed and only at roughly 1800 ^{o}C the whole spacer is molten. In contrast to heating in He the influence of the spacer is restricted to its contact region and also here the oxidized Zircaloy remains intact and shows only splits and deformations.

The heat up of preoxidized rods in He shows that already a 10 µm thick oxide layer causes the beginning of interaction between can and spacer at roughly 1350 ^{o}C. The 50 µm oxide layer causes the beginning of interaction in the neighbourhoud of a rod heated in steam. With this thickness we found no meltdown of the can outside the region of the spacer.

7. Next steps

To see the influence of the surroundings on the meltdown of the rod our next experiments are done in bundle geometry of different size up to 7 x 7 rods. Most of these experiments are done in steam with some in helium. We also plan to do experiments with spacers and guide tubes.

8. Relation with other projects

RS 73 : Theoretical investigation of the different phases of the meltdown process.

RS 2o5: Calculation of fuel pin meltdown experiments and application to MELSIM

Classification: 2.1	
Title 1 (Original Language): Kernschmelzen: Messung von Stoffwerten von flüssigen Reaktorcorematerialien (RS 80 - I.1.5., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: EURATOM, Ispra
Title 2 (english): Core Melting - Measurement of Physical Properties of Liquid Reactor Core Materials	Project Leader: R. Palinski
Initiated (Date): 1.12.1972 Status: continuing Completed (Date): 31.12.1975 Last Updating (Date): December 1975	

1. General aim

Core melting - Measurement of physical properties of liquid reactor materials

2. Particular objectives

Measurement of the viscosity and surface tension of core melts

3. Experimental facilities and program

- Measurement of viscosities
  - Method: Oscillating crucible
  - Materials: UO_2 and mixtures of UO_2, Zircaloy and steel ("Corium") (homogeneous phases only, to be identified by research project RS 74a).
- Measurement of surface tensions
  - Method: Maximum bubble pressure
  - Materials: Same as above

4. Project status

4.1 Progress to date

In the reference year the work was concerned with the measurement of the viscosity of the components of the core melt, namely UO_2, steel 1.4550 and zircaloy 4. Furthermore, a method for making samples of the core melt was developed and their viscosity was measured.

The modification of the facility for measuring surface tensions was terminated. As the depth of immersion of the capillary tip into the melt is required for the evaluation of the results, it is necessary, when immersing the capillary tip into the melt, to detect exactly the moment of the first contact with the surface. For this purpose a device was developed and tested which signalizes the contact both optically and accustically. For temperature testing experiments up to 2000°C have been made with the surface tension measurement facility. Further test measurements up to 3000°C are under preparation.

4.2 Essential results

Measurements of the viscosity of zircaloy 4. The average of three measurements in TaC-graphite crucibles was 4.7 cP at the melting point.

Steel 1.4550 and its components. The viscosity of Armco iron at the melting point (1510°C) is 7.15 cP according to the only hitherto known measurement /1/. This relatively high value, compared to the viscosity of pure iron (5.54 cP at 1532°C /1/ , 4.72 cP at 1536°C/4/), is attributed to the influence of the oxygen in the Armco iron. Our measurement of Armco iron resulted at the melting point a viscosity of 6.2 cP. The chemical analysis of the oxygen content of the utilized material is intended to give an indication whether this deviation may be due to a difference in oxygen concentration.

The viscosity of nickel at the melting point can be considered as established, since the values of /1/, /2/ and /3/, obtained with two different measurement methods, agree within 4%. According to /1/ the viscosity is 4.98 cP. Our control measurement resulted a value of 4.81 cP.

The measurements of the viscosity of steel 1.4550 gave at about 1460°C the values 6.0 cP and 6.22 cP.

Fabrication of core melt samples and measurement of their viscosity. The necessary samples were made as follows: in order to guarantee a simultaneous melting of the components and to reduce the reaction of the melt with the crucible, pellets were pressed from powder of UO_2, zircaloy and steel in corresponding weight distribution and sintered 10 minutes at 2300°C during the heating-up process for measuring the viscosity.

For determining the viscosity of the core melt in TaC the measurements were made in TaC-graphite crucibles (because of the small wall thickness of the TaC crucible of only 0.7 mm, necessary in order to obtain during the carburization of the Ta a homogeneous composition of the TaC, the TaC crucible was imbedded into a graphite crucible which guaranteed sufficient mechanical solidity).

For the determination of the viscosity of the core melt in ThO_2, crucibles of the company Desmarquest with high density and high purity were available. As in our experimental facility the outer diameter of the ThO_2 crucibles must not exceed 20 mm, only about 30 g of the core melt could be contained. For a sufficiently precise measurement, however, at least twice this quantity is necessary. Therefore two ThO_2 crucibles were inserted, one above the other, into a TaC-graphite crucible as described above. In the temperature range of the measurements the TaC crucible is well compatible with the ThO_2 crucible and prevents a reaction between ThO_2 and graphite. The lower ThO_2 crucible is closed by the bottom of the upper one. The upper one has a lid of ThO_2 onto which a further lid of TaC is placed. In order to solidify this system of super imposed crucibles by some stress in axial direction and to compensate the thermal expansions, a graphite felt was inserted between the TaC lid and a graphite lid which closes the graphite crucible. During a series of preliminary tests this arrangement of crucibles prooved to be the most suitable. Three viscosity measurements have been made in this way, determining the apparatus constant before each measurement.

For the calculation of the viscosity from (1) the density of the core melt in the liquid state must be known.

$$\eta = \frac{1}{\gamma}\left(\frac{\Delta\delta}{K}\right)^2 \quad (1)$$

η - Viscosity
γ - Density of the core melt
$\Delta\delta$ - Change of the logaritmic decrement of the amplitudes during transition from the solid to the liquid state
K - Apparatus constant

As the density of the core melt is not known so far, the result of four measurements is quoted as product $\eta\gamma$.

η_{WF} - Viscosity of the core melt during formation.
η_{NF} - Viscosity of the core melt after formation.
t - Measuring time.

Table 1

	$\eta_{WF}\delta$	t (min)	$\eta_{NF}\delta$	t (min)	core melt quantity (g)
I Measurement in TaC-crucible	0.437	-	-		62.4
II " " ThO_2 "	0.437	2	0.214	2.5	70.8
III " " " "	0.380	-	0.197	5	60.8
IV " " " "	0.420	4	0.221	2.5	71

Measurement II during the formation extended over 2 minutes and represents the average over 16 periods of the damped oscillations. The temperature increase during this time was of about 10°C. The investigated core melt was composed of the main components 35 w/o UO_2, 10 w/o zircaloy 4 and 55 w/o steel 1.4550 which according to /5/ at 2400°C are all liquid. At this temperature the formation of corium E is terminated after approximately 2 minutes. The values $\eta\delta$ were measured during this formation time. After terminated formation dropped by about 50% (measurements II, III and IV). During the measurement II after terminated formation the temperature increased by another 10°C. The strong decrease of $\eta\delta$ after terminated formation cannot be explained by this temperature increase, but is presumably the result of a change of the melt by the formation.

Assuming that the density of the liquid core melt lies about 15% below the density at 20°C, the following viscosity values are obtained:

Table 2

	η_{WF} (cP)	η_{NF} (cP)
I	5.7	-
II	5.7	2.8
III	5.0	2.6
IV	5.5	2.9

The arithmetic average is respectively $\eta_{WF} = 5.5$ cP

$\eta_{NF} = 2.8$ cP

References

/1/ Hiebler, H. and H. Trenkler: Berg- und Hütte. Monatshefte 12 (1967) P. 150/163.

/2/ Cavalier, G.: C. R. Acad. Sci. 256 (1961) P. 1308/1311.

/3/ Schenk, H., M. Frohberg and K. Hoffmann: Arch. Eisenhüttenwes. 34 (1963) P. 93/99.

/4/ Thiele, M. : Dissertation Techn. Univ. Berlin 1958, Inst. f. Metallkunde.

/5/ Peehs, M., "Investigations of Molten 'Corium' Phases", Thermodynamics (Proc. Symp. Vienna, 1974) 1, IAEA, Vienna (1975) 355.

5. Next steps

Measurement of viscosity of core melts and their components	January - December 1976
Measurements of surface tension of core melts and their components	January - December 1976

6. Relation with other projects

RS 71: "Research and development studies on the measurement of molten reactor core materials, compatibility studies between these materials and crucible materials", BATELLE-Institute e.V.

RS 74a: "Research project core melting: 5. investigation of the metallurgical interaction between melt and reactor pressure vessel wall", KRAFTWERK UNION AG.

7. Reference documents

Quarterly reports in the series IRS-FORSCHUNGSBERICHTE

Report period	Oct.	1973 - Dec.	1973	IRS - F - 19
" "	Jan.	1974 - March	1974	IRS - F - 20
" "	Apr.	1974 - June	1974	IRS - F - 21
" "	July	1974 - Sept.	1974	IRS - F - 22
" "	Oct.	1974 - Dec.	1974	IRS - F - 23
" "	Jan.	1975 - March	1975	IRS - F - 25
" "	Apr.	1975 - June	1975	IRS - F - 26
" "	July	1975 - Sept.	1975	IRS - F - 27
Annual report	Dec.	1972 - Oct.	1973	IRS - F - 18
Annual report	State:	End of	1974	IRS - F - 24

8. Degree of availability:

Restricted

Classification: 2.1	
Title 1 (Original Language): Nachrechnung von Stabexperimenten und Absicherung von MELSIM (RS 205 - I.1.5, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Inst.f.Kernenerg Univ.Stuttgart
Title 2 (English): Calculation of Fuel Pin Meltdown Experiments and Application to MELSIM	Project Leader: Prof.Dr.-Ing. H.Unger Dr.-Ing.F.Schmidt
Initiated (Date): June 1, 1976 Status: Continuing Completed (Date): November 30, 1978 Last Updating (Date): December 31, 1976	

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 module 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 module 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) als 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. A model in order to calculate the fuel pin heatup in helium atmosphere has been developed. It has been modified in order to calculate experiments in air and H_2O-vapor. The Zr-H_2O-reaction is included in the model.

to 3.2. W, UO_2 and Zr have different expansion coefficients. This causes changes in the gap widths which can be calculated by the model. Therefore, the time at which UO_2 and Zr interact mechanically can be calculated.

6. Results

to 3.1. The calculated results and measured data are in good agreement. The deviations are less than 50 K (3 %). This is satisfactory with regard to uncertainties in the emissivity.

7. Next Steps

to 3.1. Additional experiments will be calculated with the model described.

to 3.2. The diffusion between Zr und UO_2 will be modelled also as well as the meltdown of pins.

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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Classification: 2.1	
Title 1 (Original Language): Technologie und Stoffwerte von LWR-Core-Schmelzen (RS 200 (PNS 4245) - I.1.5, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT/GfK/PNS ORGANIZATION: GfK / PNS
Title 2 (English): Technology and properties of Corium melts	Project Leader: Dr.G.Ondracek
Initiated (Date): 1975 — Completed (Date): 1978/79 Status: Continuing — Last Updating (Date): December 1976	

1. General aim

The general aim is to provide data on the behaviour of the core materials in a core melt down accident with a view to assessing its course and consequences.

2. Particular objectives

Preparation of various types of core melts (Corium) that arise under varying oxidation conditions that could prevail in the accident environment. Determination of properties such as thermal conductivity, thermal expansion, density, als well as the nature of the melt that exists in contact with concrete.

3. Experimental facilities

Available equipment includes a dilatometer/D.T.A., an apparatus for measuring contact angles via sessile drop technique. Furthermore experimental facilities are available for the metling and powder metallurgical consolidation of samples as well as measurement of thermal conductivity.

4.1 Progress to date

Estimates of properties such as thermal conductivity, viscosity and heat capacity, have already been carried out and partly verified. Methods of preparation of Corium melts under inert gas conditions have been established. Experiments are under way to determine the interaction of Corium with concrete.

4.2 Essential results

It has been demonstrated that large quantities (~250 g) of Corium EX1 can be prepared by induction heating in Thoria crucibles in inert gas. Thoria is not a suitable material for the preparation of Corium EX3. Boronnitride seems to be promising, although some Boron (5 wt.%) is present in the solidified melt.

Initial experiments on the reaction of oxidized Corium EX1 with concrete (BN 250, 12 - 13.1 wt.% Ca, 0.6 - 0.7 wt.% Fe, 1.3 - 1.4 wt.% Al, 41.1 - 46.1 wt.% O_2, 2.3 - 3.9 wt.% C, rest Si) under the following conditions: Concrete to Corium volume ratio ≥ 1; 1 atm steam, max. temperature 1800 K, time at max. temperature 5 min: have revealed that the case in all mixtures, the uranium and zirconium are present as oxides along with the oxides fo the concrete, whereas Iron, Chromium and Nickel are still metallic.

5. Next steps

Estimation of the surface energies of the core components in context of a steam explosion.

Measurements of density and thermal expansion of Corium to determine the probability of a seggregated melt. Determination of the nature of the Core melt in contact with the concrete containment.

6. Relations with other projects

PNS 4244

7. Reference documents

KFK Report 2262 (1976) in German

"Eigenschaftsabschätzungen für LWR-Coreschmelzprogramm" und "Technologie und Eigenschaften von Corium" , Reaktortagung, Düsseldorf, 30.3.-2.4.76, Deutsches Atomforum e.V., 1976: S.248-51 and 260-63.

8. Degree of availability

Unrestricted distribution.

<table>
<tr><td colspan="2">Classification: 2.1</td></tr>
<tr><td>Title 1 (Original Language):
Messung der Hochtemperaturviskosität von ausgewählten Substanzen, die im Zusammenhang mit einem möglichen Kernschmelzunfall wichtig sind
(RS 214 - I.1.5, Jahresbericht A 76)</td><td>COUNTRY:
BRD
SPONSOR:
BMFT
ORGANIZATION:
Battelle-Institut</td></tr>
<tr><td>Title 2 (English):
Measurement of the High-Temperature Viscosity of Selected Substances Which Are Important in Connection with a Possible Core-Melting Accident</td><td rowspan="2">Project Leader:
Dipl.-Phys.
R. Skoutajan</td></tr>
<tr><td>Initiated (Date): June 1, 1976
Completed (Date): September 30, 1977
Status: Continuing
Last Updating (Date): December 31, 1976</td></tr>
</table>

1. General Aim

Measurement of the viscosity of an oxidic core melt of Corium-EX3 type and its components.

2. Particular Objectives

Modification of the available high-temperature rotational viscosimeter of Searle type into Couette type

3. Research Program

Concept, design and construction work for a measuring system consisting of thorium oxide crucibles, for the other components of the viscosimeter, and for non-reusable and auxiliary equipment; functional testing.

New findings, according to which thorium oxide crucibles are not sufficiently resistant to a Corium-EX3 melt /1,2/, have called for a modification of the research program to include use of more suitable crucible materials (cf. Section 7).

4. Experimental Facilities

The work carried out in the report period consisted in the constructionof the experimental facilities.

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5. <u>Progess to Date</u>

5.1 Concept work.

5.2 Preparation of design drawings for all components of the viscosimeter and for the non-reusable equipment.

5.3 The construction work for the followings parts is under way and partly completed:

- Holder for rotary crucible
- Rotary shaft transmission
- Temperature measuring equipment
- Induction furnace for viscosimeter
- Induction furnace for hot pressing equipment
- Sintering template (fixing of ThO_2 axles in ThO_2 crucibles by means of presintered granular ThO_2)
- Male and female carbon molds for hot pressing of TaC-susceptors
- Molded graphite parts for thermal insulation.

5.4 Functional testing of rotary shaft transmission, vacuum equipment and the two induction furnaces.

5.5 Fabrication of granular ThO_2 from powdered ThO_2; preliminary experiments for fixing a ZrO_2 axle in ZrO_2 and MgO crucibles by means of granular ThO_2.

5.6 Construction of the hot pressing apparatus.

5.7 Additional work has become necessary in the report period as Zircoa declared that they were unable to grind further crucible pairs in addition to the four pairs supplied to date.

Design and construction of a grinding machine for ThO_2, taking into account the radiation protection regulations; construction of components of the grinding machine, and functional testing of some of these parts.

6. Results

The concept, design and bench work was carried out on schedule until it turned out that the research program had to be modified (cf. Section 3). The results of the functional tests of the parts completed by that date were satisfactory.

7. Next Steps

In accordance with the proposed modification of the research program, new - preferably metallic - crucible materials will be selected and examined for their resistance to oxidic melts. It is planned to measure the decrease in thickness of the crucible and to determine the chemical composition of the melt. Parallel to the above work, concept work for adapting the high-temperature viscosimeter to the selected crucible material will be carried out.

8. Relation With Other Projects

The Research Project BMFT-RS 200 covers the fabrication and characterization of Corium-EX3 samples.

9. References

In the report period no interim reports were prepared by Battelle.

(1) Quarterly Report V76/3 on Research Project RS 200

(2) KFK 2220: Reaktions- und Schmelzverhalten der LWR-Corekomponenten UO_2, Zircaloy und Stahl während der Abschmelzperiode. July 1976

10. Degree of Availability of the Reports

The above cited references are available upon request from
Institut für Reaktorsicherheit/Forschungsbetreuung
Glockengasse 2
5000 Köln

Classification: 2.1

Title 1 (Original Language):	COUNTRY: BRD
Studie über die Relevanz und die Durchführbarkeit von Messungen der Wärmetönungen chemischer Reaktionen beim Kernschmelzen und über den integralen Wärmeinhalt von Kernschmelzen (RS 197 - I.1.5, Jahresbericht A 76)	SPONSOR: BMFT
	ORGANIZATION: Battelle-Institut e.V.
Title 2 (English):	**Project Leader:**
Study on the Relevance and Feasibility of Measurements of the Heat Effects of Chemical Reactions Taking Place During Core Melting and on the Integral Heat Content of Core Melts	Dipl.-Phys. R. Skoutajan
Initiated (Date): February 10, 1976 — Completed (Date): December 31, 1976	
Status: Completed — Last Updating (Date): December 31, 1976	

1. General Aim

Investigation of the heat effects during a hypothetical core-melting accident.

2. Particular Objectives

Calculation of heat effect data and development of measurement concepts.

3. Research Porgram

The research program modified and approved in September 1976 covers the following items:

3.1 Investigation of the relevance of chemical reactions for the core-melting accident

3.1.1 Evaluation of the literature concerning the integral generation of after-decay heat as a function of time

3.1.2 Calculation of the after-decay heat

3.1.3 Compilation of the possible chemical reactions occuring during the core-melting accident

3.1.4 Listing of the reaction enthalpies and reaction rates known from the literature

3.1.5 Calculation of the timedependent generation of heat of reaction

3.1.6 Comparison of the nuclear and the chemical portions of heat generated during the core-melting accident

3.2 Feasibility study on the measurement of heat effects of chemical reactions during a core-melting accident and of the integral heat content of core melts

3.2.1 Definition of the quantities to be measured

3.2.2 Selection of the most suitable measuring methods and selection of the crucible material, the heating method and the temperature measurement on the basis of theoretical considerations and calculations

3.2.3 Concept work for a measuring apparatus including data on its expected measuring accuracy on the basis of theoretical considerations and calculations.

4. Experimental Facilities, Computer Codes

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5. Progress to Date

Ad 3.1: The heat effects of chemical reactions were calculated on the basis of most recent literature data. These values were compared with the amount of heat set free by the nuclear after-decay reaction.

Ad 3.2: Various methods of high-temperature calorimetry (differential scanning calorimetry, adiabatic calorimetry and isothermal calorimetry) were studied and compared.

6. Results

Ad 3.1: Considering a 1,200-MW pressurized-water reactor, an amount of energy of 150 GWs is released by the nuclear after-decay reaction within a period of 1500 s from the beginning of the accident. Within this period, the energy released by the zirconium-steam reaction exceeds that of nuclear after-decay considerably. Substantial amounts of heat can also be liberated if steel (in particular its chromium components) is oxidized by the steam, and even more so if oxygen present in the containment takes part in the oxidation reaction.

Ad 3.2.2: According to the literature, adiabatic calorimetry is the only method which has so far been applied successfully in the temperature range above 1500 °C. Differential scanning calorimetry seems to be suited for the temperature range between 1800 and 2200 °C. Calculations of the heat exchange by radiation in an isothermal calorimeter have shown that this method is only applicable if a temperature difference as small as 1 °C can be measured and maintained.

Ad 3.2.3: A measuring apparatus according to the principle of a chemical continuous-flow reactor has been devised for investigating the reaction between metallic core melt and steam. It can be used to decide whether the heat is released during bubble formation or during ascent of the bubbles in the melt and will make it possible to obtain information about the kinetics of the above reaction. Use of an automatic microbalance will ensure a high measuring accuray.

7. Next Steps

The results will by summarized in the final report.

8. Relation with Other Projects

RS 154 (KWU) Untersuchungen zur Wechselwirkung von Reaktorbeton und Kernschmelze

RS 72a and 72b (KWU) Theoretische Aufstellung der Energiebilanzen. Bilanzgrenze RDB-Wand und Bilanzgrenze Containmentwand für DWR und SWR.

RS 74a (KWU) Untersuchung der metallurgischen und chemischen Wechselwirkung von Kernschmelze und Reaktordruckbehälterwand

PNS 4244 (GfK) Versuche zur Erstellung von Zustandsdiagrammen und zum Reaktionsverhalten von Corekomponenten sowie phasenanalytische Charakterisierung von Kernschmelzexperimenten

PNS 4245 (GfK) Stoffwerte von LWR-Coreschmelzen

9. References

Technical report

RS 197: Study on the Relevance and Feasibility of Measurements of the Heat Effects of Chemical Reactions.

September 1976, Battelle-Institut e.V., Frankfurt am Main

10. Degree of Availability of the Reports

the above-cited report is available upon request from

Institut für Reaktorsicherheit/Forschungsbetreuung

Glockengasse 1

5000 Köln

Classification: 2.1	
Title 1 (Original Language): Experimente zur Simulation großer Kernschmelzen (Vorprojekt) (PNS 4246 - I.1.5, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GFK/PNS
Title 2 (English): Experiments on Simulated Large Core Melts	Project Leader: D. Perinić RBT/IT
Initiated (Date): 1976 Completed (Date): Status: Continuing Last Updating (Date): Dec. 1976	

1. General Aim

Verification of melt front propagation and fission product release models in scale-up experiments.

2. Particular Objectives

2.1. Investigation of the penetration of the melt front.

2.2 Investigation of combined influences of thermohydraulic and reaction behaviour and their influence on activity release.

2.3. Investigation of the influence of pool depth and circulation on the release of fission and activation products.

2.4. Investigation of layer and crust formation, boiling and evaporation of the melt.

2.5. Investigation of phases and phase distribution in the melt because they are related to distribution of fission and activation products.

2.6. Investigation of atmospheric oxidation at the surface of the melt and of internal oxidation caused by the gases released from concrete decomposition.

2.7. Investigation of the long-term behaviour and of possibilities of melt cooling in concrete.

2.8. Investigation of the formation, diffusion and sedimentation behaviour of aerosols released from the melt.

2.9. Investigation of hydrogen generation and limitation of the explosion risk.

3. Research Programme

3.1. Corium melt experiments in the range of 100 to 1000 kg in concrete crucibles (particular objectives 2.1 ... 2.9).

3.2. Corium melt experiments in the range of 100 to 1000 kg in cooled crucibles (particular objectives 2.7).

4. Experimental Facilities

4.1. Preliminary experiments:

- SASCHA melting furnace (max. 5 kg)
- THERMITE melting furnace (max. 400 kg)

4.2. An experimental facility for 100 ... 1000 kg melts should be constructed.

5. Progress to Date

A detailed work programme was written which specifies the work required and the date of completion. A feasibility study simulating large core melts was begun. Preliminary investigations were conducted with three different melt masses (0.1 - 1 - 10 t). The applicability of the familiar methods of heating was investigated. Several technical data of the experimental facility were determined. The following investigations were performed in addition: chemical reactions on the melt surface, gas filtering, melt composition, fissium production and analysis, temperature measurement, handling of radioactive waste, experimental building.

Two types of experiments were carried out to investigate the melt-concrete reaction and to develop stable concrete crucibles.

- 0.5 kg corium was melted in concrete crucibles.

- 100 ... 300 kg molten iron were poured into large concrete crucibles. The use of glass fibre reinforcement in the crucible set a limit to the width.

6. Results Obtained

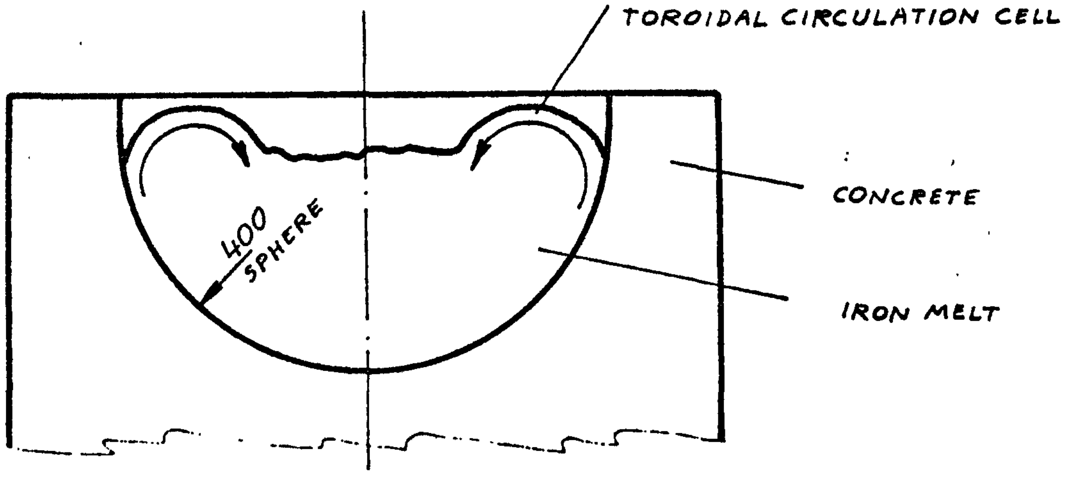
It was found that the experiments on activity release, surface oxidation, crust formation, boiling and evaporation processes call for methods such as induction heating which do not disturb the natural temperature gradients of the surface.

The maximum power generated within the melt was determined. In case of an accident, the fission product power density at the melting front is estimated at 0.7 MW/m^2. In order to match this condition, we calculated the necessary electric net power:

melt mass	kg	100	1000	10000
melt power for H = D	MW	0.213	0.987	4.586
melt power for 4H = D	MW	0.268	1.243	5.771

The maximum local power density at the melt front may be as high as [illegible] MW/m^2 because of cooling of the melt.

To simulate the accident atmosphere, a pressure vessel withstanding 6 bar will be required. In any case, the melt will be covered by steam. However, the containment pressure may be simulated by non-condensable gas (with cold pressure vessel walls) or by steam (with hot walls). Preliminary experiments will be required to investigate the electrical discharge within these atmospheres.



A convective melt circulation in a shallow crucible was observed in two pour-in experiments. At a melt temperature of about 1400°C in cool crucibles (approx. 5°C), a toroidal circulation cell developed as shown in the figure. The velocity of this movement appeared to be faster than expected from free bubble movement.

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

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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).

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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 - March 1973	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 - March 1974	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) Personnel : 3.7 men/year

11) Additional information :

Classification: 2.2	
Title 1 (Original Language): Experimentelle Untersuchungder Dampfexplosion (RS 76-L. 1. 5. "Jahresbericht A 76"	COUNTRY: BRD SPONSOR: ORGANIZATION: JRC Ispra
Title 2 (English): Experimental Investigation of the Vapour Explosion	Project Leader: H. Kottowski F. Toselli
Initiated (Date): 1.11.1972 — Completed (Date): 31.12.1977 Status: Continuing — Last Updating (Date): 31.12.1976	

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 in a core.

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.

The basic questions to be answered in a situation where molten materials come into contact with the coolant are: what is the amount of generated mechanical work, in what manner is it released and what are the mechanisms determining the fragmentation of the molten materials.

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

The aim of the theoretical studies is the development of interaction models and calculation codes for the estimation of the conversion rate of thermal energy into mechanical work and pressure load of the reactor vessel.

3.2 Experimental Studies

3.2.1 Basic Oriented Shock Tube Experiments

These experiments are performed in order to provide data in a mono-dimensional test device to check and to adjust the physical interaction model and the calculation code. The 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 15o 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 fuel 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.

An attempt has been made to visualize the fragmentation 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 Channel Experiment

- The two-dimensional computation code KAMI has been tested with constant physical parameters,

- molten stainless steel tests have been repeated for confirmation,
- **Al-H_2O tests for the validation of the test facility have been performed.**

5.2 Tankexperiment

- Interaction experiments with molten stainless steel (AISI 347) and water at small melt to coolant volume ratio (smaller than 1:1000) have been terminated,
- fabrication and sintering of 40 kg of UO_2 pellets,
- preparation of experiments with UO_2 (granulates or molten).

6. RESULTS

6.1 Theoretical Work

A two-dimensional computation code describing its interaction process in the channel experiment has been developed and tested with constant physical property values. The application of the developed extrapolation equations (which are rational functions of polynomials of 7th and 6th order) for the computation of the physical properties of water in the supercritical range (up to 2500°C and 10,000 bar) encountered numerical stability problems, which have not yet been resolved completely.

6.2 Channel Experiments

Tests have been performed with stainless steel and Al. The scope of the Al-tests was to check whether in the available test facility explosion like melt-water interactions are feasible. 5 Tests were performed with Al-melt of 750°C and water of 25 respectively 80°C.

The pressure peaks measured in all experiments were in the range of 70 to 130 bar. The stainless steel tests confirmed low pressure generation ($<$ 11 bar) measured in previous tests.

6.3 Tankexperiments

A series of 28 interaction experiments on stainless steel/water have been performed; 15 of them were successful.
The experimental conditions were as follows:

stainless steel weight	1.5 and 3 kg
stainless steel temperature	1600 to 1800°C
H_2O temperature	20, 80 and 230°C
gas content of H_2O	saturated and degassed
H_2O volume	$\sim$ 300 l
system pressure	1 to 25 bar.

Two different velocities of melt entering into the water were used: pouring it from 10 cm above the water level or destroying the crucible bottom at the end of the 3 m fall.

None of the performed experiments resulted in a vapour explosion. The maximum pressure produced by the interaction was about 2 bar.
Always a very poor fragmentation was observed; slightly higher fragmentation occurred when the melt entered the water at higher velocity.
Motion of the split melt and boiling effects during its cooling have been visualized 70 cm below the water surface by high-speed camera (up to 2000 frames per sec).

7. FUTURE STUDIES

7. 1 Channel Experiments

- Termination of the stainless steel tests at 25 bar system pressure and 200°C water temperature,
- UO_2-granulates/water tests,
- molten UO_2-water tests.

7.2 Tankexperiment

- Interaction experiments of UO_2-granulates or molten UO_2 with water at the same volume ratio used for the steel tests,
- interaction experiments of steel and UO_2 with water at melt/coolant volume ratio simulating reactor-like conditions.

8. RELATIONS WITH OTHER PROJECTS

No new relations with respect to 1975 Annual Report.

9. REFERENCE DOCUMENTS

- 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,
- 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.

10. DEGREE OF AVAILABILITY OF THE REPORTS

All Reports available.

Classification: 2.2	
Title 1 (Original Language): Theoretische Simulation von Dampfexplosionen in Tankgeometrie, Entwicklung geeigneter Fragmentationsmodelle (RS 206 - I.1.5, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** Inst.f.Kernenerg. Univ. Stuttgart
Title 2 (English): Theoretical Investigation of Vapor Explosions in Pool-Type-Geometry, Development of Models Describing the Fragmentation Process	**Project Leader:** Prof.Dr.-Ing. H. Unger Dipl.-Ing.R.Benz
Initiated (Date): May 1, 1976 **Status:** Continuing **Completed (Date):** July 31, 1978 **Last Updating (Date):** December 1976	

1. General Aim

Within the frame of the research project core meltdown experimental and theoretical calculations on hypothetical vapor explosions in light water reactors are performed. The theoretical acticities initiated are intended to lead to a broadening of the knowledge on conditions, course and extend of vapor explosions possibly during a hypothetical core meltdown.

2. Particular Objectives

The aim of the research project can be subdivided into the following tasks

- 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 for the fragmentation
- 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 at Ispra.

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 /1/, 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.

4. Experimental Facilities, Computer Codes

In order 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. A further development will be necessary however.

5. Progress to Date

The estimation of the strongly transient and the quasi-static pressure distribution in reactor geometry (PWR and BWR) has been completed.
The distribution of mechanical energy generated by vapor bubble collapse at the boundary between two different liquids has been investigated regarding large changes in the thermodynamic conditions. A lit-

erature study has been performed in order to find information on the minimum film boiling temperature and the corresponding heat fluxes under film boiling conditions for a horizontal plate and a horizontal cylinder in pool-type geometry.

6. Results

Unless a coherent reaction of more than 1 metric tons of melt with a substantial amount of water occurs, no serious damage of the pressure vessel or the containment is to be expected (entrapment was not considered).

With respect to the bubble collapse model it has been found that for both, mild and strong pressure pulses most of the mechanical energy generated will remain in the water.

7. Next Steps

The development of the bubble collapse model will be continued. Investigations on the minimum film boiling temperature will be performed for different conditions.

8. Relation with Other Projects

For pool-type-experiments performed at the EURATOM-center at Ispra (research project BMFT RS 76) theoretical calculations are done. The results obtained experimentally are used for the theoretical model.

9. References

/1/ Benz, R., G. Fröhlich, H. Goldammer, H. Kottowski, H. Unger
Theoretische Studien zur Dampfexplosion, 2. Technischer Fachbericht zum Forschungsvorhaben BMFT RS 76, Universität Stuttgart, Institut für Kernenergetik, April 1976

/2/ Benz, R., W. Schwalbe, H. Unger
Ingenieurmäßige Abschätzung der Energiefreisetzung und des Druckaufbaus bei Dampfexplosionen in Reaktorgeometrie, 1. Technischer Fachbericht zum Forschungsvorhaben BMFT RS 206, Universität Stuttgart, Institut für Kernenergetik, Dezember 1976

10. Degree of Availability of the Reports

Gesellschaft für Reaktorsicherheit (GRS) mbH, Postfach 10 16 50, 5000 Köln 1, Fed. Rep. of Germany

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 : March1975	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.

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, Zircalo" Inconel and UO_2.

4) Project status

1. Progress to date :

The Tank Facility :

- Fabrication and assembling of the supporting structure and the whole circuitry including furnace, interaction tank, pumps, vessels, valves and vacuum system is completed.
- Testing of the interlock system completed.
- Instrumentation for measuring and recording pressures and temperatures during the interaction in the tank has been commissioned.
- Adaptation of a high speed camera in order to visualize the interaction process (at least at low system pressure) is underway.
- Preliminary experimental studies of a filter system to collect and split the debris produced during the thermal interaction are underway.
- The mechanical device for catching and turning the crucible to drop the melt in the centre of the reaction tank has been fabricated and tested.
- A high frequency induction furnace for degassing the core melt materials has been adopted.

The Channel Facility :

- The mounting of the test-rig was started at the end of 1974
- The instrumentation for pressure, temperature and void measurements has been prepared.
- The crucible and heating system for preparing molten UO_2 are being tested in the light of the successful experience on UO_2/Na interactions.
- Theoretical developments for the assessment of "mild" fuel/coolant interactions are underway.

2. Essential results

The first experiments are expected in mid-1975.

5) Next steps : Completion of the test facilities and initiation of the experiments.

6) Relation to other projects :

These studies are part of the "core meltdown" analysis programme of the BMFT. The reference numbers of the project dealing with the same subject are :

RS 72a, RS 726, RS 73, PNS 4241, RS 48/1, RS 74 a, RS 746, PN 4243, RS 71, RS 79, RS 80, PNS 4311, PNS 4242.

7) Reference documents :

G. Fröhlich, H. Kottowski, F. Toselli
Theoretische und experimentelle Untersuchungen über die Wechselwirkung geschmolzener Materialien und Kühlmittel
IRS-Seminar, November 1974

Quarterly progress reports

JRC 1974 Safety Programme Progress Report

8) Degree of availability : Freely available

9) Budget : The expected total cost of the experiment, including the investment and the running costs are :

BMFT : about 123 000 UA
CEC : about 210 000 UA

10) Personnel : 10 men/year

11) Additional information : -

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

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- 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

500

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.

501

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 Status :	Completed : 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.

.70+
507

Classification: 2.3	
Title 1 (Original Language): Forschungsprojekt Coreschmelzen: 5. Untersuchung der metallurgischen Wechselwirkung zwischen Schmelze und RDB-Wand (RS 74 a - I.1.5, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Investigation of Metallurgical and Chemical Interactions Between Molten Phases of Reactor-Core-Material and the Reactor Pressure Vessel	Project Leader: Dr. Peehs
Initiated (Date): November 1972 Status: Completed Completed (Date): July 1975 Last Updating (Date): December 1975	

General Aim

Within the research program the core melting-phenomena after the two-fold improbable case of a loss-of-coolant accident and a simultaneous drop out of the emergency cooling system basic data were evaluated for the measurement of material constants of the molten phases and theoretical considerations of the reactor in the post accident stage.

Particular Objectives

It was the objective of these investigations

a) to determine the metallurgical behaviour of reactor-core material during melting and to investigate the constitution of the molten phases.

b) to study the interaction of the molten phases of the reactor core material with the pressure vessel of a LWR.

Experimental Facilities / Research Program

The first subtask in the program was to define the characteristic overall composition of the molten phases of core materials of LWR. As this composition varies from the beginning to the equilibrium stage, the core-melting had to be defined for beginning (Corium A) and for the equilibrium stage (Corium E).

The direct study of "Corium"-melting behaviour as second subtask was to be done by melting-experiments in a crucible within an inertgas atmosphere. Subsequent to the melting experiments metallographic and microprobe analysis as well as remelting experiments by direct microscopic observation had to be carried out to identify liquidus and solidus temperatures of "Corium", the disintegration mechanism of the high-melting components and the phase-composition in the "Corium".

To investigate the interaction of liquid "Corium" and reactor vessel materials as the third subtask a special test facility had to be developed. To achieve representative test conditions the possible quantity of materials to be used for interaction should be in the range of 1 - 2 kg. The foreseen parameters for the experimental study were the temperature gradient across the interaction zone, the temperature and the interaction time.

Project Status / Progress to Date

The investigations of the interaction of Corium A and the RPV wall material and the oxygen influence have been completed experimentally.

Project Status / Essential Results

At temperatures over 2000 °C the material loss from Corium during its primary liquefaction will be 20 - 25 % of the molten mass. With the molten samples the different evaporation behaviour of the Corium components has been analyzed. The evaporation behaviour of Corium depends on the O-activity of the gas phase and the steel content of the melt.

The melting temperatures of the RPV material in contact with Corium A and Corium E at 1400 °C was experimentally confirmed. Low temperature liquefactions of RPV material due to alloying effects will not occur.

Next Steps

Work on this project has been completed.

Relation with Other Projects

RS 154 Coremelting: Investigation of the Interaction between Molten Core and Concrete

RS 160 Experimental Investigation of the Interaction between UO_2-Steel Molten Core and Graphite

RS 183 Energy Balances after Hypothetical RPV Failure with Respect to Concrete Destruction

Reference Documents / Degree of Availability

No reports available.

Classification: 2.3	
Title 1 (Original Language): Forschungsprojekt Coreschmelzen: 5. Untersuchung der metallurgischen Wechselwirkung zwischen Schmelze und RDB-Wand, Feasibilitystudie (RS 74 b - I.1.5, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Feasibility Study on Experimental Fuel Element Slumping Investigations With Respect to Coremelting	Project Leader: Dr. Peehs
Initiated (Date): 1. 1. 73 Status: Completed	Completed (Date): 30. 4. 75 Last Updating (Date): 31. 12. 75

General Aim

The objective of the R & D - work was to establish a study about the feasibility of large technical experiments to investigate fuel element slumping and the interaction between melt of burned down fuel and vessel wall of reactor pressure vessel material.

Particular Objectives

The feasibility of large technical experiments was to be tested with respect to the melt down of burned out fuel elements in the hot cells. The heat, necessary for the melting down of the fuel, was to be provided by the decay energy of the fission products and secondary heating elements. Another objective was to find out if the volatile fission product release during melt down and fuel slumping. For this purpose the released fission products were to be identified and continuously measured during the experiments.

Experimental Facilities and Program

1) Identification of possible reactor and the fuel element for preirradiation
2) Dimensioning of the test bundle and check of the hot cell equipment available.
3) Design study for the experimental devices.

Project Status/Progress to Date/Essential Results

The BWR VAK is best reactor facility for pre-irradiation of test elements. The proximity of the reactor to the hot cells in Karlstein, allows to start the meltdown test already 100 h after reactor shutdown. For the meltdown experiments the middle part of a segmented element can be used with an active length of 1000 mm. The test facility "BAUTZ" has a total height of 2510 mm. Since the decay heat of a fuel rod of 0.4 % of the max. heat rate, available 100 h after reactor shutdown, is too low for test element slumping in a cool environment, it is necessary to simulate the irradiation equilibrium of an inner element, in the reactor core by auxiliar heating elements. Because of the small decay heat of 0.4 % the support heaters must be heated up to the fuel-element slumping temperature of 2300 °K. Consequently only refractive metal heaters can be used. The meltdown test needs therefore to be executed in an inert-gas atmosphere.
In the design of the test facility "BAUTZ" results of a safety analysis were integrated, resulting, that lack of electric power supply and leakages in the different coolant loops are controllable.

Next Steps

Work on this project has been completed.

Relation with Other Projects

RS 154: Coremelting: Investigation of the Interaction between Molten Core and Concrete

RS 160: Experimental Investigation of the Interaction between UO_2-Steel-Molten Core and Graphite

RS 183: Energy balances after hypothetical RPV Failure with Respect to Concrete Destruction

Reference Documents/Degree of Availability

M.Peehs, K. Mollwitz, W. Würtz
Studie über die Durchführbarkeit von Kernschmelzversuchen mit abgebrannten Brennelementen in einer HEISSEN ZELLE
Abschlußbericht Förderungsvorhaben BMFT RS 74 B
Kraftwerk Union Aktiengesellschaft (Nov. 1975)

Company Confidential

Classification: 2.3	
Title 1 (Original Language): Experimentelle Untersuchung der Wechselwirkung UO_2-Stahl-Kernschmelze mit Graphit (RS 160 - I.1.5, Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** KWU, Erlangen
Title 2 (english): Experimental Investigation of the Interaction Between UO_2-Steel-Molten Core and Graphite	**Project Leader:** Dr. Peehs
Initiated (Date): February 1975 — **Completed (Date):** December 1975 **Status:** Completed — **Last Updating (Date):** December 1975	

General Aim and Particular Objectives

The tests conducted under RS 74 A have shown, that graphite can resist considerably a melting core, which contains zirconium. This effect results from a ZrC-protective layer, which constitutes of the Zr-content of the corium.

For the fast sodium cooled reactor the behaviour of a Zr-free molten corium has to be investigated. Therefore in addition to the results of RS 74 A the barrier properties of graphite will be studied, using a typical fast reactor composition of the molten materials.

Experimental Facilities

Within this task the same experimental facilities are used as with the investigation of the metallurgical and chemical reaction between Corium and RPV-material.

Research Program

1. The chemical reactions in the system UO_2-Graphite will be investigated, using various graphite qualities and gaseous atmospheres.

2. The chemical reactions in the system Steel-UO_2-Graphite will be investigated, using various graphite qualities and gaseous atmospheres.

Project Status / Progress to Date

A standard test was worked out experimentally; the compatibility of UO_2, Steel 1.4981 and UO_2 + Steel 1.4981 on graphite was investigated.

Project Status / Essential Results

UO_2 will be dissolved completely at 1980 °C in an UO_2-Steel-Graphite-system; pure UO_2 pellet is liquidated on graphite at 2375 °C. Microanalysis showed that FeCrNi-, UFe- and UFeCr-phases have been produced. The experimental results show, that a rather strong oxigen loss due to an CO-generation occurs during the melting process. This results together with material evaporation in a rather high weight loss during the UO_2 dissolution. The carbon activity in the melt reaches within minutes its final value. The interaction of the melt and the graphite mass stops.

Next Steps

The tests are completed. A final report with a detailed evaluation of experimental results will be prepared.

Relation with Other Projects

A 74 A: Investigation of Metallurgical and Chemical Interactions between Molten Phases of Reactor-Core-Material and the Reactor Pressure Vessel.

Reference Documents / Degree of Availability

No reports available.

Classification: 2.3	
Title 1 (Original Language): Kernschmelzen: Untersuchung der Wechselwirkung zwischen Kernschmelze und Reaktorbeton (RS 154 - I.1.5, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** KWU, Erlangen
Title 2 (English): Core Melting: Investigation of the Interaction Between Molten Core and Concrete	**Project Leader:** Dr. Peehs
Initiated (Date): 1. 2. 75 **Status:** Continuing **Completed (Date):** 31. 12. 77 **Last Updating (Date):** 31. 12. 76	

1. General Aim and 2. Particular Objectives

In continuation of the investigations of the interaction between molten core and concrete the following questions will have to be answered:

- How does typical reactor concrete behave on different temperatures up to its melting point?
- Which metallurgical processes occur when "Corium" contacts concrete?
- What happens with the concrete structure, when a representative mass of corium (kg-range) attacks the concrete under accident conditions?

3. Research Program

The research program is devided into the following subtasks:

- Compilation of literature on high temperature behaviour of concrete
- Investigation of concrete up to its liquefaction
- Determination of degassing characteristics of concrete
- Thermal shock behaviour of concrete

- Corium concrete interaction
- Theoretical evaluation of the experimental results

4. Experimental Facilities

Within this task 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

Concrete from different sites and the specimens of the isochron heat treating experiments between room temperature and 1800 °C were analysed, Microsections were prepared. Other samples were investigated by dilatometry, by differential thermoanalysis and thermogravimetry.

The interaction of concrete with "Corium A + R" was studied by melting tests with arc-heating under oxidizing atmosphere.

Corium/concrete interaction tests were executed with molten masses in the kg-range. The tests include "Corium A + R" and "Corium E + R" in concrete crucibles without cooling. The interaction tests were also continued with Fe, Cr, Ni, Zr and steel.

During core melting in concrete crucibles metallic and ceramic products occur. The liquidus temperatures of these products were determined.

The gases which were set free after interaction of corium with concrete were determined.
The thermoshock-behaviour of concrete was tested with a plasma torch.

6. Results

The analysis of the concrete from different sites showed that SiO_2 was the main aggregate component. The CO_2-content was negligable

"Corium A + R" gave a two-phase melt after interaction with concrete.

The Corium/Concrete interaction tests showed that the destruction of the concrete was not caused by chemical interaction with Corium but mainly after reaching the melt temperature of concrete.
The result was confirmed by the interaction tests with Fe, Cr, Ni, Zr and steel.

During the destruction of concrete with basalt aggregates nearly only H_2 was set free; using limestone aggregates CO_2 and H_2 were the main gaseous reactants.

The evaluation of the thermoshock-behaviour showed that the concrete was destroyed about 6 mm/min at a thermal load of 140 W/cm^2. The specific volume destruction is $\sim$ 260 cm^3/kWh.

The liquidus-temperature of the concrete + core melt products varies from

1400 - 1700 °C (metallic phase)
1450 - 2000 °C (ceramic phase)

7. Next Steps
The experimental results will be evaluated. The interaction of corium with carbonatic concrete will be investigated systematically.

8. Relation with Other Projects
RS 183: Energy Balances after Hypothetical RPV Failure with Respect to Concrete Destruction
PNS 4244: Constitution and Reaction Behaviour of LWR-Materials during Core Meltdown

9. References
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10. Degree of Availability

Classification: 2.3	
Title 1 (Original Language): Energiebilanzen nach hypothetischem RDB-Versagen unter Berücksichtigung der Betonzerstörung (RS 183 - I.1.5, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU Erlangen
Title 2 (English): Energy Balances after Hypothetical RPV Failure with Respect to Concrete Destruction	Project Leader: H. Goetzmann H. Hassmann
Initiated (Date): 1. 9. 75 Status: Continuing Completed (Date): 31. 5. 77 Last Updating (Date): 31. 12. 76	

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.

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3. Research Program

3.1 Problems related theoretical investigations:

- Study of the existing knowledge of the available destruction models
- Definition and formulation of the heat transport model

3.2 Energy balance 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

A onedimensional heat conduction model, which calculates the heat up of the containment concrete foundation by the heat from the surface of the molten core, has been tested. A simple convection model was developed with reference to the results of the projects RS 48/1 and RS 166 of the T.U. Hannover. This model was programmed and introduced into the code KAVERN I. Some test calculations were carried out.

6. Results

The experiments of the project RS 154 have shown that during the concrete heat-up first melting occurs at 1300°C. The computer code KAVERN I allows the calculation of the concrete penetration after a hypothetical core melting accident. The test calculations showed that a 7 m concrete wall was molten in vertical direction after 5 days, taking into account the decay heat and the heat capacity of the melt (initial temperature 2400°C) as heat sources. The start time for the concrete penetration was assumed to be 130 min., the specific heat of the concrete was assumed to be equal to SiO_2.

7. Next Steps

The calculations of the concrete penetration will be continued. A new, more detailed onedimensional heat model will be programmed (KAVERN II).

The subroutine MELSIM will be incorporated into the program BILANZ.

8. Relation with Other Projects

RS 154	Core melting: Investigation of the interaction between molten core and concrete
RS 166	Further development of thermohydraulics on a molten CORIUM-pool

9. References

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10. Degree of Availability

Classification: 2.3	
Title 1 (Original Language): Konstitution und Reaktionsverhalten von LWR-Materialien beim Coreschmelzen (PNS 4244 - I.1.5, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: ORGANIZATION: GfK/PNS, Karlsruhe
Title 2 (English): Constitution and reaction behaviour of LWR materials at core melting conditions	**Project Leader:** H.Holleck A.Skokan
Initiated (Date): 1974 **Status:** Continuing **Completed (Date):** **Last Updating (Date):** December 1976	

General aim

Theoretical and experimental investigations of the chemical interactions between core materials, fission products and concrete.

Project status

Progress to date

Reaction behaviour of fission products with Corium under inert conditions.

Reaction behaviour of inert and partly oxidized Corium with different amounts of concrete (silicate aggregates).

Investigations of fuel rods molten down in oxygen and air.

Essential results

Under inert or slightly oxidizing conditions the rare earth fission products are homogeneously dissolved in the oxide phase whereas the alkaline earth metals are enriched in the oxide phase of the melt on places with high Zr-concentrations. The amount of Zr in the oxide and metallic phase is strongly dependent on the oxidation state. The platinum metals and Mo are dissolved in the metallic melt.

An increasing degree of oxidation causes Zr, Nb and later Mo, Tc to enter the oxide melt. The oxidation of these latter elements will occur only when the pressure vessel is molten through and the core melt is in contact with concrete. Concrete melts at about 1300°C. H_2O and CO_2 (limestone aggregates) are released from the heated concrete. They both can oxidize the metallic part of the core melt. The complete miscibility of concrete and $(U,Zr)O_2$ in the liquid state leads to a continuous dilution of the core melt connected with a decrease of the melting point. The metallic part of a core melt corresponding to the composition of partly oxidized Corium A+R is completly oxidized by approximatly twice the amount of concrete (silicate aggregates); the metallic melt of partly oxidized Corium E+R needs about fourfold amounts of concrete to be completly oxidized.

Next steps

Reaction behaviour of fission products with increasing oxydation.

Reactions of the core melt and fission products with concrete (silicate and limestone aggregates).

Investigation of molten fuel bundles.

Relation with other projects

PNS 4243: Experiments on Determination and Limitation of Fission- and Activation Product Release During Core Meltdown

RS 154 : Investigation of the Interaction Between Molten Core and Concrete

Reference documents

Report KFK 2242 (1976) in german
Report KFK 2262 (1976) in german
Report KFK 2399 (1977) in german, in print.

Degree of availability

Unrestucted Distributation

Classification: 2.3	
Title 1 (Original Language): Analyse der Zwischenphase Kernversagen-Schmelzsee und Integration von MELSIM in BILANZ zur Berechnung der Energiebilanzen nach hypothetischem RDB-Versagen (RS 211 - I.1.5, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Inst.f.Kernenerg. Univ.Stuttgart
Title 2 (English): Investigation of the Phase Between Failure of the Core and the Assembling of Molten Material in the Pressure Vessel; Integration of the Program MELSIM into BILANZ in order to Calculate the Energy Balances After a Hypoth. Failure of the Pressure Vessel	Project Leader: Prof.Dr.-Ing. H.Unger, Dipl.-Phys. H.Körber
Initiated (Date): July 1, 1976 Status: Continuing	Completed (Date): March 31, 1978 Last Updating (Date): December 31, 1976

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.

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 until the core falls into the lower plenum of the pressure vessel.
- Integration of MELSIM 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 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

3.2. Integration of MELSIM in BILANZ I and II
   - Coupling of MELSIM and BILANZ
   - Heatup and slumping of the remaining core after a first partial 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 I and II

3.5. Simulation of the accident in applicating the complete model to a standard PWR- or BWR-type and evaluation of the results.

4. Experimental Facilities, Computer Codes

Within this 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. Developing of the PWR and BWR data has been started in order to calculate the first phase of the accident sequence with the computer code MELSIM.

to 3.2. The integration of MELSIM in BILANZ is carried out at present.

to 3.3. The investigation and analysis of the accident sequence has been started.

6. Results

The analysis of the accident sequence gave information on the expected time delay due to evaporation of water remained in the lower plenum of the vessel before the formation of a pool of molten material. Small particles may fall through the holes of the fuel element endplates before softening of endplate material. These particles cannot cause any damage to the reactor vessel as long as the lower plenum is filled with water.

7. Next Steps

to 3.1. Development of the data and performance of the calculations will be completed.

to 3.2. The integration of MELSIM in BILANZ will be carried out.

to 3.3. The computer code LÜCKE will be developed. Calculations with this code will be started.

to 3.4. The integration of LÜCKE into the program system BILANZ - MELSIM will be performed.

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).

9. References

10. Degree of Availability of the Reports

Classification: 2.3	
Title 1 (Original Language): Ingenieurstudie zur H_2-Entwicklung aus der mit Beton wechselwirkenden Kernschmelze (RS 237 - I.1.5, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Engineering Study of the H_2-Evolution from the Interaction of Molten Core with Concrete	Project Leader: H. Goetzmann, H. Hassmann
Initiated (Date): 1. 10. 76 Completed (Date): 30. 4. 77 Status: Continuing Last Updating (Date): 31. 12. 76	

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 into the containment atmosphere after penetrating the melt. 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 therfore comprises analyses of the pressure and temperature build-up in the containment, conside-ring 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

Performance of 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

The PWR initial conditions as required for the maximum estimate of the H_2-releas as well as their influence on the containment pressure have been determined. The initial melt conditions was determined by GfK, while KWU is evaluating the failure mechanism, the concrete composition and the cavern formation in the concrete foundation.

6. Results

The max. estimate is based on the concrete cavern which was determined within the scope of R & D task RS 183. In the case of a first direct contact of the core melt with the concrete foundation after 7000 sec., destruction of the concrete base in vertical direction will last approx. 5 days.

7. Next Steps

Pressure and temperature history in the containment will be calculated for max. H_2-release. An estimate of the influence to be expected with H_2/H_2O-release on the integrity of the PWR containment will begin.

8. Relation with Other Projects

Preparatory work has already been performed within the scope of R & D tasks:

theoretical work: RS 72 a, b, c, RS 73, RS 74 b, RS 183, PNS 4242

experimental work: RS 74 a, RS 154

9. References

PNS-Arbeitsbericht Nr. 17/73:
Zum Problem der Wasserstofffreisetzung beim Versagen der Notkühlung eines wasserstoffgekühlten Reaktors, Nov. 1973

10. Degree of Availability

Unrestricted distribution

Classification: 2.3	
Title 1 (Original Language): Integrale Abschätzung des Einflusses von Ungenauigkeiten (RS 212 - I.1.5, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** Inst.f.Kernenerg. Univ. Stuttgart
Title 2 (English): Integral Estimate of the Influence of Data Inaccuracies: INTAB	**Project Leader:** Prof.Dr.-Ing. H.Unger Dr.-Ing.F.Schmidt
Initiated (Date): May 1, 1976 **Status:** Completed **Completed (Date):** December 31, 1976 **Last Updating (Date):** December 31, 1976	

1. General Aim

Calculations within the area of reactor safety are frequently complicated because of inaccurate knowledge of material properties, sequences of events and related reactor parameters. For these data often only widely spread experimental data or theoretically obtained estimates are available. However, confidence ranges can be defined very often. Within this project, methods were developed in order to estimate the influence of such uncertainties.

2. Particular Objectives

The complexity of the physical events as well as the uncertainties in their description and in the data used are typical conditions under which stochastic (Monte Carlo -) methods may be applied. They are generally more expensive than conventional methods. However, for the estimate of integral data of complicated systems they can be applied such that they cause considerably lower costs than the linear error propagation method for instance.

3. Research Program

The research program is subdivided into four major steps:

3.1. Model development

3.1.1. Generation of independent parameters

3.1.2. Generation of dependent parameters
3.1.3. Evaluation

3.2. Program development
3.2.1. Definition of the interfaces
3.2.2. Integration of the statistic programs

3.3. Adjustment of the program RELAP /1/
3.3.1. Classification of the statistic data and determination of their dependences
3.3.2. Providing the statistical data and issue of the parameters to be analysed

3.4. Calculations and evaluations
3.4.1. Test calculations
3.4.2. Production calculations
3.4.3. Evaluation and discussion of the calculations

4. Experimental Facilities, Computer Codes

to 3.3. The investigations are based on the computer code RELAP which is available at the Institut für Kernenergetik (IKE) and constantly used.
Supplementary work in order to guarantee the automatic recep- [illegible] the statistically obtained sets of input data had to be carried out.

to 3.4. Within the frame of RSYST /2/, modules for the generation and evaluation of statistic data had to be developed. Partially, already existing programs can be used.

5. Progress to Date

The research program has been completed. The final report is in preparation.

6. Results

According to the general aim, methods for the estimation of the influence of statistically distributed errors in the input data of complex computer program systems on the spread of the results of the calculations have been developed. These methods have been integrated into the program system RSYST /2/ and successfully applied to different reactor safety relevant problems. Within the present project the

methods have been applied to blowdown calculations with RELAP.
Results were for instance:

- The spread of the output data is often relatively high compared to that of the input data, e.g. blowdown-times between 18 s and 28 s (average 22 s, spread 15 %, input data spread appr. 5 %).
  A sample problem, based on average input data showed results which coincided partially very good (mass flow rates, e.g.) and partially only fairly well (pressure, fuel temperatures) with the average results of all calculations. Some data (e.g. clad temperatures in certain control volumina) could be found over wide time ranges on the edge of the confidence area.

- With respect to time the results for equal output parameters obtained by means of variation of the input data showed a similar behavior for a few seconds. In a few cases a strong deviation from this behavior occured.

7. <u>Next Steps</u>
The project has been completed.

8. <u>Relation with Other Projects</u>

9. <u>References</u>

/1/ Rettig, W.H. et al.
RELAP3 - A Computer Program for Reactor Blowdown Analysis.
USAEC-Bericht Nr. IN-1321, Juni 1970

/2/ Rühle, R.
RSYST, ein integriertes Modulsystem mit Datenbasis zur automatischen Berechnung von Kernreaktoren. Dissertation, Universität Stuttgart, 1973

10. <u>Degree of Availability of the Reports</u>
/1/ IDAHO NUCLEAR CORPORATION, National Reactor Testing Station, Idaho Falls, Idaho, USA

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 UO_2 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

121-11-10/4170-10

Classification : 3

Titre	**Pays :** FRANCE
Identification et étude des agressions dues aux actions humaines extérieures aux installations	**Organisme directeur :** CEA/DSN
Titre (anglais) Identification and studies of risks on nuclear in relation of outside human activities.	**Organisme exécuteur :** CEA/DSN - SESSN.
	Responsable : R.GERARD (SESSN)
Date de démarrage :01/01/72 Date prévue d'achèvement :31/12/81 **Etat actuel :** Etude en cours Dernière mise à jour : 19/11/76	**Scientifiques :** JP. MADOZ M. PICOL

Objectif général :

Identification des phénomènes extrêmes liés à l'environnement industriel et aux voies de communication, susceptibles d'affecter la sûreté d'une installation nucléaire. Mise en évidence des évènements de référence correspondants.

Objectifs particuliers :

Etablissement et mise à jour permanente d'un dossier d'information concernant les réseaux de transports de fluides et les voies de communication.
Identification et définition des évènements accidentels (avec leurs probabilités associées) liés à l'environnement industriel et aux voies de communication.
Prévision quantitative des transports de nappes dérivantes toxiques ou explosibles dans l'air.

Etat de l'étude :

1) Avancement à ce jour :

La constitution de dossiers d'information concernant : les gazoducs, les pipe lines, les stockages pétroliers, les voies aériennes, est commencée.

2) Résultats essentiels :

Les dossiers d'informations ont déjà été utilisés pour préciser ou compléter les indications données dans les rapports de sûreté. Un code de calcul décrivant l'évolution des concentrations maximales ou fonction de la distance pour une émission volumique instantanée de polluant dans l'air, a été établi.

.../...

Prochaines étapes :

Identification et définition de séquences types d'accident liées à l'environnement industriel. Création d'un code de calcul décrivant l'évaporation. Constitution de dossiers concernant les voies fluviales, maritimes et ferrées.

Documents de référence :

"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.
"Usine de séparation isotopique - étude des conséquences d'un rejet accidentel d'UF 6", R.GERARD, M.PICOL. - Rapport SESR 75/352, 26/6/1975.

120-11-10/4170-10

Classification : 3 543

Titre	Pays :
Identification et étude des phénomènes naturels extrêmes.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais)	Organisme exécuteur : CEA/DSN-SESSN
Identification and study of extreme natural conditions.	Responsable : R. GERARD (SESSN)
Date de démarrage : 01/01/72 Date prévue d'achèvement : 31/12/81 Etat actuel : en cours Dernière mise à jour : 02/1/77	Scientifiques :

Objectif général :

Etude de l'impact de l'environnement naturel et plus particulièrement des phénomènes extrêmes (tornade, cyclones, inondations, raz de marée, glissement de terrain, etc...).sur les installations.

Objectifs particuliers :

Acquisition de données statistiques sur les phénomènes naturels extrêmes.
Identification des phénomènes susceptibles d'affecter la sûreté d'une installation nucléaire.
Mise en évidence des évènements de référence correspondants.

Title 1 (Original language)	Classification
Statistical analysis of randome signals	I - 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

<table>
<tr><td colspan="2">Classification: 3.1</td></tr>
<tr><td>Title 1 (Original Language):
Seismische Kriterien zur Standortauswahl kerntechnischer Anlagen in der BRD
(RS 170 - I.3, Jahresbericht A 76)</td><td>COUNTRY:
BRD
SPONSOR:
BMFT
ORGANIZATION:
BGR</td></tr>
<tr><td>Title 2 (English):
Seismic Risk Maps For Nuclear Power Plants In The Federal Republic Of Germany</td><td rowspan="2">Project Leader:
Dr.H.-P.Harjes</td></tr>
<tr><td>Initiated (Date): 1.1.1976
Status: Continuing
Completed (Date): 31.12.1977
Last Updating (Date): December 1976</td></tr>
</table>

1. General Aim

The project undertakes a seismic regionalization of the Federal Republic of Germany. For that purpose the known earthquake data (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

design strategy for potential hazards will always include statistical elements. Therefore the population of the data ensemble is very important, it should be as complete as possible. The seismic maps of the F.R.G., which were used in the past, only contain strong earthquakes (I $\geqslant$7) of a short time interval (about 200 years).
As a result less than 100 events form a very poor population for probabilistic models. On the other side it was estimated from historical documents that about 1000 earthquakes were felt on the territory of the F.R.G. Of course it is often difficult to evaluate historical catalogues and extract a reasonable figure for the intensity or even for the epicentre of those earthquakes. Nevertheless this elaborate procedure is the only way to get a larger basis as a foundation for all further computations.

3. Research Program

The purpose of seismic risk analysis is to establish a relationship between ground motion parameters like horizontal acceleration and a probability of occurrence for a specified time period at a particular site. This kind of analysis yields a probability distribution for an earthquake intensity, for example, at a site and not a final design value. The latter has to be determined on the basis of an acceptable probability on which the regulatory authorities have to decide.

Besides these statistical aspects research has to concentrate on the relationship between seismological parameters and engineering parameters.

The problem is that the seismologically determined measures for the size or energy-release of an earthquake - e.g. magnitude or intensity - cannot be used directly by engineers. On the other hand, peak values are well defined in engineering practice, but often only weakly dependent on referenced seismological parameters. Although earthquake intensity is a qualitative and subjective measure, it probably is best related to the destructiveness of an earthquake. It also is the only information about historical events and therefore must play a central role in seismic risk evaluations because there are instrumental recordings only since the beginning of this century. The question has to be studied if peak acceleration is a characteristic value for the intensity of an earthquake or whether other ground motion parameters like peak-particle velocity and the duration of shaking should better be taken.

4. Progress To Date

A scheme has been established by which historical descriptions of earthquakes can be translated into relevant seismic parameters like intensity. Thereby the data file is continuously growing. On the other hand these data and also the instrumental data of the last decade have been stored in computer-readable format. Besides the geographical map of the F.R.G. in UTM-projection has been included into the computer file and different plots of epicentre maps have been performed. Because these maps are compiled from inhomogeneous data, they exactly show the problem at the beginning of seismic risk regionalization. In any case, the basic maps should present the original data interpreted as little as possible. A program is in preparation which plots the corresponding isolines of intensity for different limit values.

5. Further Steps

A probabilistic model will be applied to the historical data file. The territory of the F.R.G. will be devided into seismotectonic provinces which will be overlayed by a cartesian grid. Firstly the probability is computed that an earthquake of particular intensity is derived from the Gutenberg-Richter relationship between intensities and their frequency of occurrence. To get a reliable figure for large earthquakes which might be rare in the historical data file it is important to have a clear picture for small intensities. Therefore as many events as possible are included into the file despite the fact they are not of destructive size. The division of the map into seismotectonic provinces yields another estimate of the largest earthquakes this area is able to produce.
In a second step the probability distribution of the earthquake epicentre intensities will be extended to areas outside the seismic provinces. For this reason an attenuation law will be established which reflects the influence of the transfer characteristics of the earth's crust on the intensity at a specified remote site.

References:

Quarterly Reports - IRS-F
5000 KÖLN 1 , Glockengasse 2

<table>
<tr><td>Titre

Carte sismotectonique de la France.</td><td>Pays :
FRANCE<hr>Organisme directeur
SCSIN-CEA/DSN-EdF-BRGM</td></tr>
<tr><td>Titre (anglais)

The seismotectonic map of France.</td><td>Organisme exécuteur
BRGM-CEA/LDG<hr>Responsable :
Comité présidé par M. GOGUEL</td></tr>
<tr><td>Date de démarrage : 01/01/76 Date prévue d'achèvement :31/12/78
Etat actuel : En cours Dernière mise à jour :15/11/76</td><td>Scientifiques :
MM VOGT (BRGM)
MASSIRON (BRGM)
BARBREAU (CEA/DSN)</td></tr>
</table>

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.

Objectifs particuliers :

Permettre la meilleure évaluation possible du niveau de sismicité sur chaque site (localisation des épicentres, détermination de la profondeur des foyers, de la magnitude, de l'intensité, etc..)

Installations expérimentales et programme :

Réseau de détection sismique du LDG. (Laboratoire de Détection Géophysique).

Etat de l'étude :

1) Avancement à ce jour :

En cours

2) Résultats essentiels :

Données importantes déjà rassemblées en ce qui concerne le Sud-Est de la France.

Prochaines étapes :

Poursuite du travail en vue de la synthèse finale.

.../...

120-21-20/4172-10

Classification : 3-1

Titre	Pays : FRANCE
Méthodologie pour le calcul des spectres des séismes de référence des sites à partir des paramètres physiques.	Organisme directeur : CEA/DSN
Titre (anglais) Methodology for calculation of reference earthquake spectra of vibratory ground motion for sites using physical parameters.	Organisme exécuteur : CEA/DSN-SETSSR Responsable : A. BARBREAU (SETSSR-BERSSIN)
Date de démarrage : 01/01/76 Date prévue d'achèvement : 31/12/79 Etat actuel : en cours Dernière mise à jour : 15/11/76	Scientifiques : B. MOHAMMADIOUN G. MOHAMMADIOUN

Objectif général :

Evaluation du risque sismique sur les sites des installations nucléaires.

Objectifs particuliers :

Prévision des spectres de référence du site en fonction des paramètres physiques (magnitude, distance focale, loi d'atténuation des ondes). Ces spectres sont nécessaires pour les calculs de dimensionnement des installations nucléaires (analyse modale).

Installations expérimentales et programme :

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.

Etat de l'étude :

1) Avancement à ce jour :

Enregistrement des répliques du séisme d'Oleron du 7.9.1972 en plusieurs points, d'un séisme dans le Tricastin (10.5.1974), d'une campagne de tirs dans le Tricastin (été 1975), des séismes du Frioul.

2) Résultats essentiels :

Les spectres synthétiques calculés à partir de la magnitude, de la distance focale et des coefficients d'atténuation de l'énergie en fonction de la distance ont été comparés avec les spectres des enregistrements réels et seront adaptés en conséquence.

.../...

Prochaines étapes :

Amélioration des méthodes de modélisation des spectres par introduction de nouvelles données dans la détermination des coefficients utilisés. Adaptation aux séismes français. Calcul des coefficients régionaux d'atténuation de l'énergie en fonction de la distance.

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.

Documents de référence :

"Les études sismologiques effectuées au CEA dans le domaine de la sûreté des sites nucléaires", A.BARBREAU, H.FERRIEUX, B.MOHAMMADIOUN. A.I.E.A. Vienne 1975.
"Protection des centrales vis à vis des séismes", A.BARBREAU, B.MOHAMMADIOUN, H.FERRIEUX - Rapport DSN 50.
"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, 1-3/12/75.

120-21-30/4172-10

Classification : 3-1

555

Titre	Pays :
Suivi expérimental de la sismicité des sites nucléaires et de l'activité des failles.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Experimental continuous work of the seismicity of nuclear sites and fault activity.	Organisme exécuteur : CEA/DSN - SETSSR
	Responsable : A. BARBREAU (SETSSR - BERSSIN)
Date de démarrage : 01/01/76 Date prévue d'achèvement :31/12/81 Etat actuel : Etude en cours Dernière mise à jour : 15/11/76	Scientifiques : H.FERRIEUX G.MOHAMMADIOUN

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.

Objectifs particuliers :

Etude de l'activité des failles dans une région donnée par la surveillance de l'activité sismique (Cadarache, Pierrelatte).
Détermination des caractéristiques spécifiques du site en ce qui concerne la transmission des ondes sismiques (enregistrement sur la roche dure, sur les alluvions, etc.) et prévision des signaux à prendre en compte pour les calculs de dimensionnement.

Installations expérimentales et programme :

Observatoire de Cadarache qui est équipé de différents types d'appareils de mesure (accéléromètre, capteur de vitesse, capteur de déplacement) et qui dispose d'une dynamique complète en amplitude.
Réseau de surveillance de Pierrelatte, qui comprend quatre stations de mesure dont une comportant des capteurs au fond d'un forage de 80 m.

.../...

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 :

Mise en évidence des activités des failles de Jouques et de Beaumont de Pertuis près de Cadarache.
Enregistrement d'un seisme dans la région du Tricastin. Cet enregistrement à servi pour le calcul des spectres DSN proposés pour le site du Tricastin.

Prochaines étapes :

Poursuite de la surveillance des deux sites précités. Création d'une station de surveillance en Alsace.

Relation avec d'autres études :

Carte sismotectonique de la France. Prévision des spectres de référence.

Documents de référence :

Bulletin Mensuel de l'Observatoire de Cadarache.

Titre	Pays : FRANCE
Rassemblement de mesures sur les mouvements en zone épicentrale et d'informations sur les dégâts correspondants.	Organisme directeur : CEA/DSN
Titre (anglais) Connection of measurements on the motions in epicentral zone and informations on the corresponding damages.	Organisme exécuteur : CEA/DSN-SETSSR
	Responsable : A. BARBREAU (SETSSR - BERSSIN)
Date de démarrage : 01/01/76 Date prévue d'achèvement : 31/12/81 Etat actuel : Etude en cours Dernière mise à jour : 15/11/76	Scientifiques : B. MOHAMMADIOUN H. FERRIEUX

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.

Objectifs particuliers :

Etude de la répartition spectrale et du niveau de l'énergie, du niveau d'intensité sismique en corrélation avec les paramètres du séisme (magnitude, distance focale et atténuation), de la relation entre l'intensité macrosismique et les paramètres physiques, accélération, vitesse, déplacement).

Installations expérimentales et programme :

Stations sismologiques légères destinées à l'enregistrement des séismes et de leurs répliques dans la zone épicentrale. Exemples : Etude de répliques du séisme du 7.9.72 (Oléron); Etude des séismes du Frioul. 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.

Etat de l'étude :

1) Avancement à ce jour :

Exploitation d'un code de calcul mis au point à la DAM, en vue de la prévision du mouvement sismique en zone proche. Nombreux enregistrements de séismes dans la région du Frioul. (Italie). Expérimentation effectuée dans le Tricastin (juin 1975).

.../...

2) Résultats essentiels :

Spectres et traces temporelles correspondants obtenus à partir de séismes enregistrés au voisinage de l'épicentre. Comportement de certains terrains du Tricastin soumis à de fortes énergies.

Prochaines étapes :

Etude des comportements des terrains ayant des modules élastiques différents, au moyen de sources sismiques artificielles. Poursuite des enregistrements des séismes dans les zones proches : Installation d'un réseau de stations dans les Alpes du Sud (vallée de l'Ubaye)

Relation 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.

134 - 11 - 11 / 4113 - 01

Classification : 3 - 1

559

Titre	Pays :
Tenue de structures-types sous excitation sismique Essais sur table vibrante	FRANCE
	Organisme directeur CEA/DSN
Titre (anglais) Behaviour of typical structures under seismic excitation. Shake table tests	Organisme exécuteur CEA/DEMT (Saclay)
	Responsable : A. SOKOLOVSKY (DSN) LIVOLANT (DEMT)
Date de démarrage : 1/1/75 Date prévue d'achèvement : 31/12/78 Etat actuel : En cours Dernière mise à jour : 1/1/77	Scientifiques : BERRIAUD (DEMT)

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 structure 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 utilisées en ingénierie.

Objectifs particuliers :

1) Essais de structures traditionnelles
2) Essais d'éléments en béton armé ordinaire
3) Essais d'élements en béton précontraint.

Installations expérimentales et programme :

Table vibrante VESUVE (DEMT)

Etat de l'étude :

1) Avancement à ce jour :

Campagne d'essais exécutée sur murs non contreventés
1) Briques et mortier classés "anciens"
2) Briques et mortier classés "modernes"

2) Résultats essentiels :

Connaissance des fréquences propres et de leur variation au cours de la dégradation du mur. Amortissements en essai de lâcher et variation des amortissements.

.../...

Prochaines étapes :

Essais sur poteaux - types en béton armé
Essais de structures contreventées et chargées.

134-11-12/4113-01

Classification : 3-1

Titre **Analyse parasismique d'une centrale nucléaire interaction sol-fondation.**	Pays : FRANCE
	Organisme directeur : CEA
Titre (anglais) Seismic analysis of a nuclear power plant soil-structure interaction.	Organisme exécuteur : CEA/DEMT
	Responsable : A. SOKOLOVSKY
Date de démarrage : 75 Date prévue d'achèvement : 78 Etat actuel : en cours Dernière mise à jour : 76	Scientifiques : LIVOLANT(DEMT) JEAN PIERRE (DEMT)

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.

Objectifs particuliers :

1) Relation force-déplacement pour un seul radier
2) Application au cas de radiers multiples.

Etat de l'étude :

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.

Prochaines étapes :

Le même programme permettra de la même façon de calculer l'interaction entre deux ou plusieurs radiers voisins de forme quelconque.

.../...

Etat d'avancement de l'étude pour les 6 mois écoulés

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.

Titre	Pays : FRANCE
Etude de séismes synthétiques	Organisme directeur : CEA/DSN
Titre (anglais) Synthetic seismic signal studies	Organisme exécuteur : CEA/DEMT
	Responsable : A. SOKOLOVSKY (DSN/SETSSR) LIVOLANT (DEMT)
Date de démarrage : 1/1/75 Date prévue d'achèvement : 31/12/78 Etat actuel : En cours Dernière mise à jour : 1/1/77	Scientifiques : JEANPIERRE (DEMT)

__Objectif général__ :

Constructions de signaux temporels synthétiques vraisemblables avec des spectres réguliers.

__Objectifs particuliers__ :

1) Analyse détaillée d'enregistrements sismiques disponibles.
2) Mise au point de spectres réguliers.
3) Calculs de validation.

__Etat de l'étude__ :

1) Avancement à ce jour :

Une analyse détaillée des accélérogrammes de séismes réels est en cours ; 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.

__Prochaines étapes__ :

Mise au point de séries de spectres types avec séismes américains (1977).
Validation sur enregistrements de carac[illegible]iques européennes. (1978)

134 - 11 - 13 / 4113 - 01

Classification : 3 - 1

565

Titre	Pays :
Analyse parasismique d'une centrale nucléaire bâtiment réacteur PWR 900. Méthode de calcul	FRANCE
	Organisme directeur CEA/DSN
Titre (anglais) Seismic analysis of a nuclear power plant PWR 900 reactor building	Organisme exécuteur CEA/DEMT (Saclay)
	Responsable : A. SOKOLOVSKY (DSN) LIVOLANT (DEMT)
Date de démarrage : 1/1/75 Date prévue d'achèvement : 31/12/77 Etat actuel : En cours Dernière mise à jour : 1/1/77	Scientifiques : JEANPIERRE (DEMT)

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.

Objectifs particuliers :

1) Modélisation des structures
2) Analyse modale des constituants
3) Méthodologie de prise en compte des coulages.
4) Evaluation des spectres de planchers.

Etat de l'étude :

1) Avancement à ce jour :

ETUDE D'UNE CENTRALE P.W.R.

L'étude du 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).

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.

568

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)	**Country** ITALY
Seismotectonic researches	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) Seismic instruments development and seismic measurements for site evaluation	Country ITALY Sponsor / Organisation } CNEN
Date initiated May I974 (present phase) Date completed In progress Last updating March I977	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.).

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) Studio sulla possibilità di previsione di terremoti con metodi idrogeochimici	Classification 3.1
Title 2 (English) Study on the possibility of predicting earthquakes by hydrogeochemical methods	Country ITALY 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 Sponsor CNEN Organisation CNEN
Studies of site engineering	
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.

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

Classification: 3.2	
Title 1 (Original Language): Untersuchung über das Verhalten von Materialien und Bauteilen des Reaktorbaues gegen aufschlagende Fragmente und Projektile unterschiedlicher Masse und Geschwindigkeit (RS 102-07 - I.3.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Gesellschaft e.V.
Title 2 (English): Behavior of Reactor Specific Materials and Component Parts at the Impact of Fragments, Splinters and Projectiles of different Mass and Velocity	Project Leader: Dipl.-Ing. Hoffmann Ernst-Mach-Institut 78 Freiburg i.Br.
Initiated (Date): 1.9.1973 Status: Finished	Completed (Date): 31.12.1976 Last Updating (Date): December 1976

1. General Aim

The increasing need for nuclear power energy and the actual size of nuclear power plants in twin-blocks of up to 1300 MW, each, forces to investigate and to improve the security means (for people against radioactivity) especially in densely populated regions and near industrial facility concentrations.

Within the scope of the reactor safety program extreme accidents and failures in nuclear power plants are analyzed as to initiation, processing and destruction effects with the aim to improve the actual safety of light water reactors (PWR and BWR) and to optimize the actual safety installations (materials, components, systems).

2. Particular Objectives

In cases of interference and damage from the outside as well as from the inside of nuclear power plants fragments and projectiles may result with masses, ranging from several tons (e.g. from aircraft crashes) down to a few grams (e.g. from failures of components) and attaining velocities from some m/s up to more than 1000 m/s. The impact of such projectiles can endanger reactor components relevant for safety functions and must therefore be considered in the designing of nuclear power plants.

The resistance of building components of reactor plants - especially of steel - and reinforced concrete slabs of different reinforcements, compressive strength and thickness - impacted by projectiles of various masses and velocities were to be investigated in ballistic tests in the impact simulation plant of the EMI.

3. Research Programm

3.1 Reinforced concrete slabs, the usual building components of the outer shield-wall of a typical pressurized water reactor (PWR)-building, were to be impacted with low and medium mass projectiles in order to investigate the resistance behavior of reinforced concrete under short-time, dynamic loads.

3.1.1 The applicability of model-laws for the penetration- and perforation- processes were to be checked in tests with original- and model- targets.

3.1.2 The penetration depths experimentally found were to be compared with the penetration depths calculated with the classical penetration formulae known from literature.

3.2 The influence of the reinforcement of a protection-wall in a nuclear power plant against missiles impacted by heavy mass projectiles from the inside of the reactor was to be investigated.

3.3 The resistance of reactor-specific steel, such as usually built into reactor pressure vessels, against impacting projectiles of different masses and velocities was to be tested.

4. Experimental Facilities, Computer Codes

Re 3.1 through 3.3:

All tests were performed in the impact simulation plant already introduced in Annual Report 1974.

5. Process to Date

Re 3.1:

To complete the first testphase, performed in 1975 and described in Annual Report 1975, further 30 square (700 x 700 mm) reinforced concrete slabs of the grades Bn 250 and Bn 350 were impacted with solid, cylindrical steel projectiles (C 110 W 1) and all results scientifically evaluated.

The following parameter were varied:

Projectile: caliber (D = 15, 20 and 30 mm), length by diameter ratio (L/D = 1 or 2), mass (m = 20,6; 41,2; 49,2; 165 and 340 g), velocity (v = 100 up to 1700 m/s).

Target: compressibility ($ß_{w28}$ = 276 up to 391 kp/cm^2), reinforcement (depending upon model-scale), aggregate (D^+ = 8 - 16 mm), target thickness (t = 200, 300 and 400 mm).

The target-slabs were placed free-hanging in the impact chamber in order to avoid boundary effects on the penetration process of the projectile.

Re 3.1.1:

Besides the full-scale slabs also model-scale slabs (scale 1 : 2), manufactured from commercial materials (aggregates, cement, concrete matting) were impacted.

Re 3.1.2:

The measured penetration depths were compared with the penetration depths by the classic penetration formulae (Petry, BRL, CoE and HN-NDRC).

Re 3.2:

In cooperation with a nuclear power plant manufacturer four typical reinforcements were chosen for the concrete targets to be tested. Solid, cylindrical steel projectiles (material 9 S 20 K) with a diameter of 4,5 cm, L/D = 2 or L/D = 4 resp. and the corresponding masses of 1,11 resp. 2,25 kg were fired at the different, reinforced concrete slabs (70 x 70 x 23 (cm)), model-scale 1 : 3,5. The projectiles of these model-scale tests correspond to full-scale projectiles with 15,75 cm diameter, 31,5 resp. 63 cm length and 47,7 resp. 96,6 kg mass. Penetration depths and crater voluminae were measured and the beginning of scabbing at the back side of theslabs investigated for projectile velocities of 150 up to 220 m/s. The concrete parts, expelled by scabbing, can cause considerable dama ge behind the slabs. Knowledge about the scabbing thickness - the limit slab-thickness where scabbing effects begin to be just visible is therefore of great importance for the designing of nuclear power plants.

For the evaluation of the four different reinforcements the slabs were subjected to extreme dynamic loads at projectile velocities of 410 up to 427 m/s for the L/D = 2 and v = 300 up to 312 m/s for the L/D = 4 projectiles.

Re 3.3:

Reactor steel slabs (22 NiMoCr 3 7) with thicknesses of 11 up to 27 mm were impacted with solid cylindrical steel projectiles (material C 110 W 1, diameter 15 up to 40 mm, L/D = 1, masses 20 up to 390 g) at impact velocities of 366 up to 935 m/s.

At the end of 1976 this research program in its experimental and theoretical studies was essentially finished. The results are discussed now and a final report is prepared. In comparison with the planning three years ago it can be said that there is no mentionable difference between the time calculated and the time needed and between the work that had to be done and that which had been done, furthemore there is no overrunning of the total costs.

6. Results

Re 3.1:

The figures 1 - 4 show penetration depths, normalized by the projectile diameter, as a function of the impact projectiles velocities. For the same projectile geometries L/D = 1 and L/D = 2 resp. are the resulting values within each range of concrete strengths situated each on a curve.
The crater volumes V increase when double-logarithmic plotted $V = f(E_{kin})$, in the range of velocities measured, linear with the kinetic energy E_{kin} of the impacting projectile. Photographs of the craters on front- and back side of the impacted slabs were made and evaluated and also the beginning of scabbing investigated.

Re 3.1.1:

When the most important model criteria (scaling laws) were followed the applicability of model scale laws (Cranz) for impact tests with model slabs of reinforced concrete (scale 1 : 2) was proven. As shown in Fig. 1 - 4, the values of the full-scale ("Korn" 16) and model scale ("Korn" 8) penetration depths in relation to the projectile velocity are situated on one curve.

Re 3.1.2:

The comparison showed, that up to now there does not exist a known penetration formula for reinforced concrete, which permits satisfactory theoretical predictions of penetration depths for a wide range of projectile- and target-parameters.

Re 3.2:

At limit load conditions (army complete) the influence of the different reinforcements could be demonstrated.

Re 3.3:

The limit slab thickness of 22 NiMoCr 3 7 steel was determined, at which the projectile with its pertinent velocity only creates little holes and cracks and at which neither projectile- nor target-material may emerge from the back of the slab.

Summarizing the results of this three year's research program in fact there were done steps forward. In several technical reports (see point 9. "Literature" also in the Annual Report A 75) there was found out which kind of projectiles may emerge a nuclear power plant from inside and outside and how they are to classify. The laws for penetration and perforation of concrete were gathered and discussed, especially their limits of validity are shown. In this sense they were submitted, tested and enlarged by own experiments which respected the modern reinforced concrete as well as specific steel used in nuclear power plants. By that means especially when the final report is printed it should be possible for calculators, civil engineers and all those people who have to plan, to design and to build a nuclear power plant to improve the safety and to optimize the installation materials.

7. Next Steps

The project RS 102 - 7 will terminate with a report on "Impact Studies on Reinforced Concrete Slabs, Part III" and a conclusive final report.

8. Relations with other Projects

Relations with other projects do not exist except for some cooperation with other experimental workers of the institute.

9. References

1) H. Langheim Impactuntersuchungen an armierten Betonplatten, Teil I (Impact Studies on Reinforced Concrete Slabs, Part I), EMI-Report E 9/76

2) H. Langheim Impactuntersuchungen an armierten Betonplatten, Teil II (Impact Studies on Reinforced Concrete Slabs, Part II), EMI-Report E 14/76

10. Degree of Availability of the Reports

1) available with permission of BMFT

2) is being printed

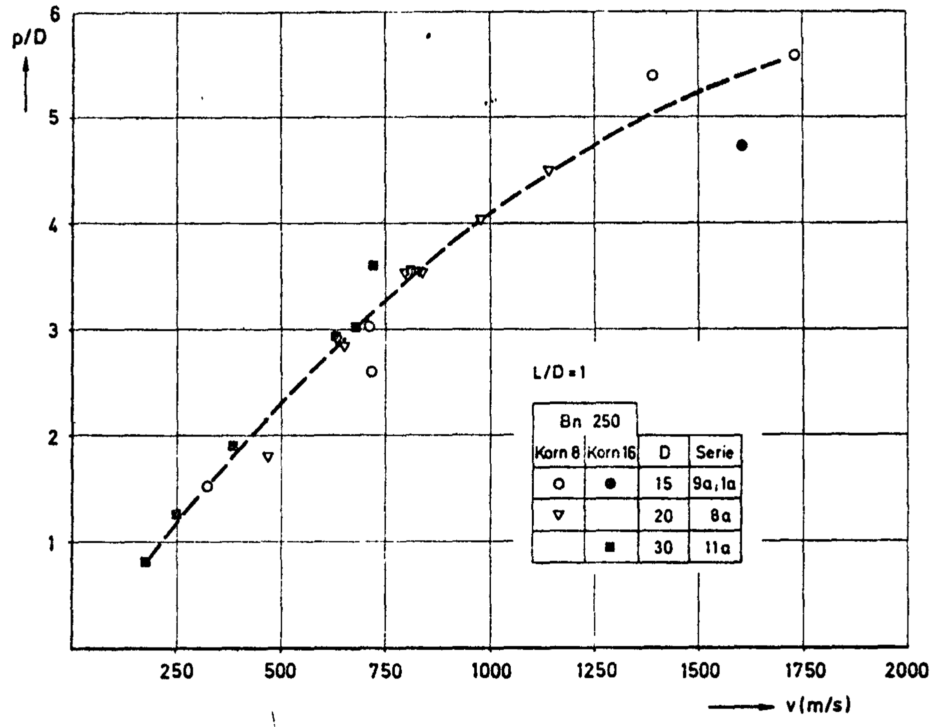

Figure 1: Penetration depths, normalized by the projectile diameter as a function of the projectiles velocities. Projektile geometry L/D = 1, concrete B 250

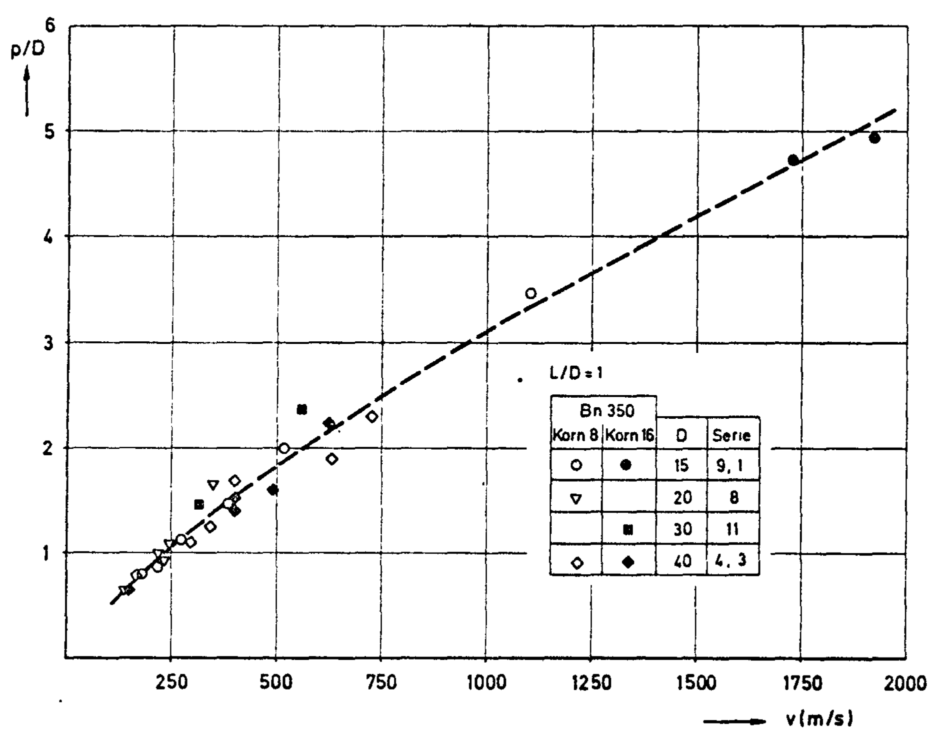

Figure 2: Penetration depths, normalized by the projectile diameter as a function of the projectiles velocities. Projectile geometry L/D = 1, concrete B 350

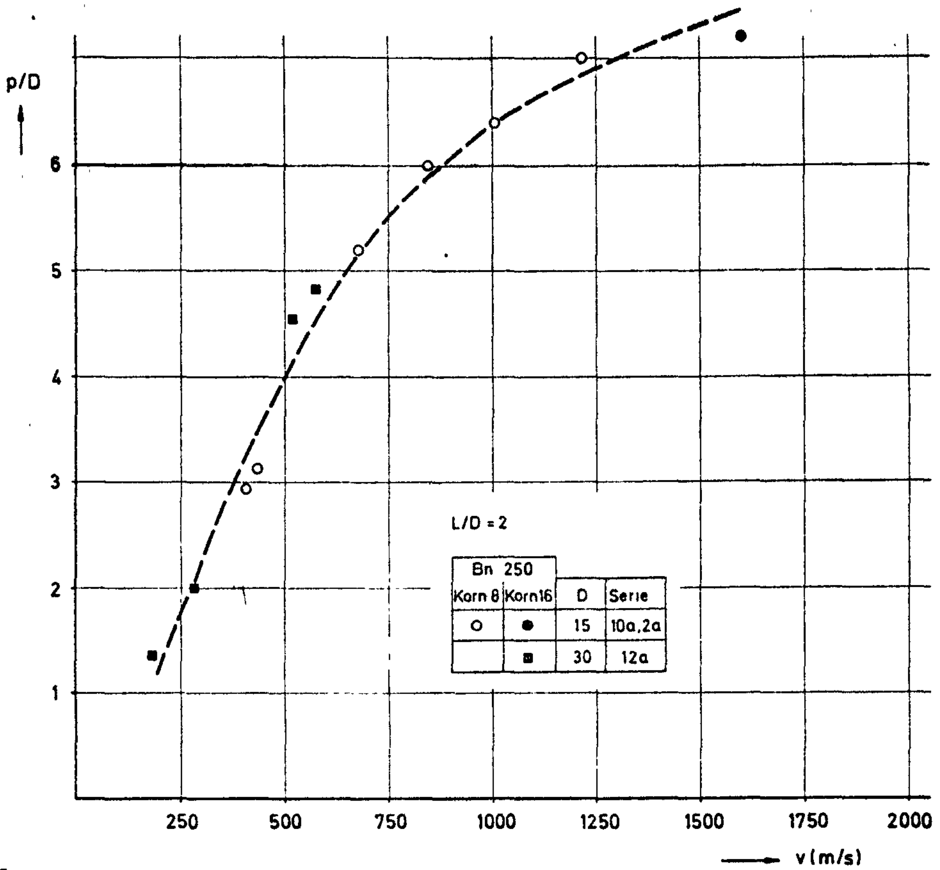

Figure 3: Penetration depths, normalized by the projectile diameter as a function of the projectiles velocities. Projectile geometry L/D = 2, concrete B 250

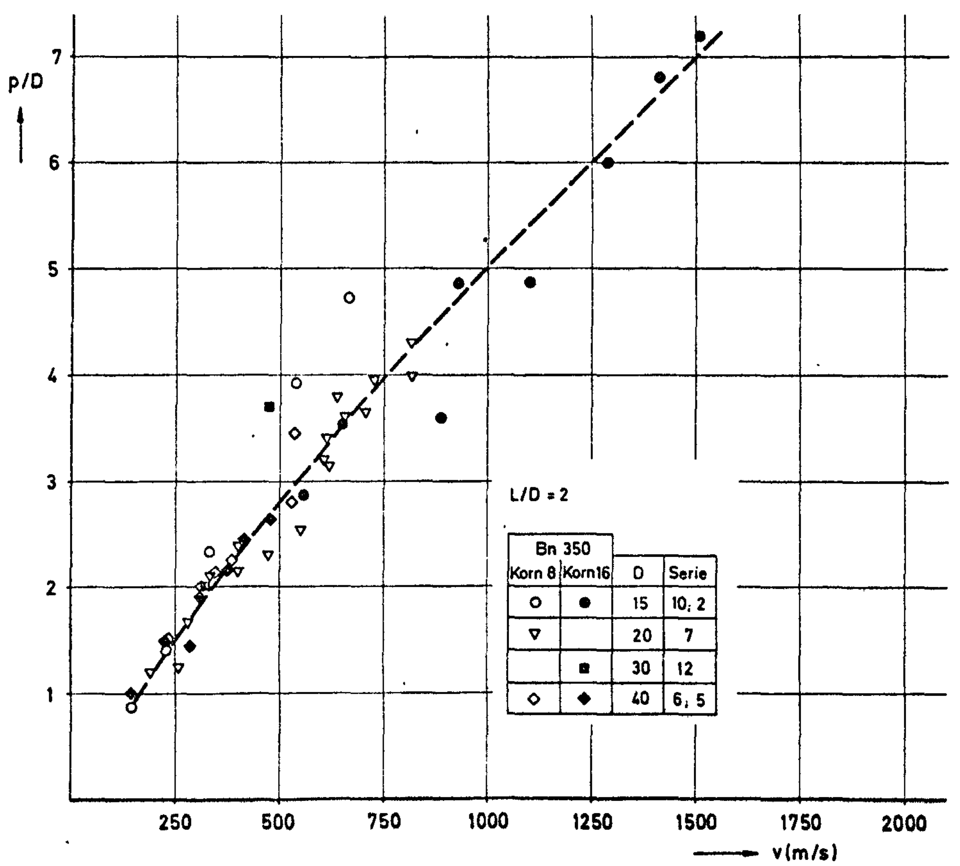

Figure 4: Penetration depths, normalized by the projectile diameter as a function of the projectiles velocities. Projectile geometry L/D = 2, concrete B 350

Classification: 3.2	
Title 1 (Original Language): Grenztragfähigkeit von Stahlbetonplatten bei hohen Belastungsgeschwindigkeiten (z.B. Flugzeugabsturz) und Untersuchung der Widerstandsfähigkeit von Betonstrukturen gegen Flugzeugabsturz (RS 165 und 149 - I.3.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Hochtief AG,Ffm. BWB, Koblenz
Title 2 (English): Ultimate Bearing Capacity of Reinforced Concrete Plates under Time-Dependent Loads (e.g. Aircraft Crash) Investigation of the Resistance of Concrete Structures against Aircraft Crash	Project Leader: Riech (coordination)
Initiated (Date): July 1, 1975 and Oct. 10, 1974 Status: Continuing	Completed (Date): Sept. 30, 1979 June 30, 1979 Last Updating (Date): December 1976

1. General Aim

In connection with research works performed on the field of reactor safety, one of the subjects covering "External Events" gives special emphasis to the investigation of the behaviour of the outer containment of nuclear power plants under aircraft crash loading. For the loading case "Aircraft Crash" it is demanded that no failure of safety components will occur and no radioactive substances may escape. This requirement will be met by an appropriate dimensioning of the outer containment of the structures surrounding the nuclear components so that a crashing aircraft cannot penetrate the outer walls.

Since the loading case "Aircraft Crash" entails high load peak values within short periods, the knowledge of the kinetic ultimate bearing loads is required for a safe and economic design of the plates and shells being used for nuclear power plants, i.e. the best utilization of all safety reserves.

Both combined research projects, RS 165 and RS 149, are dedicated to the theoretical and experimental investigation of the essential problems of the loading case "Aircraft Crash":

1. Investigation of impact load/time characteristics during the impact of deformable missiles;
2. Investigation of the kinetic bearing behaviour of reinforced

concrete plates.

2./3. Particular Objectives and 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 three-dimensional problem in the area of load introduction.

The project RS 149 comprises:

- design, 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.

For the following reasons the scale of the experiments was chosen as large as possible:
Small scale structures adjusted to laboratory conditions would cause difficulties on the following fields:

- selection of the granular size of the aggregates
- measures securing the composite of steel and concrete
- registration of the kinetic stress distribution (shock waves) within the structure (the velocity of the shock waves is independent of the geometric scale)
- installation of stirrups in the reinforcement and evaluation of their effectiveness.

The mentioned difficulties would require special activities and compromise settlements and would entail additional falsifying values or such effects which cannot be taken into account by a realistic theoretical treatment. The interpretation of the results would become very difficult or even impossible, especially with regard to the separation of the essential influence factors.

4. Experimental Facilities

The realization of large scale experiments was facilitated by the co-operation of the Federal Ministry for Research and Technology and the Federal Ministry of Defense.

At the site of the Bundeswehr-Erprobungsstelle 91 (BWB/E.St.91) at Meppen a gas-operated accelerator was built by which missiles of a maximum diameter of 600 mm, a length of up to 7 m and a mass of up to 1000 kg may be accelerated to a maximum speed of 300 m/s.

Specially constructed projectiles are used as model missiles, consisting of concentric tubes of varying length. The deformation behaviour of those tubes yields the impact load/time characteristics similar to those which may be expected in an aircraft crash. Quadratic reinforced concrete walls of about 6 m edge length and up to 0,90 m thickness are used as targets. An abutment has been constructed in order to fix the target plates.

5. Progress to Date

5.1 Experimental Plant

The gas-operated accelerator was completed and tested. The operating safety, the full serviceability and the reproducibility of each shot were proven. The plant has been serviceable since the end of November 1976.

The construction of the target abutment could be terminated in mid-December. The production of missing steelwork elements being indispensable for the performance of the tests as well as the filling of the mound behind the abutment are scheduled to be completed until early March 1977; from this date on, the complete experimental plant will be serviceable.

5.2 Test Equipment for Measuring Series I

Measuring series I of the planned tests serves to check the model idea for the determination of the time-dependent load arising from an impact of deformable missiles onto rigid structures. The series comprises a total of 4 tests. The missile model for the first test was determined. The projectile consists of an approximately 6 m long outer cylinder with a diameter of 600mm, its shell thickness being longitudinally graded from 5 mm in the front area to 10 mm in the rear area, and of a concentrical inner tube of approximately 1 m length and 500 mm diameter, which is connected to the outer cylinder by means of ribs (shell thickness = 7 mm). The front end is closed by a spherical cap, the rear end is formed by a reinforced plane plate.

For the maximum impact load, the reinforced concrete target plates of measuring series I shall display as little bending as possible and shall safely exclude perforation. They mainly serve to distribute the load to the four 10 MN-force transducers recording the impact load/time characteristics. For the determination of the outer dimensions (width/height/thickness = 3,70/3,50/3,00 m), considerations regarding construction and testing procedure were primarily decisive. The total weight of the target structure amounts to approximately 0,8 MN. By means of two chains it is slung to a steel cross-bar placed transversely to the direction of the impact on top of the abutment.

5.3 Measuring Method

For an expert study, the measurement program supplied by Hochtief/KTI in coordination with BWB/E.St. 91 was submissed to the Otto-Graf-Institute (Department for Civil Engineering and Construction of the University Stuttgart). The expertise mainly referred to measurements taken at the reinforced concrete test plates. Proposals for alterations especially dealt with the measurements of the reaction at the support and with measures to check the reaction velocity of the measuring equipment. The testing phase of the accelerator plant was utilized to obtain data on the aerodynamical and deformation characteristics of steel projectiles and on possibilities to record and measure the procedure of the impact and the velocity/time behaviour of the projectiles. In order to avoid a negative influence on the time schedule for the future proceedings of the entire project, the difficulties encountered with measurements and the uncertainty in connection with the evaluation of the deformation behaviour necessitated two supplimentary preliminary tests prior to the start of measuring series I. These tests essentially aim at achieving unobjectionably reliable methods of measurement to record the impact behaviour of the missile models. In addition, the precomputation of the impact load/time behaviour shall be based on at least qualitative test results in such a way that the dimensioning of the projectiles for the first major tests is sufficiently reliable to avoid extreme failures during these tests.

Figures 2 and 3 show parts of the deformed missile model used during the first preliminary test. The missile consisted of a steel tube of approximately 6 m length (Ø 600 mm, shell thickness = 5 mm), the front end of which was closed by a spherical cap and the rear end was formed by a plane plate. By means of photoelectric cells, the impact velocity was determined to be 296,2 m/s. The free trajectory reached approximately 35 m. A remarkable deviation from the theoretical axis of the trajectory could not be noticed.

As a target, a concrete plate was used, which was vertically placed into the mound. Due to the impact, the steel tube was axially compressed. Because of a technical failure of the high speed camera, the process of the impact could unfortunately not be recorded. As information on the deformation behaviour is undispensable during the major tests, supplimentary measuring methods will be tested during a second preliminary test to ensure the achievement of the required data by means of additional measurement methods in case of a renewed failure of the high speed camera. The following measurement procedures are intended to be used for recording the time sequence of the flying phase and the impact of the missile onto the concrete plate:

- acceleration transducer fixed at the rear end plate inside the missile; wireless transmission of the measured values by means of a transmitter being also installed in the interior of the missile (telemetry);

  Aim: Record of the time-dependent acceleration of one point of the missile shell;
- by means of radar, the velocity/time behaviour of a point located at the front part of the projectile has to be recorded until the impact process is initiated;
- determination of the velocity of the missile at one point of the trajectory by means of photoelectric cells;
- high speed camera ("Lexander" camera):
  quantitative evaluation of the deformation behaviour (no possibility of reproducing moved pictures!)

Improved technique for taking pictures by means of supplimentary lighting and by marking the missile with a Scotch-light-foil;
- second high speed camera for reproduction of the motion process.

5.4 Preparing Theoretical Works

- Investigation concerning the consideration of the physically non-linear behaviour when calculating the forces and deformations of re-inforced concrete plates.
- Calculation of plates by means of triangular and rectangular finite elements, with special regard to the accuracy of the results and the quantity of the calculation.
- Determination and testing of absolutely stable numerical integration systems for a solution of the nonlinear differential equation system as they are for example arising when applying the FEM-method for the calculation of plates and shells.

6. Results

First results cannot be expected prior to the completion of the total experimental plant.

7. Next Steps

7.1 In 1977, the entire measuring series I shall be performed and the pertaining documentation of the measurement values and the results of the accompanying theoretical research activities in connection with these tests shall be submitted in a technical report.

7.2 Towards the end of the year, the first tests of measuring series II shall be effected.

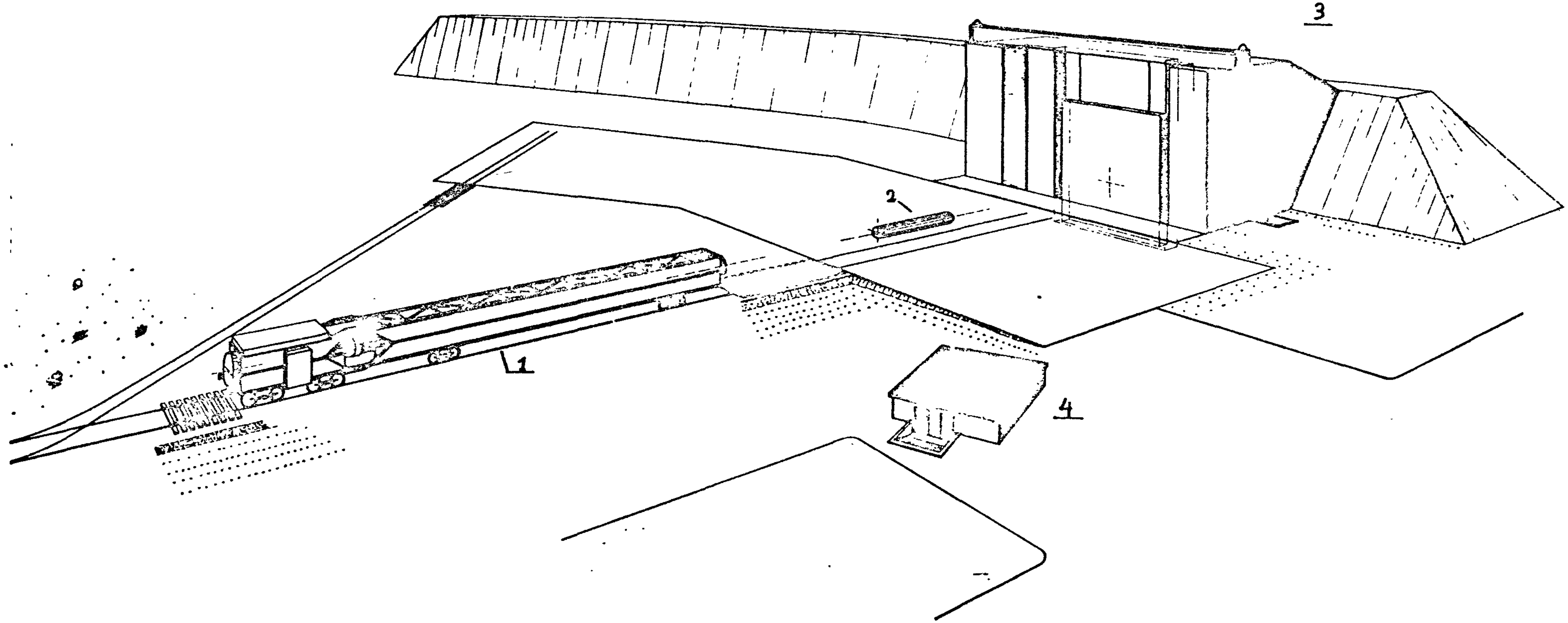

Figure 1: <u>Experimental Plant</u>

(1: accelerator   3: abutment with test plate for measuring series II
 2: projectile   4: measuring dug-out)

Figure 2: Missile model 1
Rear end and part of the deformed tube Ø 600 x 5 mm

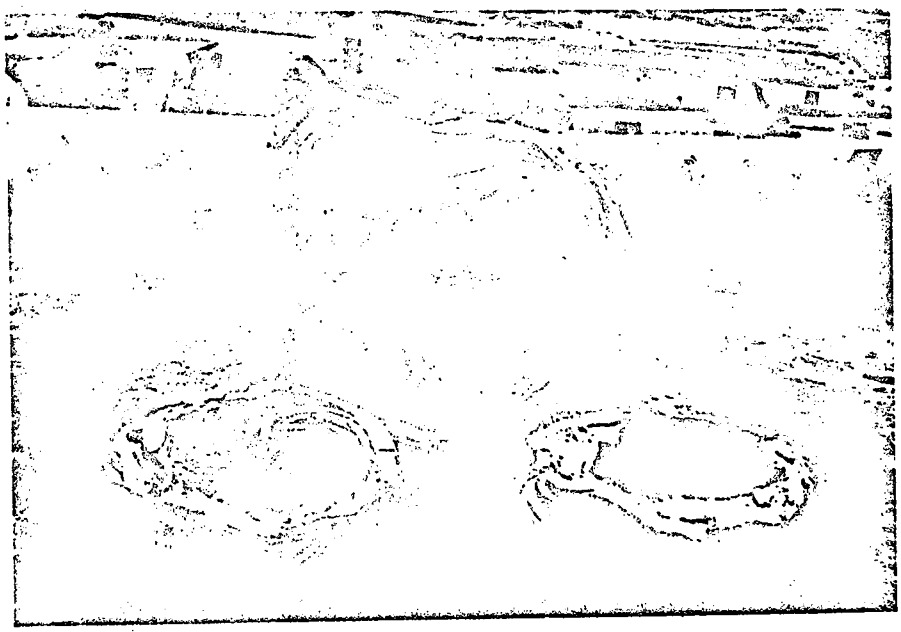

Figure 3: Parts of missile model 1

Classification: 3.2

Title 1 (Original Language):

Geschoßbelastung von armiertem Beton,
Rechenprogramm und Berechnungen
(RS 226 - I.3.2, Jahresbericht A 76)

COUNTRY: BRD

SPONSOR: BMFT

ORGANIZATION: KED

Title 2 (English):

Missile Impact on Reinforced Concrete,
Computer Code and Calculations

Project Leader:

B. Nowotny

Initiated (Date):	Completed (Date):
Sept. 1, 1976	April 30, 1977
Status:	**Last Updating (Date):**
Continuing	December 1976

1. General Aim

Nuclear Power plants have to be designed to resist an airplane crash without endangering the environment. It is a complicated problem to find out the effects of the imact of the airplane against the concrete wall, which has not yet been sufficiently investigated by experiments and theory.

2. Particular Objectives

In this research program a mathematical model for reinforced concrete should be developed and the results be compared with experiments. The aim of this code is to be able to extrapolate experimental results, especially for the effects of an airplane crash against a nuclear power plant.

3. Research Program

3.1 Development of a dynamic anisotropic mathematical model of reinforced concrete

3.2 Comparative calculations with missile impact experiments

4. Experimental Facilities, Computer Codes

Item 3.1 Anisotropic model for concrete

The mathematical model consists of special subroutines for the finite difference code PISCES 2DL /1/, which is usable for everybody at the Control Data Corporation (CDC). The model has cylindrical symmetry corresponding to the two-dimensional PISCES code.

One zone of the model contains concrete and steel. The steel stresses are calculated separately from the concrete and are smeared over the whole zone and superimposed the concrete stresses. The 3 possible reinforcements in the 3 directions are independent of each other and may be different.

There are the following conditions of the concrete

- undestroyed
- radially broken
- conically broken
- completely destroyed

The undestroyed concrete is linear-elastic and isotrop.

The concrete breaks, when the tensile principal stress exceedes the tensile strenght of the concrete, and spalls perpendicular to the principal stress. The gaps can close again. The fragments can slide on each other if the friction force is exceeded.

If the concrete is completely destroyed, its behaviour becomes isotropic again. Then it is described as normally provided for in PISCES: by an equation of state (pressure = f (density)) and by a yield model (yield limit = f (pressure)).

The equation of state considers the following phenomena

- elastic volume compression
- permanent volume decrease
- compact elastic behaviour (no more voids do exist)
- spalling

The yield limit describes the breaking limit of the undestroyed concrete under shear, and the yield condition of the completely destroyed concrete (yield model by Mohr-Coulomb, see /2/ e. g.).

The behaviour of the reinforcement is ideally elastic-plastic under tension. When the concrete is undestroyed, the steel can take compression forces too, when it is destroyed, the steel is assumed to buckle.

The reinforcement steel can slide in the concrete, when the adhering force is exceeded. This adhering force is a function of the condition of the concrete (condition of destruction, pressure).

5. Progress to date

The computer code for the anisotropic concrete model has been developed and tested by simple load cases (see /3/). The work has been performed in time according to the original schedule.

6. Results

The code is working.

7. Next Steps

Calculation of 3 missiles and comparsion with measurements of the Ernst-Mach-Institute.

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8. Relation with other Projects

Qualification of the code to accompany eventually the planned tests in Meppen, Germany (air plane simulating missiles against concrete slabs)

9. References

/1/ PISCES 2Dl, Manual A, B, C, D
Physics International Company
2700 Merced, San Leandro, Calif.

/2/ J. C. Jaeger,
Elasticity, Plasticity and Flow
Methuen & Co Ltd and Science Paperbacks 1974

/3/ Geschoßbelastung von armiertem Beton,
Zwischenbericht: Entwicklung eines anisotropen Materialmodelles für armierten Beton. KED-Bericht Dezember 1976

10. Degree of Availability of the Report

/1/ also available at Control Data Corporation, e. g.
Stresemannallee 30, 6000 Frankfurt/Main 70

/3/ the report is present at IRS

	Classification $\boxed{3.2}$11.4.
Title 1 Flystyrt på containment-bygning	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2 Aircraft impact on containment building.	Project leader Per Lundsager Scientists:
Initiated (date): March 1975 Status: progressing Completed: (date) 1976 Last updating (date) February 1976	Per Lundsager S. Krenk

1. General aim

1.1. Investigations on structural consequences of an aircraft impact on a containment building.

1.2. Comparison of several FEM-codes available to Risø and evaluation of their potentials in this type of analysis.

2. Particular objectives

2.1. To study the transient response of an idealized axisymmetric containment structure subject to an impact force of the linearized Riera type.

2.2. To study the influence on the displacement and stress results of the element types available in a number of FEM-codes.

3. Experimental facilities and programme

None

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4. Project status

4.1. Progress to date

A number of analyses have been carried out using 3 linear and 1 nonlinear code. 3 types of shell elements and 2 types of isoparametric solid elements have been applied.

4.2. Essential results

The stress results seem to a remarkable extend to depend on proper matching of element type and grid layout.

5. Next steps

Evaluation of results.

Internal and external reporting.

6. Relations with other projects

No direct relations.

7. Reference documents

Dynamic Analysis of Aircraft Impact Using the Linear Elastic Finite Element Codes FINEL, SAP and STARDYNE.
Per Lundsager, Steen Krenk.
Risø-M-1817, Aug. 1975.
Presented at the ELCALAP Seminar, Berlin, Sept. 1975.

8. Degree of availability

Project information is freely available

134 - 41 -11 / 4111-06

Titre	Pays :
Tenue des structures aux projectiles	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Behaviour of structures under missile impact	Organisme exécuteur : CEA/DEMT - CEA/CESTA
	Responsable : R.AVET-FLANCARD (DSN) LIVOLANT (DEMT)
Date de démarrage : 74 Date prévue d'achèvement : 78 Etat actuel : En cours Dernière mise à jour : 05/11/76	Scientifiques : BERRIAUD (DEMT) (CESTA)

Objectif général :

Cette étude est destinée à mieux faire connaître la tenue d'une paroi en béton armé sous le choc d'un projectile. Les conditions de résistance limite sont systématiquement recherchées afin de permettre la mise au point d'une formule empirique permettant d'évaluer la tenue à la perforation de différentes structures (NB: le projectile est "dur").

Objectifs particuliers :

1) Calculs d'évaluation préliminaire
2) Recherche de conditions typiques pour maquettage
3) Etude de l'influence de diamètre et de la masse de projectile
4) Etude de l'influence du ferraillage
5) Cas représentatifs de cas réels
6) Influence de différents rapports géométriques
7) Influence des caractéristiques mécaniques des parois
8) Condition extrême (très grande vitesse)
9) Forme de projectile.

Etat de l'étude :

1) Avancement à ce jour :

Ont été réalisés à ce jour outre les calculs d'évaluation:
Les tirs préliminaires portant sur les efforts de masse et de diamètre du projectile,
Les tirs pour apprécier l'influence du taux de ferraillage,
Les expériences représentatives d'aéronefs réels,
Les tirs sur double paroi,
Les tirs relatifs au paramètre diamètre/épaisseur de paroi.
Quelques tirs à très grande vitesse ont été effectués.

.../...

2) Résultats essentiels :

Un ensemble de résultats a permis de situer les vitesses de juste perforation dans une assez large gamme.
Un relevé graphique a permis d'approcher une formule empirique faisant intervenir pour la dalle : l'épaisseur, la densité, la résistance à la compression, et pour le projectile : la masse, le diamètre et la vitesse .

Prochaines étapes :

Etude de l'influence de la disposition des aciers.
Etude de l'influence de la résistance de compression du béton.
Etude de la forme du projectile.

TNO - IWECO	CLASSIFICATION: 3.2/3.3/7.1
TITLE: Responsiebeberekeningen voor reactorgebouw	COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs ORGANIZATION: TNO-IWECO
TITLE: (ENGLISH LANGUAGE): Dynamic response of reactor structures (building and containment)	PROJECTLEADER: Meijers
INITIATED: June 1974 — LAST UPDATING: April 1977	SCIENTISTS: Van Beek Geertsema
STATUS: Progressing — COMPLETED: 1978	

General aim

Development and application of calculational tools for the evaluation of structural response of reactorbuildings under dynamic loading conditions.

Particular objectives

The calculations are directed to the evaluation of the effects of
(1) pressure/blast waves of gasexplosion in the vicinity of the site
(2) impact of striking aircraft on the reactor (containment) buildings.

Experimental facilities and program: None

Project status

A computational method has been developed and implemented in the MARC-computercode to describe the non-linear behaviour of reinforced concrete with a finite element technique. This method has been applied to a typical reactorbuilding to analyse the structural response due to (1) blast wave, (2) impact of large commercial aircraft.

Next steps

Further application of method to aircraft-impact. Development and implementation in the MARC-computercode of a more refined Mohr-Coulomb criterion to improve the numerical description of the inelastic behaviour of concrete. Investigation of the load-carrying capacity of flat platepanels, of reinforced concrete, subjected to local dynamic loads.

Relation to other projects: None

Reference documents

"Mathematical description of the non-linear behaviour of reinforced concete" by Ir. H. Geertsema, June 1976, TNO-IWECO, report no. 11261/2
"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 no. 11261/3.

Degree of availability

Through Ministry of Social Affairs

Budget: Hfl. 306.000,-- (1975 + 1976), approx. Hfl. 72.500,-- (1977 + 1978)

Classification: 3.3	
Title 1 (Original Language): Entstehung chemischer Explosionen und deren Wirkung auf sicherheitstechnisch wichtige Reaktorkomponenten (RS 1o2-o6 - I.3.2, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** ICT/EMI (FhG)
Title 2 (English): Formation of Chemical Explosions and their Effects upon Reactor Components with Important Safety Functions	**Project Leader:** Dr. Pförtner, Dipl.-Ing. Hoffmann
Initiated (Date): 1.9.73 **Status:** finished	**Completed (Date):** 1976 **Last Updating (Date):** December 1976

1. General Aim

Experiments with unconfined ignitable clouds of explosible gases are performed in order to supply information on the hazards to nuclear installations from gas explosions resulting from accidental gas leak or spill in the course of production, storage and transport of explosible gases near nuclear power plants.

2. Particular Objectives

The project will include all the following aspects:

Studies of gas dispersion after a leak or spill of explosible gas from liquid-gas containers or pipes and determination of the amount of ignitable mixture at any time after the spill.

Determination of the propagation functions of pressure and shock waves generated by the explosion of unconfined vapour clouds and assessment of their impact on a nuclear power plant.

3. Research Program

3.1. Evaluation of the available literature and selection of explosible gases which are to be investigated

A comprehensive review of the current status of the subject with numerous references is contained in a paper by Strehlow and Baker (1975).

The gases methane, ethylene and propane will be investigated.

3.2. Studies of leakage or spill phenomena associated with liquified explosible gases and the evaporation, spreading and mixing of these substances in the surrounding air

As parameters, the surrounding conditions (temperature, wind, nature of the soil) and source strength are varied.
A hall of 6 x 12 m^2 ground area will be used for the gas dispersion tests. Liquid gas will be spilled onto a heatable platform placed at one end of the hall. The gas evaporation rate is measured by weighing the platform with a force transducer. The extension of the cold gas cloud is observed by filming the foggy region in scattered light and measured by several traverses with 8 suction probes placed at a plane at varying distances from the spill point. Analysis is performed by infrared-light absorption measuring devices. At the front side of the hall near the platform there are facilities to produce a uniform stream of air with wind velocities up to 2.5 m/s. The gases will be spilled in amounts of 2, 5 and 1o kg each.
The extension of the flammable region over the first minutes and the amount of gas mixture within the flammability limits will be calculated and compared with computational predictions based on a Pasquill-diffusionmodel for an area source growing with time during evaporation.

3.3. Explosive properties of the substances to be investigated

The explosive properties will be studied dependent on cloud size, mixture ratio with air and kind of ignition.

3.3.1. Limits of inflammability

The tests will be done in small balloons with a volume of some liters.

3.3.2. Reaction process in the case of an explosion

The experiments will be performed by means of a detonation chamber whereby the ignition- and shock wave [illegible]eration process will be observed by high speed optical photography (Schlieren method in connection with a rotating mirror camera).

3.3.3. Flame propagation rates

The gas mixtures will be enclosed in transparent balloons and ignited by different sources. The initiated deflagration or detonation will be

observed by means of high speed optical equipment.
Additional studies of this type will be carried out in thin walled PE tubes to collect information about the acceleration of the flames and hence, the possible change from a deflagration into a detonation.

3.3.4. Critical amount of explosive

The amount of explosive which is sufficient for the initiation of a detonation in a stoichiometric gas/air-mixture is determined.

3.4. Determination of the propagation functions of exploding unconfined vapour clouds

The flow field outside of a detonating or deflagrating vapour cloud will be determined for at least two different cloud sizes. The properties of the flow field from large scale explosions should be derived from the results of these tests on a much smaller scale.

3.4.1. Detonation

The detonation of a gas/air mixture, enclosed in balloons of 1.5 and 15 m^3, generates a shock wave in the surrounding air, whose pressure-time history is measured at several different distances.

3.4.2. Deflagration

The experiments will be performed with hemispherical balloons on the ground. Again the pressure-time history of the generated pressure waves is recorded.

3.5. Gas detonation transfer to adjacent clouds

3.6. Estimation of the effect of a gas explosion on reactor components

4. Experimental Facilities, Computer Codes

5. Progress to Date

3.2.

- Mounting of the heating appliance in the test platform
- electrical supply for blower and heating
- electronic control of the heating
- production of 3 propane pressure vessels with rapid opening mechanism

- adaptation of infrared-light-absorption devices to a 13 channel light spot recorder
- adaptation of impeller anemometer to light spot recorder for measuring of wind tunnel flow
- production of a mobile device for a thermo diffusion probe to measure the turbulence in the wind tunnel
- measurement of flow velocity field in the wind tunnel

3.3.3. For the determination of flame propagation rates the following experiments were filmed with high speed cameras:
- deflagration of stoichiometric gas/air mixtures enclosed in transparent balloons of 1/2 x 1.5 and 1/2 x 15 cm^3.
- deflagration of stoich. C_2H_4/air mixtures with reduced nitrogen concentration (O_2/N_2 = 4o/6o)
- deflagration of C_2H_4/ air mixture (1o.6 vol.% C_2H_4) in a PE cube with a length of side of 8 m.
- deflagration of gas/air mixtures in PE tubes (diameter o.8 m, length 3o m)

Most of the tests had to be done during daylight; in order to make the flame fronts visible on the film it was necessary to add appropriate substances to the mixtures ($Sr(OH)_2$ or NaCl).
The films of all experiments - except the PE tube tests - were analysed with respect to space velocities of the flames.

3.3.4. The critical amount of explosive was determined for stoichiometric propane/air mixtures. This was not possible for methane/air mixtures.

3.4.1. The propagation functions for detonating stoich. propane/air mixtures were determined.

3.4.2. The propagation functions for deflagrating stoich. CH_4/air, C_3H_8/air - and C_2H_4/air- mixtures were determined. The pressure-time history of the generated pressure waves was recorded by 23 pressure transducers which were installed along the axis of a x-y-coordinate system on the ground at distances between 2 and 12 m from the explosion center.

The pressure waves of the other balloon experiments mentioned above (3.3.3.) were recorded as well.

3.5. Two 15 m^3-balloons filled with a stoich. C_2H_4/air-mixture were mounted side by side connected by an opening of 2o cm in diameter. The detonation which was initiated in one of the two was not transferred to the adjacent balloon but was stopped at the opening.
In two other tests an acetylene/oxygene-mixture was ignited in a detonation chamber. The generated detonation continued in a PE tube containing a stoich. C_2H_4/air mixture,seperated by a thin foil from the C_2H_2-O_2-mixture. This was not the case when the tube was replaced by a balloon.

Results

3.3.3. A characteristic example of the flame propagation of a stoich. C_2H_4/air mixture, contained in a hemispherical balloon of 1/2 x 15 m^3, is shown in figure 1. The propagation rate in horizontal direction reaches a maximum value of about 1o m/s; for laminar propagation one would expect a space velocity of 5 m/s. The resulting difference may be due to turbulent effects. Some time before the rupture of the balloon the flame is accelerating in vertical direction and reaches a maximum of about 38 m/s. This behaviour is associated with buoyant effects. The buoyancy is accompanied by a wake on the ground which leads to a stagnation of the horizontal flame position.
n figure 2 the flame propagation of a stoich. C_2H_4/air-mixture with a O_2/N_2 ratio of 4o/6o is shown. The maximum velocity is about 7o m/s.

3.3.4. For stoich. propane/air mixtures the critical amount of explosive is between 7o and 8o g.
Even with 2.5 kg of explosive it was not possible to get a detonation in a stoich. methane/air mixture; a further increase of the amount of explosive is not reasonable unless larger gas volumes than 15 m^3 are used. Kogarko et al. (1965) reports a critical amount of explosive of 1 kg whereas Strehlow (1976) quotes about 2o kg.

3.4.2. The deflagration of the investigated gas/air mixtures produce pressure waves which propagate at sonic velocity. The positive pressure

phase is followed by a negative one; this is in accordance with theory that claims $\int p\, dt = 0$ at any fixed radial position.
In figure 3 the maximum overpressure Δp^+ is plotted versus the scaled distance r/r_o (r_o = balloon radius) for different stoich. mixtures and different balloon sizes. The same is done in figure 4 for stoich. C_2H_4/air mixtures with reduced nitrogen concentration (O_2/N_2 = 4o/6o).
The experimental part of the research program has been concluded besides of points 3.2. and 3.3.1. Dataanalysis of the deflagration tests is in progress.

7. Next Steps

The results of the deflagrative experiments will be documented in a partial report.
The data of the detonative balloon tests with propane/air mixtures will be analysed and combined with those of the ethylene tests.
Composition of the final report.

8. Relation with Other Projects

9. References

1. S.M. Kogarko et al. (1965)
   Investigation of spherical detonation of gas mixtures
   Combustion, Explosion and Shock Waves, Vol.1 No.2 pp. 15-22

2. R.A. Strehlow and W.E. Baker (1975)
   The characterization and evaluation of accidental explosions

3. R.A. Strehlow (1976)
   Private Communication

1o. Degree of Availability of the Reports

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815
6/15

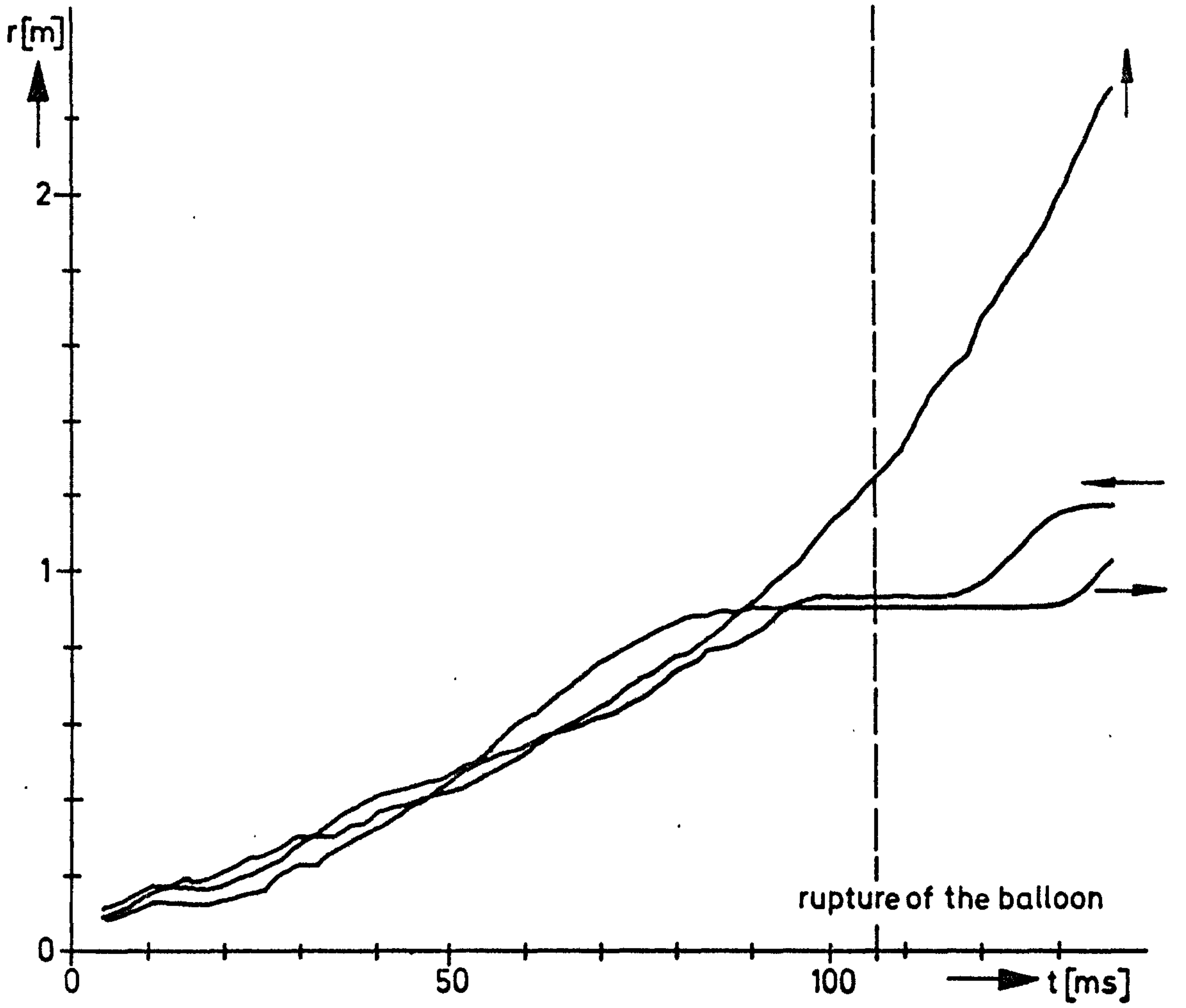


Fig.1 Flame propagation in stoichiometric ethylene/air mixture.

↑ vertical flame position

⇄ horizontal flame position

G16

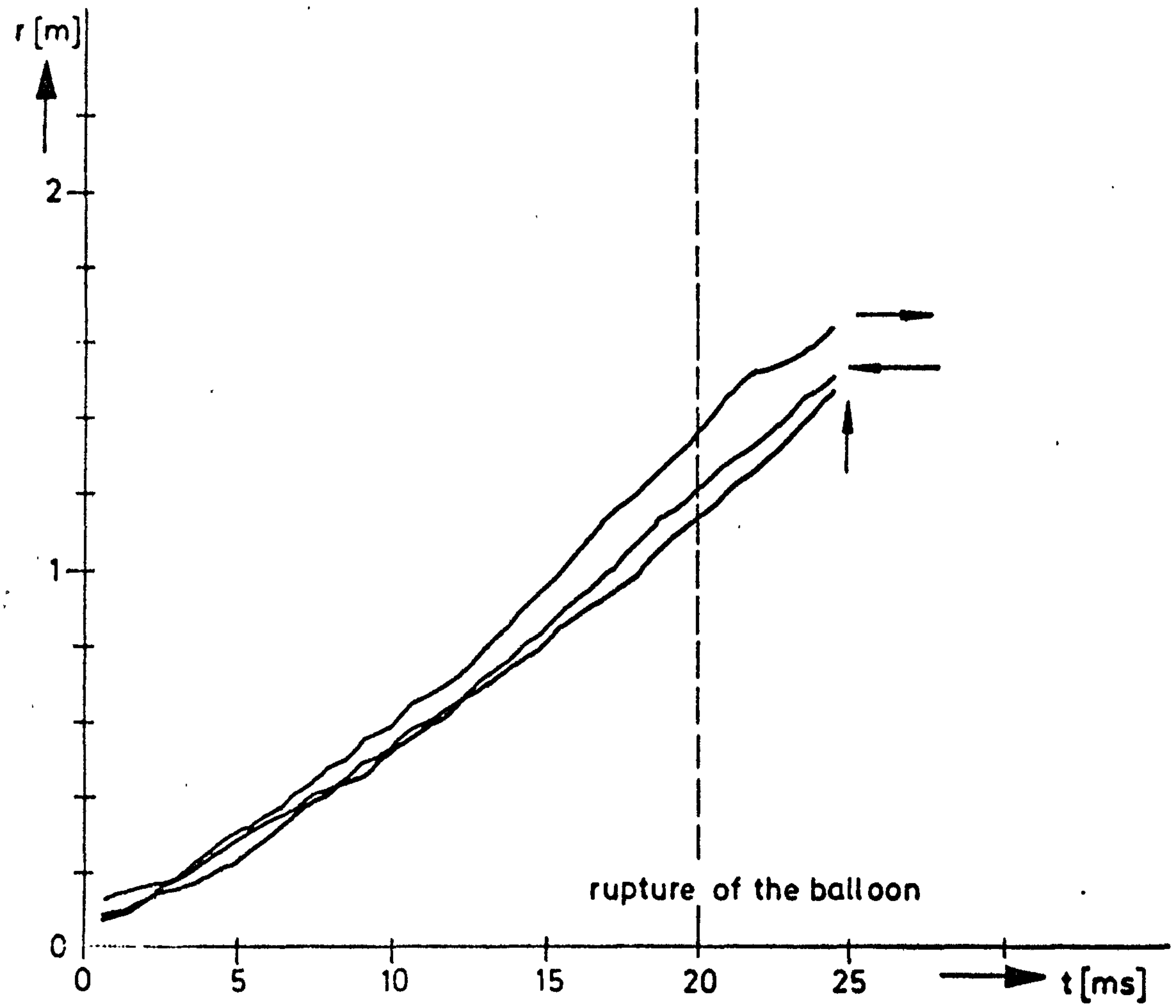
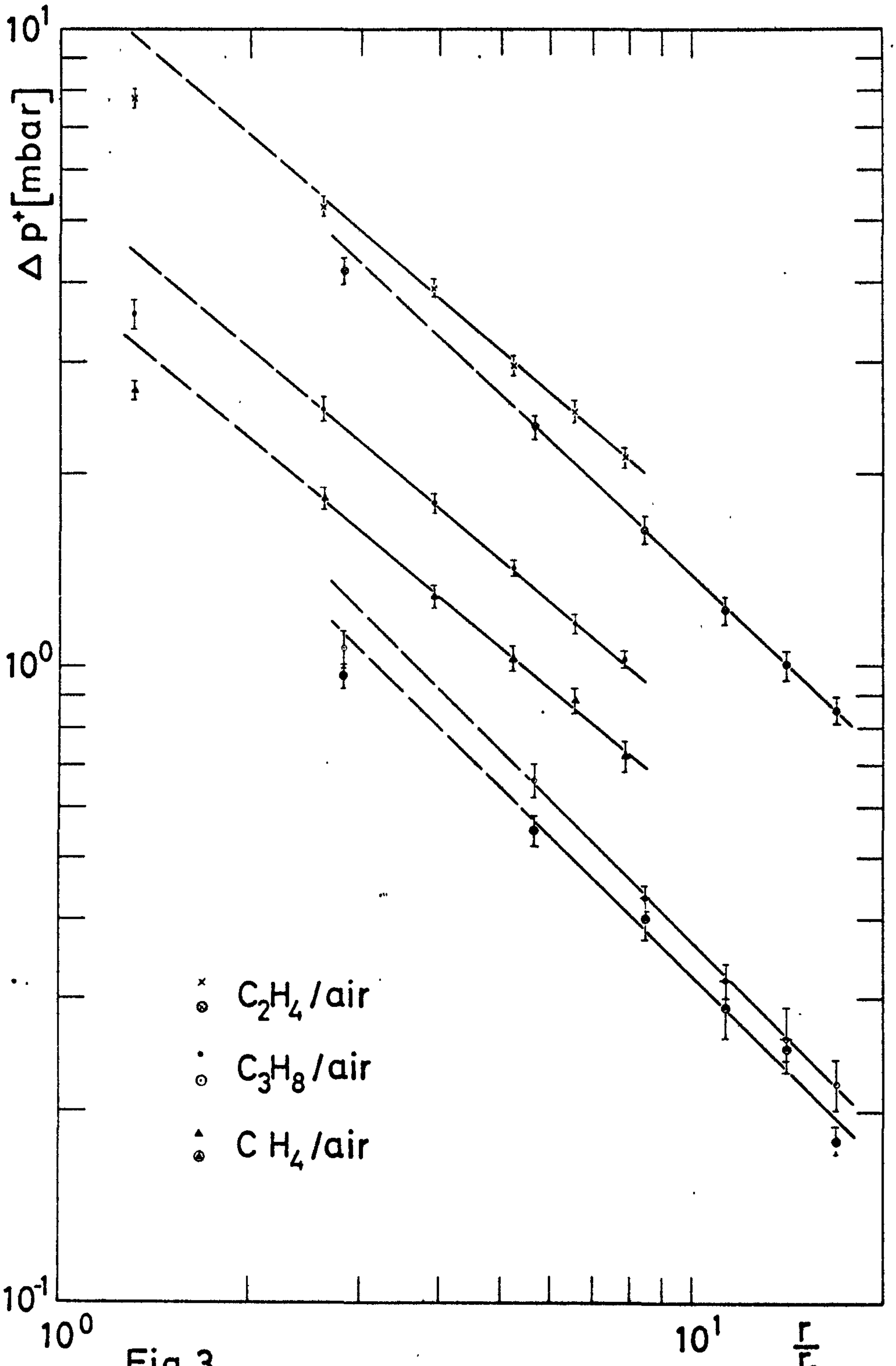


Fig. 2 Flame propagation in stoichiometric ethylene/air mixture (O_2/N_2 =40/60).
vertical flame position
horizontal flame position



10^1
10^0
10^{-1}
Δp^+ [mbar]
10^0
10^1
$\frac{r}{r_o}$
C_2H_4/air
C_3H_8/air
CH_4/air

Fig. 3

618

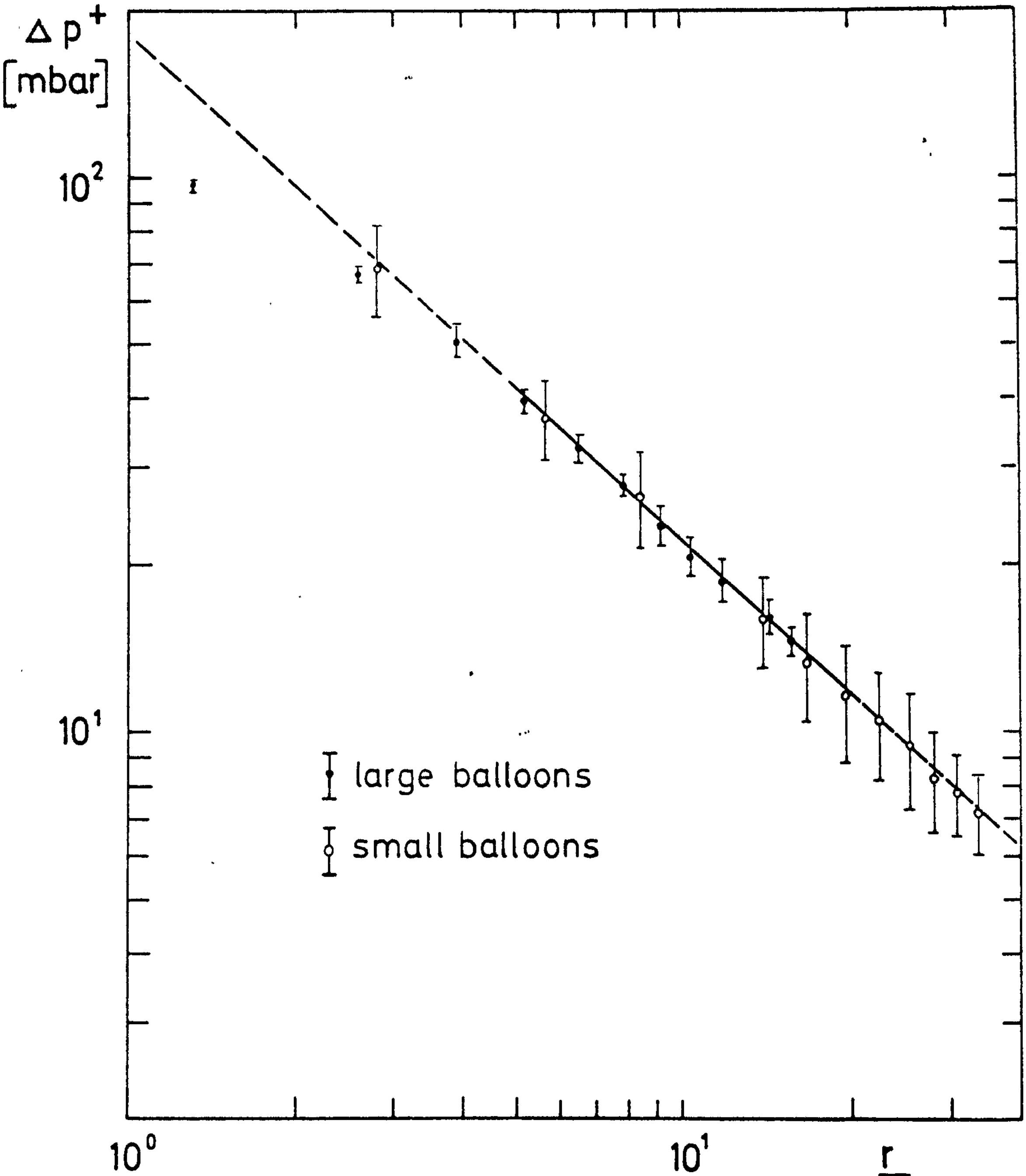


Fig. 4

Classification: 3.3	
Title 1 (Original Language): Beugung von Druckwellen um Reaktorgebäude (RS 102-9 - I.3.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BFMT ORGANIZATION: Fraunhofer-Ges. e.V. München
Title 2 (English): Diffraction of Shock Waves on Reactor Buildings	Project Leader: Dipl.-Ing. G. Hoffmann Ernst-Mach-Inst. 78 Freiburg
Initiated (Date): 1.10.73 — Completed (Date): 31.12.76 Status: Finished — Last Updating (Date): December 1976	

1. General Aim

For the security of nuclear power plants also dangers coming from outside have to be taken into account. One of them are the dynamic loads of the reactor containment caused by pressure waves. They are formed in all cases where energy is transferred by air, for example when a chemical or nuclear explosion takes place.

Therefore it is the aim of this project to find out the pressure-time-history of the pressure waves which are reflected and diffracted, when hitting a nuclear power plant, by means of model experiments in the shock tube.

2. Particular Objectives

Because there exist no data on pressure loads and their pressure-time-history running around a combination of buildings the objectives of this program are

2.1 to measure precisely the pressure loads of the single reactor buildings and to correct, if necessary, the data given in the standard literature

2.2 to get information on the shielding-, focusing- and reflection effects occurring in a reactor plant hit by a shock wave

2.3 to try to elaborate some general rules for the design of reactor plants in order to protect them optimally against shock waves.

3. Research Program

3.1 A representative nuclear power plant had to be to find out and from this the buildings which should not be destroyed by shock waves.

3.2 Work in literature had to give directions about the studies which had already been done in this matter or a similar one - and in the last case to try to transform it to the actual problem - and about studies which should be done. The valid model laws had to be exposed because the experiments could not be made in the original size.

3.3 The models were put in a small shock tube, and in shadowgraph experiments the critical places of the nuclear power plant were to be found, where high pressures would be expected.

3.4 At the places, found according to point 3.3 pressure transducers were installed in larger models. These models were mounted in a large shock tube (2.4 m in diameter). Shock waves of different strengths coming from different directions had to overrun the model. The pressure-time-history was measured.

3.5 Pressure-time-history was transformed to impulse-time-history. In connection with the results from point 3.3 and 3.4 this at last had to give the possibility of finding out the real critical places, the overloading of pressure and impulse with respect to pressure and impulse at the front of the containment.

4. Experimental Facilities, Computer Codes

All tests were made in two shock tubes; the shadowgraph experiments in a shock tube of 200 mm in diameter, the pressure-measurements in another of 2.4 m in diameter with a length of about 50 m. The principle of the plant is already described in the Annual Report A 74.

5. Progress to date

The work was started in 1973 with a literature-research, being the conditions for realizing the program. After settlement of the nuclear power plant to be investigated and after finding out the loading, experiments were made with shadowgraphs which finally enabled pressure measurements. As in 1975 such experiments were made on the reactor as well as the powerhouse (as single building and as building combination) now in the second test series building complexes reactor-supply plant-switch boards and reactor power house-supply plant-switch boards were tested whether at the diffraction process by superposition and focussing of shock waves at unsuitable places, increased pressure can occur compared with the peak reflection overpressure at the front of the reactor building. Here the building combinations were hit by two shock strengths (1,2 $\hat{=}$ 0,45 bar peak reflection overpressure, 1,4 $\hat{=}$ 1,0 bar) from eight different directions (each at 45°).

The extensive experiments supply numerous measuring records from which for each gage position and after normalizing the function

$$\frac{P}{P_{ro}} = f\left(\frac{t}{D/U}\right)$$

P = local pressure
P_{ro} = peak reflection overpressure
D = diameter of the reactor model
U = shock wave velocity
t = time of arrival of the shock front at the front edge of the reactor building

was found
and from that the impulse function

$$\frac{I}{(D/U)\ P_{ro}} = f\left(\frac{t}{(D/U)}\right)$$

I = impulse

resulted.

In order to show the critical places of the reactor plant where at the pretended building system an increase in pressure and impulse has to be taken into account and by which building walls these were caused, the pressure-time-history was chosen as comparative size at the front edge of the reactor. The other building models were removed for in any case the containment has to stand against the pressure or impulse appearing there. The measurements were made at rigid models. Therefore it is not possible to say anything about the destructive effect at buildings, moreover only the relative danger intensity can be determined. In order to find a dimension for this, pressure resp. impulse was chosen as reference size appearing at the front edge of the containment for the containment should in any case resist these loadings. The tests often lead to unexpected results as shown at the following example. Thus it is absolutely possible that at the same gage position completely different pressure-time-histories were measured if only the impacting direction is changed and all other test parameters are kept constant. If for instance the containment is placed behind the powerhouse met by the shock front under 45°, a pressure magnification of 54 % appears at the narrowest place between two buildings as shown in the pressure-place-graph of Fig. 1. If, however, the shock front hits normally the power house being also in front of the containment, at the same place a maximum pressure being by 40 % lower than that at the containment front edge (Fig. 2) is registered. The pressure-time-history in Fig. 3 shows that a building being in front of the containment is not always enough protective. Then a pressure-reduction appeared at conditions outlined in Fig. 2 (power house in front of the containment). Thus a pressure-rise resulted provided that the impacting direction was changed by 45°. Owing to superposition of reflected and diffracted shock front parts also building walls being lateral to the containment can resist a far greater pressure at an unfavorable arrangement - if a corner is formed - than the mentioned reference pressure. The pressure-time-histories for two gage positions confirm this in Fig. 4. The influence of auxiliary buildings on the pressure-time-history and on the impulse-time-history,resulted from there, is shown in Fig. 5.

6. Results

In the final report it is written about the total result of this research program having been finished in 1976. The final judgement about the effect of shock waves on nuclear power stations cannot yet be given as by means of future photos and diagrams and an extensive discussion of experts of the institute a revision could be necessary. However, it can already be stated, that the hitherto critical observation of the test results lead to the perception that a pure theoretical prediction about the pressure and impulse histories to be expected on different places of complex building plants is not possible. At the time being they can only be achieved experimentally.

Finally it can be said that the aims set at the beginning of the program were achieved and that neither costs nor time have been exceeded.

7. Next Steps

The final report will be elaborated.

8. Relations with other Projects

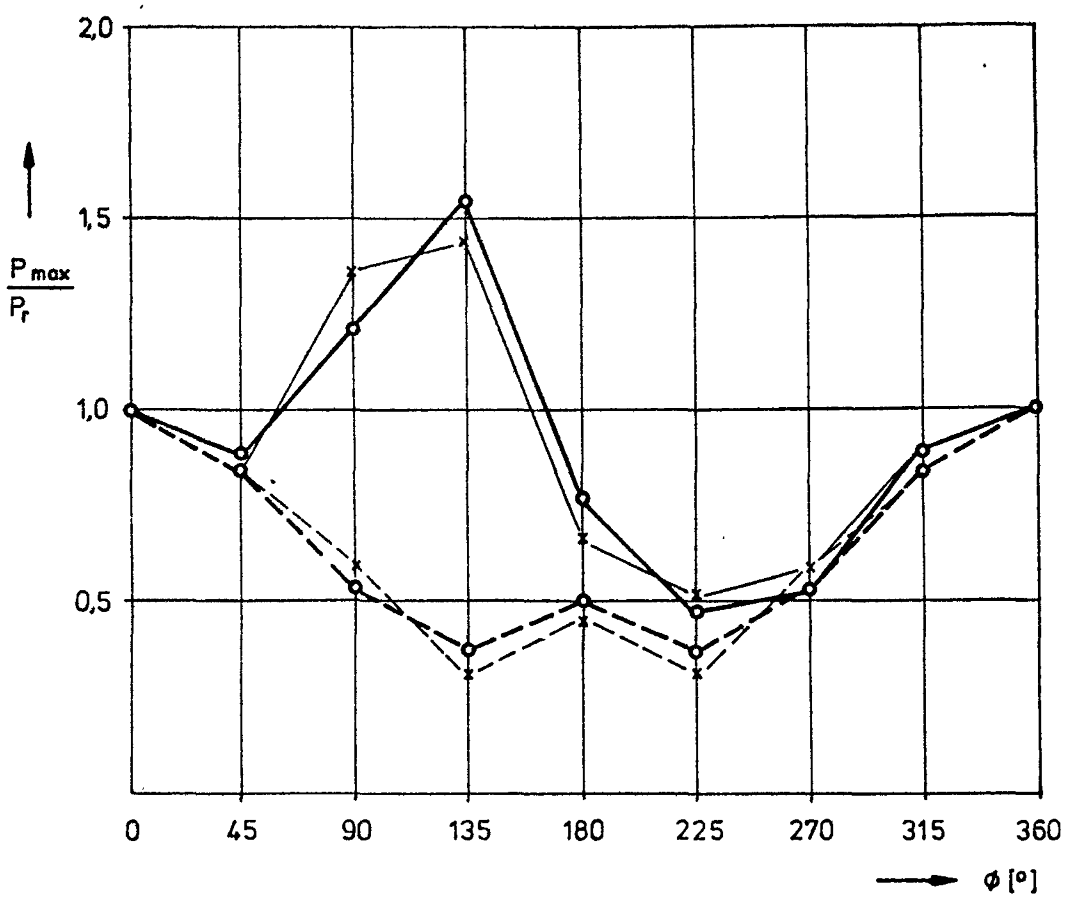
No relations exist to other projects with similar aims except in the own institute.

9. References

-

10. Degree of Availability of Reports

The final report is available after finishing and if BMFT gives the permission.



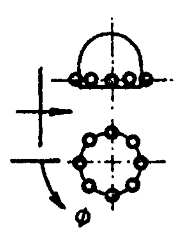
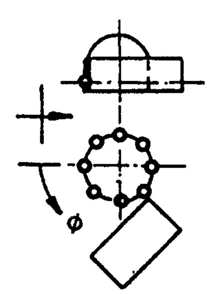
Reflection pressure [bar]	Symbol	Model	Symbol	Model
1,0 0,45	o---o x---x		o—o x—x	
┼→ Shockfront ● Gage position		P_{max} = Maximum pressure P_r = Reflection pressure		

Fig. 1 Pressure distribution on the reactor building
Comparison: Single building-combination

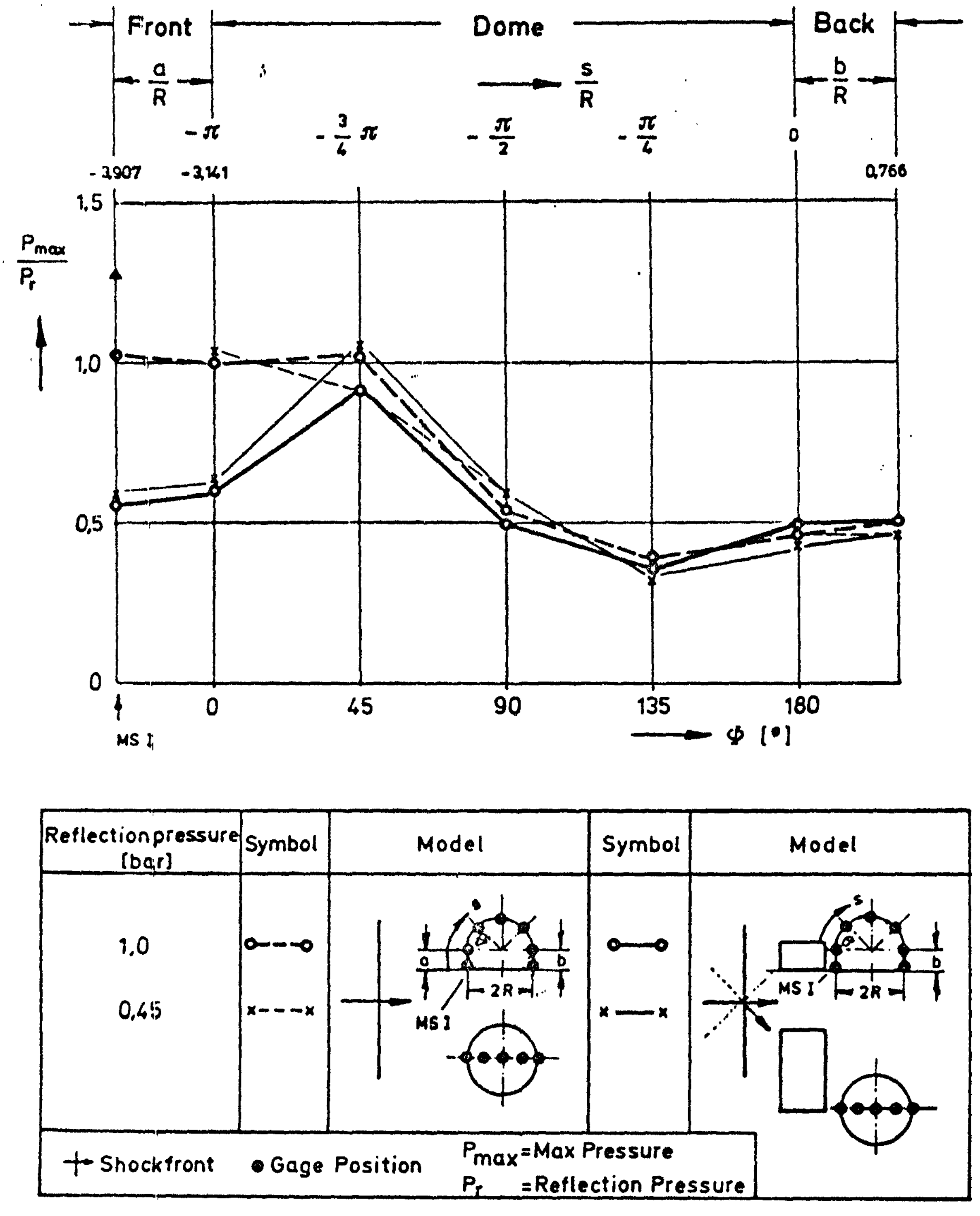


Fig. 2 Pressure distribution on the reactor building
Comparison: Single building-combination

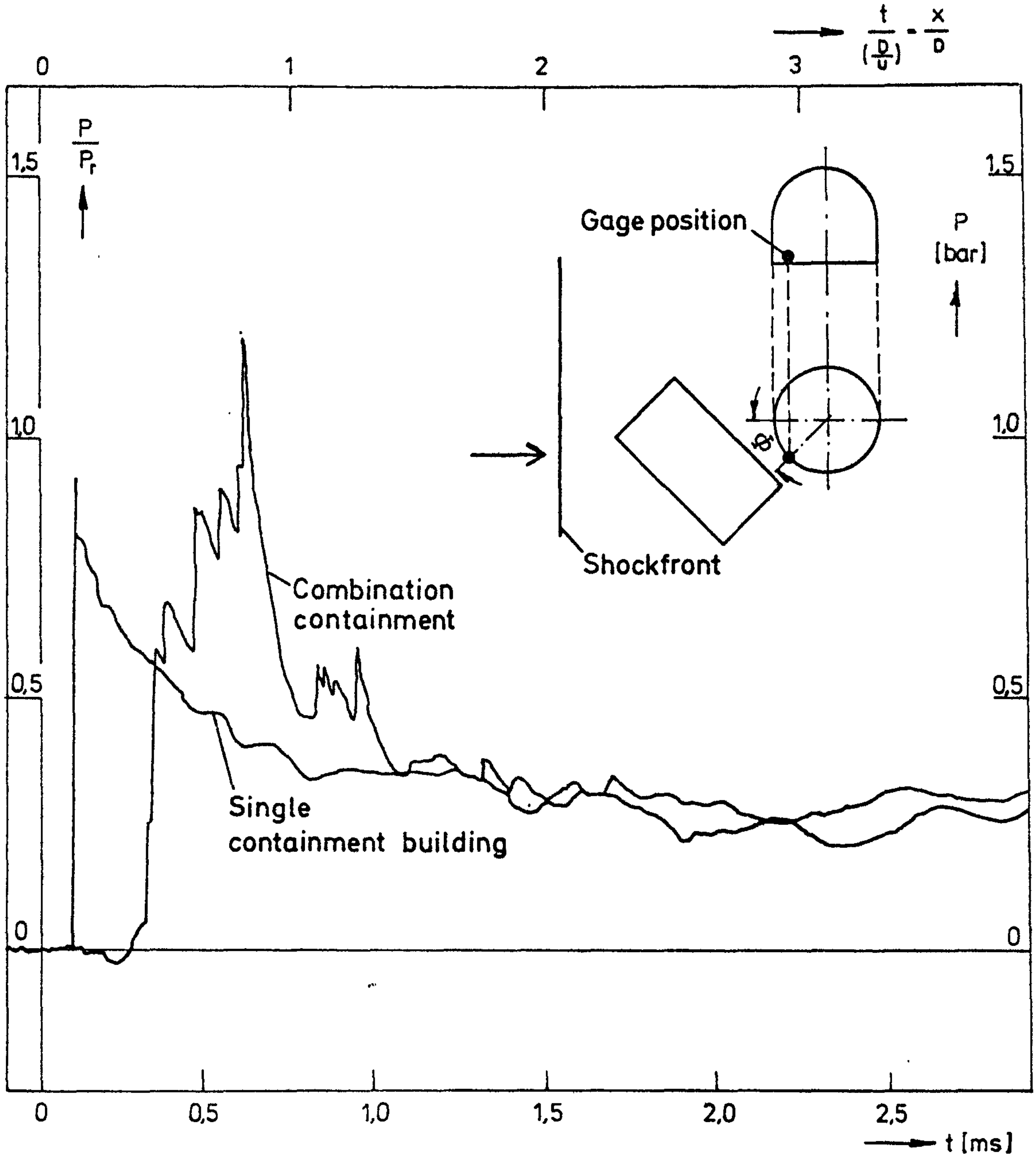


Fig. 3: Pressure-time-history on the reactor building at Φ = 45 ° Comparison single building-combination

P = local overpressure

P_r = peak reflection overpressure at the front of the reactor building = 1 bar

t = time; t = 0 - arrival of the wave at the front of the reactor building

D = reactor diameter = 30 (cm) $\frac{D}{u}$ = 0,76 (ms)

u = shock velocity

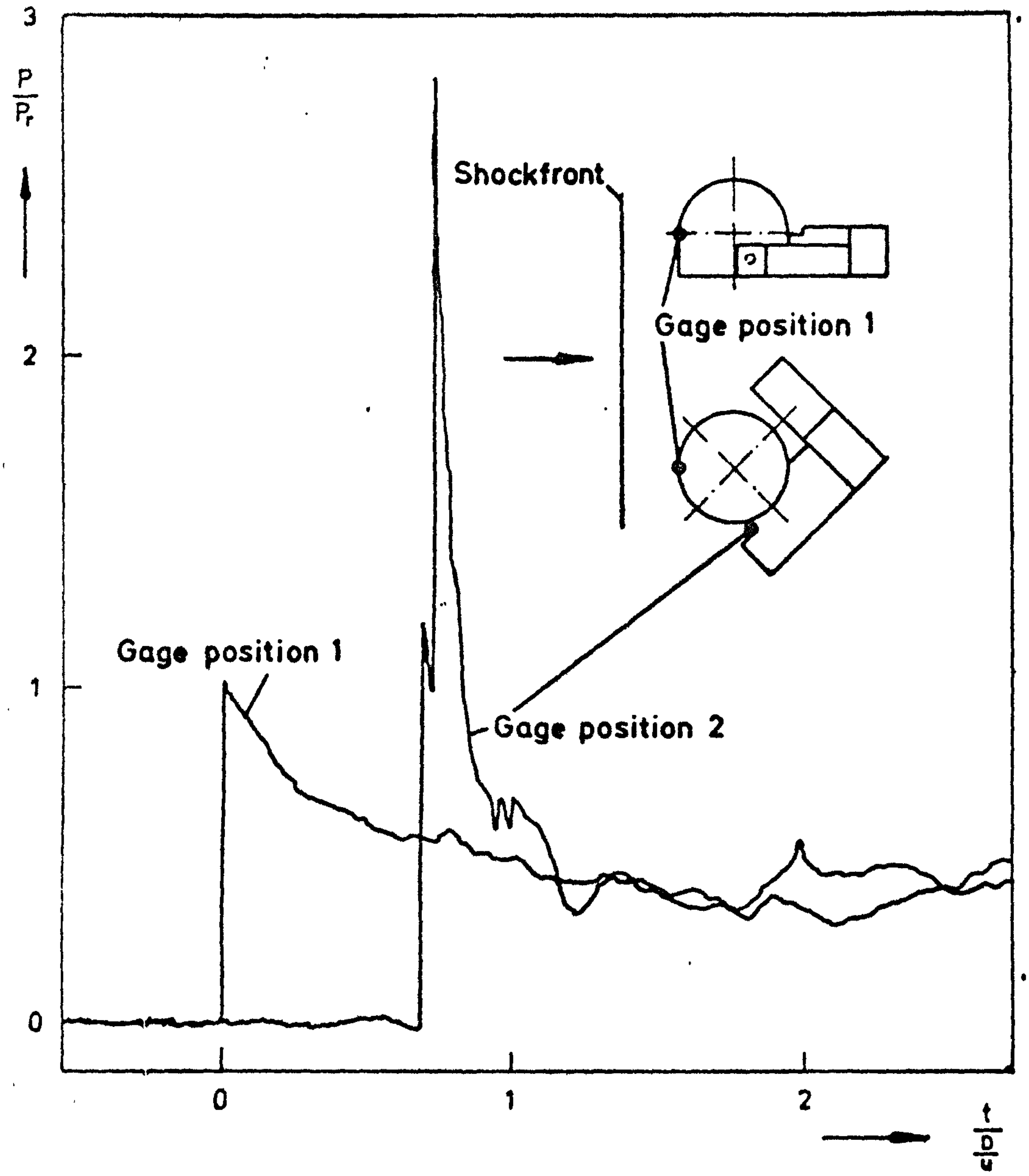


Fig. 4 Pressure-time-history on the reactor- and supply buildings at selected gage positions

P = local overpressure

P_r = peak overpressure at the front of the reactor building = 0,45 bar

t = time; t = 0 arrival of the wave at the front of the reactor building

D = reactor diameter = 30 (cm)

u = shock velocity $\frac{D}{u}$ = 0,81 (ms)

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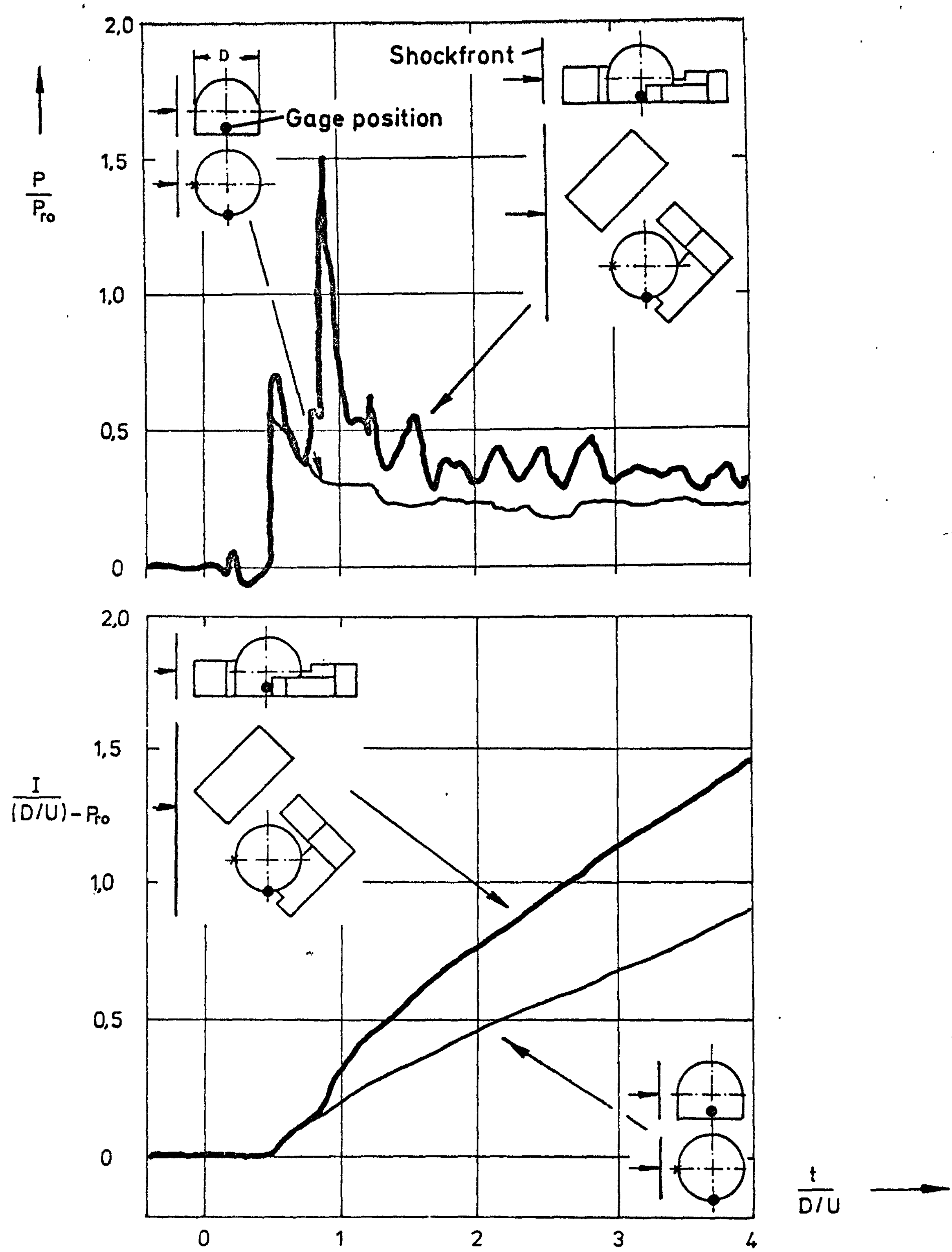

Fig. 5 Pressure-time-history and impulse-time-history (normalized)

P = local overpressure
I = impulse
D = reactor diameter (30 cm)
U = shock velocity
P_{ro} = peak reflection overpressure
t = time, t = 0 - arrival of the wave at the front of the reactor building

Classification: 3.3

Title 1 (Original Language): DICE THROW-Vorstudie (RS 173 - I.3.2., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** SDK, Lörrach
Title 2 (english): DICE THROW-Feasibility Study	**Project Leader:** H. Hofmann
Initiated (Date): 3[illegible].7.1975 **Status:** finished **Completed (Date):** 31.12.1975 **Last Updating (Date):** December 1975	

1. General Aim

In autumn 1976 reinforced concrete buildings shall be tested under pressure waves originating from explosions. These tests will take place at the test grounds "White Sand", New Mexico, USA, under the project name DICE THROW (4-nations project). An explosion of 500 tons TNT is planned, which will mainly serve military purposes.

The Bundesminister für Forschung und Technologie (BMFT = Federal Minister for Research and Technology) considered to take part in this project, mainly being interested in detailed analysis concerning the

- formation and propagation of pressure waves in atmosphere and ground
- reflexion and flow of pressure waves around the buildings of nuclear power plants
- maximum bearing capacity, safety against tumbling and vibration of structures induced by pressure waves

with respect to large structures. Another point he is interested in, is the quantification of the respective safety margins.

A review of design methods worked out during former nuclear weapon tests as well as a control of the results by means of advanced computational and measuring methods were of further interest. As a basis for the decision on whether a participation of BMFT is useful, pertinent and organizational conditions had to be clarified by this feasibility study under the aspect of nuclear safety considerations.

2. Particular Objectives

–

3.1 Experimental Facilities

Not relevant.

3.2 Research Program

–

4. Project Status

4.1 Progress to Date

Since a participation of BMFT with own objectives requires an adaption to the general test philosophy, first it was necessary to work out a concept and the possibilities given by the test arrangement. As a basis, the technical documents ot the Bundesminister für Verteidigung (BMVg = Federal Minister of Defence) and the Infrastrukturstab der Bundeswehr (InfrastrStBW = Infrastructure Staff of the German Federal Armed Forces) were used. BMVg also takes part with own objectives.

Furthermore, the possibilities and limits of an additional participation could be discussed in personal conversations with the competent representatives of the designing and performing institutions - Defense Nuclear Agency (DNA) and Waterways Experiment Station (WES). These discussions took place at the test grounds "White Sand" and in Albuquerque, New Mexico, on the occasion of a DICE THROW pilot test in August 1975 (explosion of 100 tons TNT).

To simplify matters and for the sake of clarity, first essential results have been summarized in the draft version of two research proposals - "Experimental Investigations and Preliminary Computational Analysis"; which have been submitted for judgement to the BMFT. The proposals include the design of a characteristic containment building, the corresponding pre-dimensioning and a distribution of mass. The final report is available, too.

4.2. Essential Results

4.2.1. Possibilities of the Large Scale Tests

The high energy release resulting from an explosion of 500 tons TNT allows to conduct tests with large buildings of nuclear power plants, e.g. containments, under reduced scale problems and at different excitation levels - low, medium and high - depending on the distance between the test building and the explosion center - about 100 to 300 m.

By the choice of an excitation level, tests in the linear and nonlinear region of structural behaviour are possible. Such large-scale tests have never been performed up to now and would be a valuable extension of present practice. For safety reasons, however, the performance of those tests will scarcely be possible in Germany.

4.2.2. Technical Performance

The large-scale test DICE THROW especially will present

- a better information on the structural behaviour of large buildings
- a comparison between modern computational and experimentally results, particularly a check of the extrapolation of computational methods and theoretical fundamentals to large structures and thus
- an extension of the range of knowledge required for nuclear reactor safety considerations with respect to the large nuclear power plant structures.

On account of a participation of BMFT, it is possible to build two additional characteristic large scale containments and to test them under the influence of pressure waves resulting from explosion.

Furthermore, the aboveground and underground cubic buildings erected by BMVg may be used for the objective of BMFT (e.g. for cable channels etc.). DNA may provide up to 300 channels for the instrumentation. As a completion, free-field records of other participants will be available in exchange for own data.

In addition material properties necessary for preliminary analyses and pre-dimensioning of the structures will be determined and made available on the test grounds. Results and evaluations of the pilot tests may be obtained under data exchange agreements.

From the technical point of view a participation in the large-scale test DICE THROW is recommended.

The final recommendation still depends on the clarification of several conditions. These include e.g. binding cost estimates for the structures to be ordered in the USA and further details concerning the efficiency of the measuring technique.

5. Next Steps

The project "DICE THROW-Feasibility Study" is completed.

6. Relation to Other Projects

At the time there are no other projects in execution concerning the behaviour of nuclear power plant structures under the influence of explosions with respect to tests of such a large scale.

7. Reference Documents

- Abschlussbericht RS 173 "DICE THROW-Vorstudie zur analytischen und experimentellen Erfassung von Reaktorbauwerken unter simulierter (Gas-) Explosionsbelastung im Grossmaßstab", Oktober 1975
- Anträge an den BMFT vom 3. September 1975
  - DICE THROW - Experiment   BMVg - Rü III 8 -
  - DICE THROW - Analytik   SDK

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- Reise- und Besprechungsbericht vom 26. August 1975
  - Informationsgespräche DICE THROW, Albuquerque, N.M., USA
  - Vorversuche zu DICE THROW (100 t TNT) auf dem Testgelände "White Sand"

8. Degree of Availability

-

Titre	Pays : FRANCE
Agressions d'origine externe sur les installations nucléaires : explosions chimiques non confinées dues à un environnement industriel ou aux voies de communication	Organisme directeur CEA/DSN-EdF/SEPTEN
Titre (anglais) External impacts on nuclear plants : unconfined chemical explosions due to industrial environment or to communication roads.	Organisme exécuteur CEA + ENSMA Responsable : J. DUCO (DSN) M. GOBERT (SEPTEN)
Date de démarrage : 30/09/75 Date prévue d'achèvement : 31/12/78 Etat actuel : Etude en cours Dernière mise à jour : 17/11/76	Scientifiques : M. PERROT (CEA-CESTA) M. LEYER (ENSMA)

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.

Objectifs particuliers :

1) Recherche d'une loi d'échelle pour l'onde de pression engendrée par la détonation (au sens strict) de mélanges gazeux.
2) Recherche de l'effet de divers paramètres sur la cinétique de l'explosion : caractéristiques du mélange, intensité et localisation de l'initiation, présence d'obstacles,...)
3) Recherche de modèles représentant les caractéristiques de l'onde de pression pour différents régimes d'explosion(déflagrations, déflagrations rapides, détonations).

Installations expérimentales et programme :

Les essais d'explosions sont effectués à deux échelles différentes.

a) En Laboratoire d'Energétique et de Détonique de l'Université de Poitiers (ENSMA) où une étude à caractère scientifique fondamental est menée sur des volumes de mélanges de quelques litres. On essayera de préciser la tendance des flammes de mélanges d'hydrocarbures et d'air à s'accélérer dans certaines conditions : points privilégiés de l'initiation de l'explosion dans le mélange, présence d'obstacles dans le mélange, mélanges avec gradients de concentration.
b) Sur un champ de tir du CEA (CESTA) où des essais avec volumes atteignant 100 voire 1000 m^3 peuvent être réalisés. Les moyens mis en place sont utilisés actuellement à l'étude de détonation de mélanges air-hydrocarbures et permettront par la suite de comparer, surtout dans le champ lointain, les effets des détonations et de déflagrations de mélanges identiques. Enfin on vérifiera si les résultats obtenus en laboratoire sont extrapolables au cas des grands volumes.

.../...

Etat de l'étude :

1) Avancement à ce jour :

a) L'installation du CESTA a permis d'étudier essentiellement les détonations de mélange Air-Hydrocarbures (Acetylène-Ethylène) sur des volumes allant jusqu'à 50 m^3. Les valeurs des différents résultats tels que mesures de surpression et vitesses de l'onde de choc ont été rassemblées et sont en cours d'interprétation à l'ENSMA.

b) Le laboratoire d'Energétique et de Détonatique de l'Université de Poitiers a entrepris le montage de l'installation qui lui permettra d'étudier l'accélération de la flamme.

2) Résultats essentiels :

a) Mise au point de méthodes de mesure des différents phénomènes liés aux explosions de mélanges de gaz et d'air à grande échelle.

b) Dans le cas des détonations, mise en évidence d'un effet de taille des ballons : pour une même valeur de coordonnée réduite ($d.m^{-1/3}$) on constate une augmentation de la valeur de la surpression en fonction du volume de gaz.

Prochaines étapes :

1) Après interprétation des essais de détonation des mélanges Air-Acetylène et Air-Ethylène (jusqu'à 50 m^3 inclus), étude de l'intérêt de procéder à des détonations de volumes supérieurs à 50 m^3.
2) Comparaison, entre déflagrations et détonations, des effets (surpression-impulsion) dans le champ lointain pour des mélanges identiques.
3) Etude de l'accélération de la flamme en fonction de l'intensité et du point d'initiation de l'explosion, de la présence d'obstacles, de gradients de concentration dans le mélange.
4) Etude de la secousse tellurique induite.

Relation avec d'autres études :

Formation et évolution de nappes de gaz dérivantes (en cours).
Réponse des structures d'une centrale nucléaire aux impacts liés aux explosions de gaz (lancement à l'étude).

Documents de référence :

"Risques sur les réacteurs nucléaires dus au voisinage d'installations et de transports pétroliers", J.DUCO, R.LE QUINIO - Rapport DSN 38, mai 1974.
"Compte rendu d'essais d'explosions de mélanges Air-Hydrocarbures (essais AMEDEE-1)", J.LEYER - Note CESTA/EX/ESP EA 76-1, 10 février 1976.
"Essais d'explosions de mélanges Air-Acetylène", J.PERROT - Procès-Verbal d'Essais CESTA/EX/ESP 1354, 26 octobre 1976.

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TNO - IWECO			CLASSIFICATION 3.2/3.3/7.1
TITLE: Responsieberekeningen voor reactorgebouw			COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs ORGANIZATION: TNO-IWECO
TITLE: (ENGLISH LANGUAGE): Dynamic response of reactor structures (building and containment)			PROJECTLEADER: Meijers
INITIATED: June 1974		LAST UPDATING: April 1977	SCIENTISTS: Van Beek Geertsema
STATUS: Progressing		COMPLETED: 1978	

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 (in progress)	PROJECT LEADER : M. MARINI

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, we 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 Sponsor CNEN Organisation CNEN
Studies of site engineering	
Date initiated November 1974 (present phase) Date completed in progress Last updating April 1977	Project Leader S. Polinari

4. POWER TRANSIENTS

Classification: 4	
Title 1 (Original Language): Untersuchung von Betriebstransienten bei Versagen des Schnellabschaltsystems (ATWS-Studie) (RS 153 - I.1.8, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Investigation of Anticipated Transients without Scram (ATWS-Study)	Project Leader: F. Winkler
Initiated (Date): 1. 11. 74 Status: Continuing Completed (Date): 30. 6. 77 Last Updating (Date): 31. 12. 76	

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

The design conditions for a second scram system can be specified on the basis of the results of this investigation if necessary. For this purpose some pre-investigations shall be executed.

3. Research Program

3.1 PWR:

The consequences of a failure of the shut down system are to be investigated on the transient behaviour, 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: $10^{o}C/sec$) speed, and the collective electro-mechanical insertion of the control rods (insertion time: 120 seconds) are considered.

4. Test Facilities

No test facilities necessary.

5. Progress to Date

PWR:

The ATWS accidents "loss of main heat sink and loss of auxiliary power" and "pressure drop after failure of the pressurizer safety value" were calculated with the code IQSBOX in order to study the time behaviour of the integral reactor power and the axial power density, Doppler- and coolant density feed back effects included.

The input-data, especially the coolant flowrate, the inlet enthalpy and coolant pressure were taken from the code LOOP 7. The cases were simulated for the power station Biblis A.

The influence of the capacity of the pressurizer-safety valves on the coolant pressure and other process parameters were investigated with LOOP 7.

BWR:

In the longterm emergency cooling program SAFE some subroutines were installed with respect to the neutron kinetic and steam-bubble feedback included.

The transient program DRAMP was modified and improved.

A modified circuit model was coupled with a simplified core model and tested.

6. Results

PWR:

The comparison of the calculations with IQSBOX and LOOP 7 showed that the integral power data were in a good agreement. As in the point kinetic model LOOP 7 the power density distribution is fixed on a constant value only the results of IQSBOX showed that the local power maximum was moved towards the lower part of the core because of the existing steam bubbles in the upper part. This is a good result concerning the film-boiling in the upper core region after "loss of main heat sink".

For the "pressure drop after opening of the safety value" the integral power was higher in LOOP 7 compared with a one dimensional IQSBOX calculation. This shows that the point kinetic results are rather conservative.

After a "loss of main heat sink" and "discharge over the pressurizer valves" a rather strong rise of the pressure was observed, caused by thé water filling the pressurizer. The discharge rate depends on the pressure; at higher pressure the coolant flowrate into the pressurizer over the surge line was lower. This effect however strongly depends on the coolant, whether the water is boiling or not.

BWR:

DRAMP now considers the different energy storage after rise and decrease of the pressure, the change of the core flowrate after decrease of the water level a.s.o..

First results with the modified circuit model showed good agreement with the expected behaviour of reactor pressure and water level after accidents.

7. Next Steps

PWR:

The code LOOP 7 will be further improved to reduce some conservativity (e.g. coupling with an one dimensional kinetics-code).

BWR:

Repeating of some calculations with the code "SAFE + Neutronkinetic" in order to reduce conservative data of earlier calculations.

Investigation of ATWS-transients during pump slow-down, stop of the feedwater pumps a.s.o.

Improvement of the models: Test of the modified circuit model and integration into the transient model DRAMP.

Variation of parameters, e.g. rod input time, speed of the internal pumps.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Classification: 4	
Title 1 (Original Language): Untersuchung von Betriebstransienten bei Versagen des Schnellabschaltsystems (ATWS-Studie) (RS 153 - I.1.8., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Investigation of Operation Transients during Failure of the Scram System (ATWS-Study)	Project Leader: G. Frei
Initiated (Date): 1. 11. 1974 — Completed (Date): 31. 7. 1976	
Status: Continuing — Last Updating (Date): 31. 12. 1975	

General Aim

The aim of this project is to determine the dynamic behaviour of PWR and BWR plants during different anticipated transients with failure of the shut down system (ATWS = Anticipated Transients Without Scram). The investigations shall show, whether limiting-values of the fuel elements, the core and the loop components are exceeded.

Particular Objectives

The design conditions for a second scram system can be specified on the basis of the results of this investigation if necessary. For this purpose some pre-investigations shall be executed.

Project Status/Progress to Date

PWR:

The consequences of a failure of the shut down system were investigated on the transient behaviour, corresponding to the RSK-guide lines. The analysis were conducted for the power plant Grafenrheinfeld and is valid for all 1300 MW-reactors of the present generation. The analysis started from normal operation of the reactor; it was assumed that only the shut down system failed and all the other systems were operating well.

The accident analysis was concentrated on the new core with high boron content (about 1100 ppm at full power), because the density effect for power reduction is rather low in this case. Besides this, the Dopplereffect has its maximum value for a now loaded core. Other reactivity effects were not considered in this study.

The case of feedwater break down was investigated, considering a dead time of the steam generator safety valves of 50 and 500 msec respectively. After this the opening of a life steam line safety valve was investigated at the end of a power cycle (strong negative temperature backfitting).

To verify the computer code LOOP 7, some commissioning tests of Biblis A under full power were recalculated:

- Reactor-Scram
- Loss of auxiliary power
- Load rejection to zero power with rod injection

The most important process data were compared with calculated values. Some boundary conditions however, e. g. variable speed of the coolant pumps, time behaviour of the reactivity after control rod injection had to be adepted to the test series. Some initial values, e. g. water content of the pressurizer and coolant pressure, had to be compared and adepted to values at the begin of the test.

Some corrections were made with respect to the time behaviour of the instrumentation. The influence of the undercooling water in the pressurizer was investigated, which has to be considered during the auxiliary power case or after scram.

BWR:

The behaviour of a typical BWR power plant was investigated for the KKB and KKI power plants, corresponding to the RSK guide lines (8 transient cases after failure of the shut down system). As protective actions the controlled slow down of the pumps to minimal (rate: 10 $^{o}C/sec$), speed, and the collective electromechanical insertion of the control rods (insertion time: 120 seconds) were considered.

The simulation code DRAMP was improved in some details.

Project Status/Essential Results

PWR:

The reslut of the ATWS-study was that the pressure of the primary system does not exceed ~10 % of the design pressure value in all cases; the transients were limited by voiding of the core. In spite of film boiling in some cases, the maximum fuel rod surface temperature will be below 650 °C. It can be stuted, that a second shut down system is not necessary.

The recalculation of the data from the commissioning tests of Biblis A with the code LOOP 7 used for the ATWS-calculations showed rather good agreement between experimental and theoretical values.

BWR:

The ATWS-investigations show, that in most cases the pressure does not exceed ~10 % of the design pressure. The highest calculated pressure value is 112 % of design pressure.

In most cases film boiling is avoided. Only in some cases of "loss of main heat-sink" critical heat flux is exceeded. However in these cases the maximal calculated cladding temperatures are below 700 °C for the 8 x 8 cases (KKI) and below 850 °C for the 7 x 7 cases (KKB). Maximal fuel center temperatures are in these cases 2150 °C (8 x 8) respectively 2840 °C (7 x 7).

The maximal amount of steam blown into the pressure suppression system within the first 200 seconds following the start of the transients is less than 50 full-power seconds, resulting in a 12-K temperature increase of the pool-water (After this period the reactor is subcritical).

In cases of loss of feedwater flow or onsite power the core shroud (fuel-bundles, upper core plenum, separators) remains completely filled with water. Outside the shroud the level, where the start of the emergency core cooling systems is initiated, is defined only for some serials. Therefore the depressurization system is not started.

Next Steps

PWR: Additional work is still under consideration.

BWR: Investigation of ATWS accidents; Improvement of the computer codes; Parameter variations: rod input time, slow down time of the pumps.

Relation with Other Projects

see RS 153 - A 74

Reference Documents/Degree of Availability

No reports available.

Classification : 4.

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Title 1 (original language) Etude analytique des transitoires accidentels des réacteurs P.W.R.	COUNTRY : FRANCE SPONSOR : CEA ORGANIZATION CEA
Title 2 (english) Analytical study of accidental transients for P.W.R	Project leader CEA/DSN/SETS J.P.MERLE Scientists :
Initiated (date) 1973 Completed : (date) Status : progressing Last updating (date) Janvier 1975	

1. But général

Etude de la réponse des principaux paramètres de fonctionnement d'un réacteur à une perturbation d'origine accidentelle.

2. Objectifs particuliers

Etudes des principaux transitoires accidentels qui interviennent dans l'analyse de sûreté des réacteurs de puissance.

3. Installations expérimentales

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4. Etat du projet

Code de calcul SIRENE (P.W.R) en cours de tests (comparaison avec les résultats du constructeur).

5. Prochaines étapes

Extension des possibilités de programme (quant aux types de perturbation pouvant être traitées).

Titre	Pays : FRANCE
Développement des moyens de calcul nécessaires à l'étude des transitoires anormaux sur les réacteurs PWR.	Organisme directeur : CEA/DSN
Titre (anglais) Development of computer codes necessary to study of abnormal transient conditions on PWR.	Organisme exécuteur : CEA/DSN-SETSSR Responsable : M. GOMOLINSKI (SETSSR)
Date de démarrage : 01/01/74 Date prévue d'achèvement 31/12/77 Etat actuel : Etude en cours Dernière mise à jour : 25/11/76	Scientifiques : JP. MERLE

Objectif général :

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.

Objectifs particuliers :

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.

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 les générateurs de vapeur et un modèle de pressuriseur fonctionnant indifféremment en simple phase (vapeur ou liquide) et en double phase peuvent être intégrés à la version actuelle.

2) Résultats essentiels :

Etude de transitoires de fonctionnement et de transitoires accidentels de classe 2 et 3. Etude de transitoires de dimensionnement du circuit primaire.

.../...

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 boucles et d'une répartition dissymétrique des températures dans le coeur.
Calcul de la répartition des débits dans les lignes vapeur lors d'une rupture de tuyauterie vapeur.
Extension des possibilités de la version actuelle (traitement de la double phase).

Relation avec d'autres études :

Etude des ATWS (142-31-10).

Documents de référence :

"Analyse de fonctionnement d'un PWR : le code SIRENE", M.TRAN TUC VI - Note SERMA.
"Etude des transitoires accidentels - centrale de Fessenheim", JP.MERLE - Note Technique SETSSR 88, mai 1976.
"Un modèle d'études dynamiques de pressuriseur", Note SERMA-SPM 169 T.
"Adaptation du code SIRENE à l'étude des transitoires accidentels des centrales à eau pressurisée", JP.MERLE - Note SETS 69, mai 1975.
"Régulation de puissance dans le code de fonctionnement pour les PWR", JP.MERLE - Note SERMA 381 T, janvier 1974.

CLASSIFICATION
4

TITLE 1	CODES DE CALCULS DES TRANSITOIRES ACCIDENTELS	COUNTRY FRANCE SPONSOR E.D.F/SEPTEN ORGANIZATION E.D.F.
TITLE 2	CALCULATION CODES FOR ANTICIPATED TRANSIENTS	Project Leader E.D.F. /SEPTEN/T
Initiated 1973	Completed 1975	Scientists M. LARMINAUX
Status Codes opérationnels.	Last updating: 20.01.75	

I - GENERAL AIM

Etude des conséquences de régimes transitoires accidentels de dimensionnement.

II - PARTICULAR OBJECTIVES

- Code BABEL: études d'accidents en régime symétrique. Le code comporte une représentation simplifiée de la chaudière nucléaire et de la partie secondaire, un calcul de pression et de niveau est fait dans le pressuriseur et dans la partie secondaire du générateur de vapeur.

- Code ETINCEL : études d'accidents dissymétriques. Le code (modèle 2 boucles) comporte la représentation de deux canaux dans le coeur et peut prendre en compte des mélanges plus ou moins importants dans les zones d'entrée et de sortie. Il permet l'étude d'accidents conduisant à des inversions de débits primaires.

III - EXPERIMENTAL FACILITIES AND PROGRAMME

Néant.

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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.

Titre Etude de transitoires anormaux sans chute de barres sur les PWR (ATWS)	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) Anticipated Transients analysis without Scram on PWR (ATWS)	Organisme exécuteur : CEA/DSN-SETSSR Responsable : A. CARNINO (SETSSR)
Date de démarrage : 01/01/77 Date prévue d'achèvement :31/12/78 Etat actuel : Etude en cours Dernière mise à jour :01/01/77	Scientifiques : J.P.MERLE

Objectif général :

Analyse et calcul des "ATWS" pour élaborer une doctrine sur le problème de prise en compte (ou non) de ce type d'accident dans le dimensionnement des centrales.

Objectifs particuliers :

Etude des transitoires dans l'ordre suivant :
- Perte de l'eau alimentaire
- Perte des alimentations électriques
- Dépressurisation primaire par ouverture d'une soupape.

Examen de dispositions particulières éventuelles pour assurer un dimensionnement correct des centrales.

Etat de l'étude :

1) Avancement à ce jour :

Analyse détaillée des différents documents américains sur les ATWS (WCAP 8330 - Status Report de NRC)

2) Résultats essentiels :

Document définissant les hypothèses à utiliser pour les premières séries de calculs.

.../...

Prochaines étapes :

Calcul détaillé des transitoires et de leurs conséquences par le constructeur Framatome et à l'aide des codes de Westinghouse, et mêmes calculs par les codes DSN.
Comparaison de ces calculs, discussion des hypothèses et modèles

Documents de référence :

WCAP 8330

Status Report

Title 1 (Original language)	Classification
Dynamic studies for safety analysis	$\boxed{4}$ - 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.

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.

Title 1 (Original language)	Classification
Statistical analysis of randome signals	$\boxed{I}$ - 3 - 4 - 8 IO - I4
Title 2 (English)	Country ITALY Sponsor / Organisation CNEN
Date initiated 1966 Date completed in progress Last updating April 1977	Project Leader A. Federico

Classification 4.1

Title 1 ANDYCAP: Tré dimensional dynamisk model af kogendevands reaktor kerne.	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 ANDYCAP: 3-D-dynamical model of a BWR-core.	Project leader: P. Skjerk Christensen
Initiated: 1969 Completed: 1972 Status: in use, being improved Last updating: Currently	Scientists: P. Skjerk Christensen

1. General aim

The purpose of the model is to describe and follow transients in a BWR core due to pertubations of process variables in timescale 1-100 seconds.

2. Particular objectives

The project is particularly aimed at normal and abnormal conditions in the reactor. The model is based on a three dimensional nodal description of the core as the neutronic part whereas the hydraulics model consists of a number of parallel one dimensional channels coupled at the lower and upper plenum. A recirculation loop containing a pump is included. In practical calculations the number of nodes has to be limited to some 2000, and the number of hydraulic channels to 30, due to the computer time which on a CDC-6600 is a factor of 100 times the reactor time, strongly depending on the character of the transient. The transients can be initiated by control rod movement, steam load disturbance, feed water disturbance, and main circulation pump disturbance.

3. Experimental facilities

4. Project status

1. Progress to date: A version of the code is in use
2. Essential results:

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5. Next steps

Work is in progress directed to speed up the code.

6. Relation with other projects

7. Reference documents

To be issued

8. Degree of availability

Not available

Classification 4.1

Title 1	COUNTRY Denmark
Kontrolstavs udskydning i en kogendevands-reaktor	SPONSOR DAEC Risø
	ORGANIZATION DAEC Risø
Title 2 Rod ejection transients in BWR	Project leader: P. Skjerk Christensen
Initiated: 1974 Completed: Status: in progress Last updating: currently	Scientists: B. Thorlaksen

1. General aim

The aim is to construct a model which describes the reactor transient following the ejection of one of the control rods in a BWR. The model is an extension of the ANDYCAP code which describes the dynamics of a 3-D nodal model of a BWR-core and pressure vessel. The movement of the control rod is calculated assuming critical flow in the guide tube, and the velocity limiter on the rod has been taken into account.

2. Particular objectives

1. The power distribution is calculated in three dimensions, but the local peaking factor has to be found by adequate box calculations.

2. The fuel model includes the possibility of a steam explosion initiated by molten fuel.

3. The consequences of the transient pertinent to the core and pressure vessel are calculated.

3. Experimental facilities

4. Project Status

1. Progress to date: A model for the movement of a control rod after a failure of the housing has been made. 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 will be freely available.

Classification 4.1, 4.2, 4.3

Title 1 PWR-stations dynamik model	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 PWR: A PWR power plant dynamics model	Project leader:
Initiated: 1972 Completed: Status: in 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.
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 DAEC Risø ORGANIZATION DAEC Risø
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, feed-water 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

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)	**Country** ITALY **Sponsor** CNEN **Organisation** CNEN
Overpower tests	
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

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: 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 1977	SCIENTISTS: Bruens
STATUS: in progress — COMPLETED: 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 programme : –

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.

Next steps
- digital model of the pressurizer
- 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

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Classification $\boxed{4.1}$ 4.2, 4.3

Title 1

BWR-stations dynamik model

COUNTRY Denmark

SPONSOR DAEC Risø

ORGANIZATION DAEC Risø

Title 2 Development of a Dynamic Model of a BWR Nuclear Power Plant.

Project leader: P. Skjerk Christensen

Initiated: 1973

Completed: 1976

Scientists: E. Nonbøl

Status: in use

Last updating: currently

Classification $\boxed{4.1}$, 4.2, 4.3

Title 1 PWR-stations dynamik model	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 PWR: A PWR power plant dynamics model	Project leader:
Initiated: 1972 Completed: Status: in use, being improved Last updating: Currently	Scientists: P. la Cour Christensen P. Skjerk Christensen

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TH-Delft	CLASSIFICATION: 4.1 4.2 4.3
TITLE: Ontwikkeling van een hybried computermodel voor de simulatie van storingen en ongevallen in een drukwaterreactor.	COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs ORGANIZATION: TH-Delft
TITLE: (ENGLISH LANGUAGE): Development of a hybrid-computermodel for the simulation of transient and accident conditions in a PWR.	PROJECTLEADER: Latzko
INITIATED: October 1974 / LAST UPDATING: May 1977	SCIENTISTS: Bruens
STATUS. in progress / COMPLETED: 1978	

Classification: 4.3	
Title 1 (Original Language): 3D-Transientenprogramm - Modifizierung eines 3D-Transientenprogramms für den SWR (RS 178 - II.1.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU Offenbach
Title 2 (English): Modification of a 3 Dimensional Transient Code for BWR	Project Leader: Dr. Lockau
Initiated (Date): 1. 9. 75 Completed (Date): 31. 8. 77 Status: Continuing Last Updating (Date): 31. 12. 76	

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 this conservative factors can be corrected, without loosing safety margin. Therefore a 3 dimensional program shall be developed, which gives more realistic details for the calculation of unsymmetric incidents in the core.

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 concept for the coupling of IQSBOX and FRANCESCA was developped, using the existing feedback-model THEDY. The data transfer between nuclear and thermohydraulic overlay was not changed, only the thermohydraulic overlay was modified. (The physical content of THEDY was replaced by FRANCESCA).

The steam void relation of FRANCESCA was tested by the recalculation of Studsvik-experiments.

The treatment of the radial and axial reflector caused some difficulties. The input was rather complicated.

Comparative 1D-tests were conducted with the COSBWR and IQSBWR codes.COSBWR uses a finite difference solution, whereas IQSBWR uses Galerkin-Functions of 4^{th} order.

For the integration of IQS-BWR into the physical program system the program PANBOX was developped.

6. Results

The coupling of FRANCESCA and IQSBOX was completed. The thermohydraulic module calculates from the 3D power density distribution the feedback parameters (fuel temperature, moderator density). From the local fuel temperature and moderator density the change of the neutron cross-sections was determined. The effective cross sections are calculated by a special fit program.

The initial steady-state is calculated by a 3D-reactor program before the transient starts. A special program transfers the 3D burn-up distribution, the control rod distribution and the initial thermohydraulic conditions.

The recalculation of bundle tests by FRANCESCA showed that in the subcooled region the recondensation constant should be $R_1 = 1$. The drift velocity had only a small influence in the boiling region. Totally a good agreement was found between experimental and theoretical results.

The comparative tests of the 1D COSBWR and IQSBOX-SWR calculations were in good agreement (within 2 %). After using 40 axial boxes instead of 20 the differences in the power density distribution were only very small.

The single channel model was extended to three channels with common nuclear cross-sections and feedback coefficients in order to study the influence of the coolant redistribution.

An example of preheater failure (KKW test 134) was used to compare transient calculations of 1D IQS-BWR and COSBWR calculations. The IQS-BWR results were somewhat lower than the experimental results.

7. Next Steps

Some tests shall be conducted with postcalculations of pump failures, measured at KKW and a single scram at KRB.

The preliminary version of IQS-BWR shall be improved with respect to larger problems, integration into the physical program system and better handling.

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 PWR-stations dynamik model	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 PWR: A PWR power plant dynamics model	Project leader:
Initiated: 1972 Completed: Status: in use, being improved Last updating: Currently	Scientists: P. la Cour Christensen P. Skjerk Christensen

Classification $\boxed{4.1}$, 4.2, 4.3

Title 1	
BWR-stations dynamik model	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
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

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TH-Delft	CLASSIFICATION: 4.1 4.2 4.3
TITLE: Ontwikkeling van een hybried computermodel voor de simulatie van storingen en ongevallen in een drukwaterreactor.	COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs ORGANIZATION: TH-Delft
TITLE: (ENGLISH LANGUAGE): Development of a hybrid-computermodel for the simulation of transient and accident conditions in a PWR.	PROJECTLEADER: Latzko
INITIATED: October 1974 / LAST UPDATING: May 1977	SCIENTISTS: Bruens
STATUS: in progress / COMPLETED: 1978	

5. BEHAVIOUR, TRANSPORT AND RELEASE OF RADIOACTIVE SUBSTANCES

Classification 5

Title 1	
Dosisbelastninger i A-kraftværker	COUNTRY Denmark SPONSOR Research Establishment Risø ORGANIZATION Research Establishment Risø
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.

Titre	Pays : FRANCE
Programme d'étude sur la contamination des circuits primaires PWR (coordination)	Organisme directeur : CEA
Titre (anglais)	Organisme exécuteur : CEA/IPSN/DSN
Studies program on PWR primary systems contamination	Responsable : J. GUIRLET
Date de démarrage : 1977 Date prévue d'achèvement : Etat actuel : en cours Dernière mise à jour : 30/04/77	Scientifiques : J. SEVEON

Objectif général :

Bilan sous l'angle de la sûreté des diverses actions entreprises au CEA liées au problème de l'activité des circuits.

Objectifs particuliers :

Emission des produits de fission à partir du combustible .
Etude du comportement dans les circuits en particulier des phénomènes de dépôt.
Etablissement de modèles et codes de calcul.

Installations expérimentales et programme :

BOUFFON
BIHAN ELDORADO CIRENE
IRENE

Etat de l'étude :

1) Avancement à ce jour :

Début de recensement et de classification des expérimentations faites au CEA.

Relation avec d'autres études :

Expérience de fonctionnement des PWR.

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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 SEMA.

IV - PROJECT STATUS

4.1 - Progress to date

- début des essais sur SEPAI,
- campagne de mesures sur SEMA.

4.2 - Essential Results

La campagne de mesures effectuée à SENA a permis de montrer que les produits de corrosion, qui se dissolvent du fait de l'abaissement de température, sont efficacement retenus sur résines synthétiques.

V - NEXT STEPS

- fin des essais de dissolution sur SERAI,

- étude du taux de relâchement,

- essai de décontamination d'une boucle de SENA.

VI - RELATION WITH OTHER PROJECTS

Néant.

VII - REFERENCE DOCUMENTS

VIII - DEGREE OF AVAILABILITY

E.D.F.

130-11-14/4112-01

Classification : 5-1 703

Titre	Pays :
Investigation et développement des méthodes d'identification et de localisation des défauts de gaine pour les réacteurs à eau.	FRANCE
	Organisme directeur : CEA
Titre (anglais)	Organisme exécuteur : CEA/SES-SAI
Investigation and development of identification and localisation method of clad failure for PWR.	Responsable : P.DOUET (SAI)
Date de démarrage : 01/1/76 Date prévue d'achèvement 01/12/78 Etat actuel : en cours Dernière mise à jour : 15/2/77	Scientifiques :

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.

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).

Etat de l'étude :

Avancement à ce jour :

- 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 l'EDF.
- Mesure du rapport de l'activité de deux nucléides. Un appareil a été installé.

.../...

Prochaines étapes :

Installation d'un appareillage expérimental sur le réacteur PWR de Bugey 3.
Elaboration d'une méthode de détection puis de localisation de rupture de gaine .

Documents de référence :

"Evaluation de la concentration d'un nucléide dans les boucles de refroidissement d'un réacteur" (premiers résultats) - Rapport SES/SAI.

Classification : 5-1

130-16-13/4112-04

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Titre	Pays : FRANCE
Ruptures de gaine et dissémination des produits de fission dans les réacteurs à eau. Programme BOUFFON.	Organisme directeur : CEA-EDF
Titre (anglais) C[illegible]ng ruptures and fission products dissemination in [illegible]. BOUFFON project.	Organisme exécuteur : CEA/DMECN-DMG (GRENOBLE) Responsable : JC.JANVIER (DMG)
Date de démarrage : 01/01/76 Date prévue d'achèvement : 31/12/80 Etat actuel : en cours Dernière mise à jour : 01/03/77	Scientifiques : P.CHENNEBAULT G.KURKA

Objectif général :

Etude des régimes d'émission des produits de fission par un crayon présentant une rupture de gaine.
Etude de la formation des dépôts de produits de fission sur les parois du circuit primaire.

Objectifs particuliers :

Modélisation des phénomènes étudiés.
Etude des divers cas de fonctionnement :
- Rupture de début de vie,
- Rupture à haut taux de combustion,
- Régime stable de fonctionnement,
- Suivi de réseau,
- Téléréglage,
- LOCA.

Installations expérimentales et programme :

Réacteur utilisé : SILOE.
Dispositifs expérimentaux : BOUFFON et BOUFFON-JET.

Etat à l'étude :

1) Avancement à ce jour :

La première irradiation suivi de réseau est terminée.
Une maquette permettant de provoquer une rupture de gaine en cours de fonctionnement est réalisée.
Une maquette chauffante pour mise au point du dispositif d'étude du LOCA est en cours d'essai.
L'étude du BOUFFON-JET pour étude des dépôts est en cours.

.../...

2) Résultats essentiels :

Les cyclages journaliers (suivi de réseau) conduisent à une augmentation de l'activité du circuit primaire par rapport au régime stable.
On observe une bouffée d'activité lors du transitoire de baisse de puissance.

Prochaines étapes :

- Essai de confirmation pour les dégagements en régime stable.
- Essai du dispositif BOUFFON-JET.
- Essai de confirmation du comportement suivi de réseau.
- Essai avec rupture à haut taux de combustion.

Documents de référence :

"Programme concerté EDF-CEA sur le comportement des éléments combustibles défectueux ou rompus en régime normal" - Etat d'avancement en décembre 1976, Note DMG 118/76.

Titre Expérience de fonctionnement PWR	Pays : FRANCE Organisme directeur CEA
Titre (anglais) Working experience in PWR	Organisme exécuteur CEA/IPSN/DSN Responsable : J. GUIRLET
Date de démarrage : 1977 Date prévue d'achèvement : 1981 Etat actuel : en cours Dernière mise à jour : 30/04/77	Scientifiques : J. SEVEON

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.

Objectifs particuliers :

Détermination des sources radioactives 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.

Etat de l'étude :

1) Avancement à ce jour :

Des expérimentations ont été effectuées à la Centrale Nucléaire des Ardennes sur les circuits et les enceintes.

Un programme expérimental important a été mis au point pour le démarrage de FESSENHEIM.

.../...

2) Résultats essentiels :

Le premier résultat est la démonstration de la difficulté d'interprétation des valeurs expérimentales obtenues à la CNA due aux problèmes de mesure et à l'impossibilité d'obtenir des recoupements précis avec l'histoire du fonctionnement du réacteur.

Ces travaux effectués à la CNA doivent être considérés comme une mise au point des méthodes à utiliser pour les expérimentations futures. Elles ont cependant permis d'avoir une première idée du comportement de certains produits de fission, en particulier de l'Iode.

Prochaines étapes :

Etude du transfert de la contamination à FESSENHEIM.

Exploitation des premiers résultats.

Lancement d'expériences analogues à BUGEY 2

Classification: 5.2	
Title 1 (Original Language): Versuche zur Erfassung und Begrenzung der Freisetzung von Spalt- und Aktivierungsprodukten beim Coreschmelzen (PNS 4243 - I.1.5, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: ORGANIZATION: GfK/PNS
Title 2 (English): Experiments on Determination and Limitation of Fission and Activation Product Release During Core Meltdown	Project Leader: Dr. H. Albrecht DI. D. Perinic
Initiated (Date): 1972 Status: Continuing	Completed (Date): 1978 Last Updating (Date): December 1976

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-fraction 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 very precisely in the tests with inactive Corium (e.g. Ni, 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. Forty melt experiments have been carried out with variations in atmosphere (air, steam, argon and Ar + H_2), temperature (1700... 3000 $^\circ$C), pressure (0,8...2,2 bar), and melt composition.

5.2. Preparations have been made for continuous measurement of radioactive products from activated Corium or Fissium during the release tests.

5.3. The Fissium production facility (FIFA) has been completed. After technical acceptance of the facility, first test runs are being conducted.

In preparation for experiments with larger melt masses, a new high frequency generator with higher power than currently available (40 KVA) was ordered.

6. Results

6.1. In melting tests of 30 g Corium in air, a maximum release of 1,2 wt% was found at 2850 $\pm$ 100 $^\circ$C. This value corresponds to the total release of all main components of the melt determined by activation analysis. The most volatile elements were found to be Mn and Sb with relative release values of 11-17 % at 2850 $^\circ$C. The remaining steel components (Fe, Cr, Ni, Co) exhibited a moderate volatility with release values between 1 and 5 %, while the maximum release of U and Zr was not higher than 1 %. In the temperature region 2100-2900 $^\circ$C, all elements showed an increase of release of about two orders of magnitude. The variation of the air pressure did not lead to a measurable difference in release.

In melting tests in Ar and Ar + 5 % H_2, however, an unusual, pressure dependent behavior was discovered in the temperature region 2600-2800°C. The melt rose up in the crucible and there was a sudden initiation of bubbling and splattering followed by a large increase in the amount of released material (about one order of magnitude). The cause of this behavior appears to be boiling of the steel components. It was found that the temperatures (as a function of pressure) of these boiling points lie between and parallel to the boiling point curves of Fe, Cr and Ni.
In steam, on the other hand, the release seems to be much lower than in air and Ar up to temperatures of 3000°C. Quantitative values for the individual elements are not yet available.
Metallurgical examination of the crucibles and the melts have shown that during heatup in air to 1900°C at a rate of about 50°C/min the Zircaloy is completely oxidized. The steel remains mostly unoxidized, even with further heating to 2900°C so that coupling of the specimen to the induction field is still possible in this temperature region.
The attack of the ThO_2 crucible is moderate in air and Ar up to 2600°C but increases strongly when the steel components start boiling. At 3000°C the crucible wall is almost dissolved within a few minutes. In steam, however, where no boiling was observed, the crucible attack at 3000°C was much less severe.

7. Next steps

The most important work to be done in the near future will be
- conduction of additional release tests in steam
- release experiments with specimens containing activated material (steel, Zircaloy or Fissium)
- preparations for installation of the larger high frequency generator.

8. Relation with other Projects

PNS 4244: Constitution and Reaction Behavior of LWR-Materials at Core Melting Conditions.

9. References

Report KFK-2262 (1976) p. 89-91 (in English)
p. 347 - 365 (in German)

Report KFK-2375 (1976) p. 100-102 (in English)
p. 404-413 (in German)
Report IRS-F-28 (1976) p. 179-181 (in German)
Report IRS-F-31 (1976) p. 165-167 (in German)
Report IRS-F-29 (1976) p. 267-270 (in English)

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

Titre	Pays : FRANCE
Accidents PWR. Transfert de la radioactivité à l'intérieur de la centrale, et niveau et cinétique des rejets hors confinement	Organisme directeur : CEA/DSN
Titre (anglais) Reactivity transfert in the plant and release out containment.	Organisme exécuteur : CEA/DSN-SETSSR Responsable : J.GUIRLET (SETSSR)
Date de démarrage : 1/1/76 Date prévue d'achèvement : 1/12/79 Etat actuel : en cours Dernière mise à jour : 15/4/77	Scientifiques : A.MANESSE L.ROUSSEAU

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.

Objectifs particuliers :

Elaboration d'un catalogue d'accidents types.
Mise au point d'un faisceau de données nécessaire au calcul des effets radiologiques pour chaque accident type.
Etude des dispositions de secours actuelles (systèmes de sauvegarde), examen des choix et de leur aptitude à réduire les conséquences de l'accident.
Etude des éventuelles dispositions nouvelles de secours, comparaison avec l'étranger.
Etudes sur les mécanismes de transfert de la contamination radioactive à l'intérieur des différents milieux de la centrale et sur les spectres, les niveaux et les cinétiques des rejets pour chaque accident considéré.

.../...

Etat de l'étude :

1) Avancement à ce jour :

Etude de quelques cas accidentels présentés dans le rapport RASMUSSEN (WASH 1400).
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.

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)

Relation avec d'autres études :

Application du code ALICE sur les transferts de radioactivité, rejets et conséquences des accidents
Expérience PIREE, étude des problèmes de rétention de la radioactivité par l'eau d'un piscine.
Programme expérimental BOUFFON, expérience FLASH crayon rompu en séquence LOCA
Programme expérimental PHEBUS
Accident de dépressurisation des réacteurs à eau sous pression.

Documents de référence :

WASH 1400 (NUREG 75/014).

143-12-20/4165-40

Classification : 5-2 715

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Titre	Pays :
Migration de la contamination dans les piles à eau ordinaire en situation accidentelle.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Migration of the contamination in PWR in accident condition.	Organisme exécuteur : CEA/DSN-SESTR (Cadarache)
	Responsable : J.PORCHERON (SESTR)
Date de démarrage : 01/01/73 Date prévue d'achèvement :31/12/81 Etat actuel : Etude en cours Dernière mise à jour : 22/11/76	Scientifiques : G.BENEZECH J.MIRIBEL

Objectif général :

Etude des conséquences des accidents de manutention de combustible dans les piles à eau.
Etudes des rejets de contamination en cas d'accident de dépressurisation (participation au programme PHEBUS). Transfert des Iodes entre phase gazeuse et liquide dans les réacteurs PWR (expérience RCVE).

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.
Participation 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 RCVE : Etude des formes d'iode dans les phases liquides et gazeuses des circuits d'une pile à eau sous pression.

Installations expérimentales 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.

.../...

Etat de l'étude :

1) Avancement à ce jour :

Expérience PIREE-MANUTENTION : Etude de la rétention de l'Iode par l'eau dans le cas d'une injection de 40 à 80 bars. Expérience REGARDE : rétention de Krypton par l'eau en injection sous pression et efficacité du RCV d'un PWR.
PHEBUS : Construction du dispositif.
RCVE : idem.

2) Résultats essentiels :

Expérience PIREE-MANUTENTION : Efficacité de décontamination de l'eau vis à vis de I_2. Expérience REGARDE : 10 expériences réalisées.
PHEBUS : Mise au point de la mesure et hydrogène réalisée. Etude théorique réalisée.

Prochaines étapes :

Etude rétention du Xènon, influence du piegeage de l'Iode dans l'espace gaine combustible sur l'émission. Montage des appareils. Construction de l'installation.

Documents de référence :

"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, Congrés de la SFRP, Versailles 1974.
"Mesure des produits de fission sur la boucle Phebus", G.MANENT, J.PORCHERON - Rapport SESTR 75/01
"Etude des transferts de l'Iode en phase gazeuse dans les réacteurs à eau légère", JM.VINSON, G.MINGUELLA - Note SESTR/24, 1976.

Classification 5.2	
Title 1 Berekening van hoeveelheden radioactiviteit vrijkomend bij een ernstig reactorongeval	Country The Netherlands Organization KEMA
Title 2 Calculation of the quantities of radioactivity released as a result of a serious reactor accident	Physicist K.P. Termaat

1. General aim

In analysing the risk of a nuclear power plant for the surrounding population one has, amoung other things, to consider the quantity and the nuclide spectrum of the radioactivity released to the environment as a result of a reactor accident with a non zero probability.

2. Particular objectives

Calculations are performed with the programme CORRAL developed by Battelle NW USA. Minor changes were introduced.
The basic objective has been to contribute in the risk analysis referring to an enlarging nuclear power programme in this country.

3. Experimental facilities

Not applicable.

4. Project status

Necessary calculations have been performed. An improvement in applied input data to the programme CORRAL could be usefull.

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5. Next steps

Not applicable.

6. Relation with other projects

Equivalent calculations are performed in the USA as a contribution to the Rasmussen-study (WASH-1400).

7. Reference documents

See 2 and 6.

8. Degree of availability

Internal report.

Classification: 5.3	
Title 1 (Original Language): Untersuchungen zur Wechselwirkung von Spaltprodukten und Aerosolen in LWR-Containments (PNS 4311 - I.1.4, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** **ORGANIZATION:** PNS/GfK
Title 2 (English): Radioactive Pollutants in the Post Accident Atmosphere of the LWR-Containment	**Project Leader:** Dr. W. Schöck
Initiated (Date): January 1972 **Status:** Continuing **Completed (Date):** December 1977 **Last Updating (Date):** December 1976	

1. General Aim

The deposition of radioactive aerosols inside the containment building between formation and release to the environment has been recognized as a means of mitigating the radiological consequences of hypothetical accidents. To be able to describe and assess the various attenuation and removal mechanisms quantitatively will improve the validity of calculations in safety analyses.

2. Particular Objectives

The objective of the project is to describe the removal of airborne particulate radioactivity from the post accident atmosphere of an LWR-containment. For this purpose a computer code NAUA will be developed on the basis of an experimentally verified numerical model.

3. Research Program

3.1 Development of the theoretical NAUA-model to describe the aerosol behavior in the post accident atmosphere of an LWR-containment.

3.2 Specification and construction of the experimental facility.

3.3 Experimental measurement of necessary input data for the model and subsequent model development.

3.4 Verification of the model and extrapolation to real containment systems.

4. Experimental Facilities, Computer Codes

The NAUA-facility comprises a 3 m^3 thermostated vessel with an operating temperature range from 20 to 150°C in a saturated steam atmosphere. Steam condensation is initiated by adiabatic expansion of the volume. Peripheral instrumentation includes aerosol sources, steam generation and all the necessary particle measurement devices.

The computer code NAUA (current version Mod2) calculates the aerosol removal processes coagulation, sedimentation, thermophoresis and diffusion as well as steam condensation and the significant thermodynamic functions.

5. Progress to Date

Version Mod1 of the NAUA-code has been completed. On the basis of a sensitivity study (Refs. 1, 2, 3) with Mod1 the experimental facility was constructed and ordered.

6. Results

The results of the sensitivity study showed the strong influence of the particle form parameters on the aerosol decay. The feasibility of the planned experiments has been demonstrated. The expected enhancement of aerosol removal by the operation of spray systems was confirmed. The experimental facility was ordered. The main components of the instrumentation are operating.

7. Next Steps

The first experimental series on steam condensation on the aerosol particles will be started. The results will be integrated into the NAUA-model.

8. Relation with other Projects

PNS 4243

9. References

1. PNS semi annual report 1/1976, KFK 2375 (1976) (German)
2. G. Haury, W. Schöck, Modell zum Aerosolverhalten im Containment eines LWR nach einem schweren hypothetischen Störfall, KTG-Fachtagung Spaltproduktfreisetzung, 1.-2.6.1976, Karlsruhe (German)
3. G. Haury, W. Schöck, The Removal of Radioactive Aerosols from the Post Accident Atmosphere of an LWR-Containment, 14th ERDA Air Cleaning Conference, 2.-4.8.1976, Sun Valley, Idaho (English)

10. Availability of Reports

Unrestricted distribution

Classification: 5.3	
Title 1 (Original Language): Krypton- und Xenon-Entfernung aus der Abluft kerntechnischer Anlagen (PNS 4140 - II.6.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: ORGANIZATION: PNS/GfK
Title 2 (English): Separation of Krypton and Xenon from the Offgas of Nuclear Facilities	Project Leader: R. v. Ammon E. Hutter R.D. Penzhorn
Initiated (Date): January 1972 Status: Continuing Completed (Date): Last Updating (Date): December 1976	

1. General Aim

For environmental protection the gaseous radioactive fission product Kr-85 is to be separated from the offgas of nuclear facilities, especially of reprocessing plants. It is, furtheron, to be conditioned for final storage.

2. Particular Objectives

The feasibility of the rare gas separation from a gas mixture is to be demonstrated with the low temperature distillation process.

3. Research Program

3.1 Column behavior and separation factors achievable during the low temperature distillation of the three component system N_2-Xe-Kr.

3.2 Prepurification of the offgas.

3.2.1 Testing of catalysts and development of a catalytic test bed including a controlled dilution loop for the reduction of O_2 and NO_x with H_2.

3.2.2 Measurement of the adsorptive capacity of some inorganic sorbent materials like molecular sieves for Kr and Xe and trace gases e.g. H_2O, CO_2, NO_x and NH_3.

3.3 The adsorptive separation of Kr and Xe on activated charcoal is studied as an alternative or auxiliary method to the distillation process.

4. Experimental Facilities, Computer Codes

Ad 3.1 Pilot Plant KRETA (feed gas flow: 50 Nm^3/h)

Ad 3.2.1 Two laboratory scale installations for testing catalysts: a static and a dynamic type (throughput max. 1 Nm^3/h); a test-loop (throughput 10 Nm^3/h) for the development of H_2-control and H_2-safety studies.

Ad 3.2.2 and 3.3 A dynamic laboratory scale apparatus for the measurement of adsorptive capacities of various sorbents.

5. Progress to Date

Ad 3.1 The pilot plant KRETA has been put into operation and first runs have been conducted. Computer programs were developed for the calculation of the influence of parameter variations on column performance. Calculations were performed for the simulation of the steady state of the column using the five-component system N_2-Ar-O_2-Kr-Xe and for the simulation of its time dependence.

Ad 3.2.1 The laboratory tests on the catalytic reduction of O_2 and NO_2 with H_2 using the small dynamic testapparatus were completed. The last experiments were concerned with the poisoning of selected Ru catalysts by iodine and tributylphosphate (TBP). The small static test-apparatus for kinetic measurements has been put into operation as well as the recycling loop. The analytic determination of H_2 in the presence of NO_x has been developed. A large catalytic recycle loop (throughput 50 Nm^3/h) has been opened for bids. The reduction of O_2 and NO_x was investigated without catalyst (thermal reduction) in two series of experiments: using flame combustion and electrical heating of the reaction zone. These experiments were carried out in collaboration with Fa. Decatox.

Ad 3.2.2 Dynamic adsorption coefficients of NO, NO_2, NH_3 and NH_3/H_2O were determined using various molecular sieves.

Ad 3.3 An experimental program initiated by us on the separation of Kr-Xe by adsorption on activated charcoal was concluded by Bergbauforschung Essen.

6. Results

Ad 3.1 A static and a dynamic computer code were developed for computation the column behavior. Now the influence of Kr-85-heat can be represented.

The calculation of the concentration profiles of the gas components to be expected in the first column indicates a decontamination factor (DF) of Kr/Xe from the carrier gas N_2 of $>10^4$. These data obtained for the steady state were verified by the results of the statistical mode of calculation. During the first period of operation of the KRETA plant some of these values were also confirmed experimentally: In the two-component systems N_2-Ar a DF (Ar) = 10 was obtained, and in the system N_2-Kr a DF (Kr) $>10^3$. In the second column the separation Kr-Xe resulted in a Kr head product of 99,99 % purity and a Xe bottom product containing <100 ppm Kr.

Ad 3.2.1 Laboratory studies of the poisoning effect of J_2 and TBP on Ru catalysts used for the reduction of O_2 and NO with H_2 showed that the activity of these catalysts remains almost unchanged. However, their specificity towards the formation of N_2 decreases substantially: an increasing coverage of the catalyst with HJ (formed by catalytic reduction of J_2) or TBP results in an increasing formation of NH_3.
High-temperature tests carried out up to 1000°C resulted in a decrease of the catalytic activity causing the temperature at which the reaction starts to rise by about 100°C.
An experimental study of the thermal reduction of O_2 and NO_x with H_2 (without catalyst) showed that only at temperatures ≥ 1100°C reaction rates are short enough to provide high reaction yields.

Ad 3.2.2 The adsorption of NO on molecular sieves is too weak to provide an efficient separation of this gas component from offgas streams. NH_3 is adsorbed strongly, but the adsorption coefficients are only half as high on acid resistant sieves as on the conventional 5A type. The presence of H_2O in the gas stream increases the NH_3 adsorption. The desorption of NO_2 from molecular sieves is strongly temperature dependant and proceeds in several steps.

Ad 3.3 The separation Kr-Xe by means of selective desorption from activated charcoal (vacuum and purging with N_2 or H_2O-vapor) is feasible, but relatively large adsorption beds are required for large process gas streams.

7. Next Steps

Ad 3.1 The operation of the KRETA plant will be continued using the three-component system N_2-Xe-Kr. The molecular sieve unit ADAMO which was ordered at the end of 1975 will be mounted in the beginning of 1977. It will be operated at first separately and then simultaneously with KRETA.

Ad 3.2.1 Catalyst tests will be continued on laboratory scale. Of prime concern will be safety studies in connection with the control of large amounts of H_2. The semi-scale equipment for the catalytic O_2-, NO_x-reduction will be ordered.

Ad 3.2.2 and 3.3 The dynamic adsorption studies will be continued using the rare gases and various other trace gases as well as various adsorbents.

8. Relation with Other Projects

Within GfK: LAF II (Offgas filters in reprocessing plants); IRCH (Analysis and decomposition of O_3 in liquid gas mixtures).
External relations: KFA/ICT Jülich (Cryogenic Kr-separation); CEN Mol (Cryogenic Kr-separation); US ERDA (Separation and storage of Kr); Bergbauforschung Essen (Adsorptive separation of Kr/Xe); Fraunhofer-Institut für die Chemie der Treib- und Explosivstoffe (H_2-safety); Fa. Decatox (thermal reduction of NO_x).

9. References

R. v. Ammon et al., Reaktortagung Düsseldorf 1976;
PNS 1st semiannual report 1976.

10. Degree of Availability of the Reports

GfK/LA

Classification: 5.3	
Title 1 (Original Language): Spalt-Jod-Abscheidung in Kernkraftwerken und Wiederaufarbeitungsanlagen: Störfall-Umluftfilter zur Abscheidung von Spaltprodukten aus der Sicherheitsbehälter-Atmosphäre (PNS 4110/4111 - II.6.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Fission Product Iodine Removal in Nuclear Power Plants and Reprocessing Plants: Post Accident Recirculation Air Cleanup for Fission Product Removal from the Containment Atmosphere	Project Leader: J.G. Wilhelm H.G. Dillmann
Initiated (Date): 1971 Status: continuing	Completed (Date): 1976 Last Updating (Date): December 1975

1. General aim

Fission product removal from the containment atmosphere by post-accident recirculating air filter systems.

2. Particular objectives

Removal of fission product iodine and aerosols.

3. Experimental facilities and research program

3.1. Technical test rig for testing of original filter components (iodine and aerosols) under simulated accident conditions.

Gas flow : up to 2000 m^3h^{-1}

temperature: up to 200°C

r. h. : up to 100 % at temperatures $\leq$ 151°C

pressure : up to 5 bar

Laboratory scale rigs for testing of sorption materials.

3.2. Testing of sorption materials in annular filter beds for operation under steam atmosphere under pressure and at high temperature.

Examination of aerosol filters under simulated accident conditions. Testing of single technical components e. g. moisture separators, heaters, prefilters under simulated accident conditions.

Irradiation tests on fiber mats and aerosol filters, respectively.

Testing of complete filter systems under simulated accident conditions and, if necessary, improvement of the systems (in cooperation with other partners).

Experimental investigations concerning Ru, Cs, Te filtration from the containment atmosphere.

4. Project status

4.1. Progress to date

A prototype post-accident recirculation filter has been tested under various conditions in the filter test rig.

The adsorber material AC 6120 as well as molecular sieves were examined for transportation of the impregnation by the air-steam mixture and for their stability, respectively, under simulated accident conditions. A fundamental concept was elaborated for an aerosol generator working under simulated accident conditions. Investigations have been made to find noncorrosive test aerosols.

4.2. Essential results

The prototype post-accident recirculation filter was exposed to hot air of 160^{o}C and to steam up to 1 bar at temperatures between 108^{o} and 160^{o}C. After 18 days of operation it yielded removal efficiencies of $\geq$ 99.98 % for $CH_3{}^{131}I$ (residence time 0.2 sec, Ø air flow 1200 m^3h^{-1}).

An iodine filter element was exposed for more than 10 days to an air-steam mixture of 160^{o}C and 72 % r. h. at a pressure of 4 bar. After that time a penetration of < 1 % for radioactive methyliodide was found (residence time 0.2 sec).

The laboratory experiments with AC 6120 performed in superheated steam over a period of 48 h yielded no measurable transportation of impregnation under different conditions up to temperatures of 200^{o}C.

5. Next steps

Experiments under simulated accident conditions will be continued with higher pressure, relative humidity and temperature of the air-steam mixtures.

Investigations of the sorption materials are to be continued.

Tests of aerosol filters under simulated accident conditions will be carried out after installation of the aerosol generator in the technical filter test rig.

6. -

Classification: 5.3	
Title 1 (Original Language): Entwicklung von Störfall-Umluftfiltern für Reaktorsicherheitsbehälter (PNS 4111 - II.6.1, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** **ORGANIZATION:** PNS/GfK/LAF II
Title 2 (English): Post Accident Recirculation Air Cleanup for Fission Product Removal from the Containment Atmosphere	**Project Leader:** H.-G. Dillmann
Initiated (Date): 1971 **Status:** Work in Progress **Completed (Date):** **Last Updating (Date):** Dec. 1976	

1. General Aim

Fission product removal from the conainment 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 3.1 Tests on the long term performance (~ 100 h) of inorganic sorption materials in steam air atmosphere at elevated temperature, pressure, and humidity have been performed and finished.

to 3.2 Tests on the performance of an all steel droplet seperator are performed.

6. Results

to 3.1 Molecular sieve Linde 13 X Ag performed with high removal efficiency for $CH_3{}^{131}I$ under all tested conditions up to 85 % r. h., 300°C and a pressure of 5 bar.

In the long time tests AC 6120 was only useful in a lower temperatur range (< 250°C).

to 3.2 The specially constructed wave plate droplet seperator showed reasonable performance in the particle size range above 10 µm.

7. Next Steps

to 3.1 Experiments on loaded iodine sorption material will be performed with respect to desorption of iodine in a high irridiation field. This will be the last tests for the iodine sorption filters.

to 3.2 The measurements on droplet seperators will be continued.

8. Relation with Other Projects

Results of PNS 4114, part 2, may contribute to the program described here.

9. References

KFK 2375, p. 121-127.

10. Degree of Availibility of the Reports

unclassified, reports are available without restriction.

Classification: 5.3	
Title 1 (Original Language): Entwicklung von Abluftfiltern für Wiederaufarbeitungsanlagen (PNS 4112 - II.6.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: ORGANIZATION: GfK/LAF II
Title 2 (English): Development of Off-Gas Filters for Reprocessing Plants	Project Leader: J.G. Wilhelm, LAF Dr. Furrer, LAF K. Jannakos, RBT
Initiated (Date): July 1971 Status: Continuing Completed (Date): Last Updating (Date): December 1976	

1. General Aim

To remove fission product iodine and other contaminations from off-gas of fuel reprocessing plants a filtration process in the head end is under development capable to retain the fission product iodine, including ^{129}I, in a form ready for final storage.

2. Particular Objectives

Filters have been developed for decontamination of the off-gas from large reprocessing plants. The filter systems should be able to handle the majority of the fission product iodine and other contaminants from spent fuel elements, including contaminants in the form of aerosols. The loaded iodine adsorber material should be in a form ready for final storage of the ^{129}I without needing additional processing.

The final objective is to develop, build and test a prototype off-gas filter system retaining the aerosols and iodine from the off-gas of the reprocessing plant.

3. Research Program

3.1 Testing of iodine sorption material under the conditions prevailing in the off-gas of a reprocessing plant.

3.2 Evaluation of droplet seperator, demister, HEPA filter, and iodine sorption filter for a large reprocessing plant (single components and arrangement for the whole filter train).

3.3 Development of the remote handling system for the components inside of the filter cells (with respect to low contamination level, emergency handling and leaktight operation of filter units).

4. Experimental Facilities

to 3.1 Sampling stations and test rigs in the Karlsruhe pilot reprocessing plant (WAK).

to 3.2 Test rigs (no radioactivity above tracer level) simulating head end off gas conditions for dummy components (HAUCH, 50 m^3/h of volumetric gas flow, PASSAT, 250 m^3/h). PASSAT will include remote handling equipments and is designed for the testing of 1 : 1 filter components for the large reprocessing plant(throughput up to 1500 t of heavy metal per year).

5. Progress to Date

to 3.1 Various tests have been performed with the AC 6120/H_1 iodine sorption material.

to 3.2 and 3.3 The test rig HAUCH is in operation.
Part of the filter equipment has been completed for PASSAT and tested for remote handling suitability. PASSAT has been designed and is under construction.

6. Results

to 3.1 The influence of high H_2O-vapour pressure and of small amounts of TBP on the loading capacity of AC 6120/H_1 is very low, AC 6120/H_1 can be used up to a consumption of 95 % of the Ag-impregnation by reaction with iodine.

to 3.2 First measurements on HAUCH showed relatively high concentrations of droplets with diameter < 16 µm in the simulated dissolver off-gas. The sources could be detected and eliminated.

to 3.3 A prototype iodine filter drum is mechanically tested, the concept used is selected for the final version of the iodine filter.

7. Next Steps

After startup of the PASSAT rig it is scheduled to test the filtration components (droplet separators, HEPA-filters, and iodine-filters) under the conditions of the head end off-gas in a 1500 t/y reprocessing plant. Measurements will be made of the off-gas composition and the aerosol- and iodine-concentrations during fuel dissolution in the Karlsruhe reprocessing plant.

8. Relation with Other Projects

The first filtration step will be the droplet separation and the second step will be HEPA-filtration (PWA 5152). The iodine removal from the dissolver off-gas will be necessary for gas cleaning with respect to the ^{85}Kr-separation process (PNS 4140).

9. References

WILHELM, J.G.; FURRER, J.: Abscheidung von Spaltjod aus dem Auflöserabgas einer großen Wiederaufarbeitungsanlage an festem Sorptionsmaterial. Reaktortagung, Düsseldorf, 30. 3. - 2. 4. 1976, Deutsches Atomforum e.V. Leopoldshafen 1976: ZAED. S. 347 - 50.

WILHELM, J.G.: Beiträge in: Waste Management Research Abstracts, No. 10 (1975).

WILHELM, J.G.; FURRER, J.; SCHULTES, E.: Head end iodine removal from a reprocessing plant with a solid sorbent. 14. Air Cleaning Conference, Sun Valley, Idaho, August 2-4, 1976.

FURRER, J.; KAEMPFFER, R.: Untersuchung der Reaktionen von organischen Jodverbindungen mit feinst auf amoprpher Kieselsäure verteiltem Silbernitrat. Monatshefte für Chemie, 107(1976), S. 933-38.

FURRER, J.; JANNAKOS, K.; WILHELM, J.G.; APENBERG, W.; KAEMPFFER, R.; LANGE, W.; MENDEL, W.; POTGETER, G.; ZABEL, G.; PFAUTER, C.; POETSCH, G.: Entwicklung von Abluftfiltern für Wiederaufarbeitungsanlagen. In: Projekt Nukleare Sicherheit. 2. Halbjahresbericht 1975. KFK 2262 (Juni 1976), 113-22.

10. The reports above are available without restriction.

Classification: 5.3	
Title 1 (Original Language): Bestimmung von Jodkomponenten in der Abluft von kerntechnischen Anlagen (RS 221 (PNS 4114.1) - II.6.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT/GfK/PNS ORGANIZATION: PNS/GfK/IAF II
Title 2 (English): Determination of Radioiodine Species in the Exhaust Air of Nuclear Installations	Project Leader: Dr. Deuber
Initiated (Date): 1973 Completed (Date): Status: Continuing Last Updating (Date): Dec. 1976	

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 inorganic iodine (I_2), particulate iodine, and organic iodine.
(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 or improvement of selective sorption materials for radioiodine species samplers.

3.2 In situ tests: Operation of radioiodine species samplers in the exhaust air.

4. Experimental Facilities

to 3.1 Apparatuses 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.

5. Progress to Date

to 3.1 Determination of the removal efficiencies of the I_2 sorption materials DSM 10 and DSM 11 for $^{131}I_2$ and $CH_3{}^{131}I$. Determination of the removal efficiencies of particle filters for $^{131}I_2$.

to 3.2 Operation of radioiodine samplers in different exhaust air ducts in 3 nukes (BWR, PWR) and 1 research reactor.

6. Results

to 3.1 No difference in the removal efficiencies of DSM 10 and DSM 11 for $^{131}I_2$ and $CH_3{}^{131}I$. No desorption of $^{131}I_2$ from DSM 10 under extreme conditions.
Small removal efficiency of the particle filter GF/A for $^{131}I_2$.

to 3.2 Radioiodine species samplers with DSM 10 or DSM 11 generally suitable for the classification of radioiodine species.
(DSM 10 not suitable in the presence of oxidants.) No desorption of $^{131}I_2$ from DSM 10 during 7 days.
Percentage of particulate iodine generally very small ($\leq$ 3 %).
Variation of the percentage of inorganic iodine between 0 and 91 %.
Percentage of inorganic iodine in the effluent of iodine filters very small ($\leq$ 2 %).

7. Next Steps

to 3.1 Improvement of the radioiodine species sampler for nukes (inclusion of further radioiodine species).
Development of a radioiodine species sampler for reprocessing plants.

to 3.2 Continuous measurements of the radioiodine species in different exhaust air ducts of a PWR nuke during a representative period (ca. 1 year).

8. Relation with Other Projects

Ageing and Poisoning of Iodine Sorption Materials (PNS 4114, part 1).

9. References

KFK 2262, p. 126; KFK 2375, p. 147.

10. Degree of Availability of the Reports

Literature department of GfK, unclassified.

Classification: 5.3	
Title 1 (Original Language): Alterung und Vergiftung von Jod-Sorptionsmaterial (PNS 4114.2 - II.6.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: ORGANIZATION: PNS/GfK/LAF II
Title 2 (English): Ageing and Poisoning of Iodine Sorption materials	Project Leader: Dr. Furrer
Initiated (Date): 1973 Status: Continuing Completed (Date): Last Updating (Date): Dec. 1976	

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 Gaschromatographic 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 Gaschromatographic 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 Impregnated charcoal was poisoned by operation in exhaust air from shut-off rooms of a PWR. Some of the adsorbed poisons were desorbed in the laboratory by an optimized desorption process.

to 3.2 A conception was developed for an improved iodine sorption filter.

6. Results

to 3.1 The filter poisons with low vapour pressure could be removed by a prefilter. The iodine sorption material of a filter bed in the effluent air of the prefilter was poisoned by highly volatile solvents which penetrated the prefilter. These solvents could be removed with a desorption process. The ^{131}I removal efficiency of the iodine sorption material could nearly be restored by the desorption process.

to 3.2 The prototype of an improved iodine sorption filter is under construction.

7. Next Steps

Further developments for an in-place regeneration process by desorption of the highly volatile filter poisons.

to 3.1 Testing of the behaviour of the already removed iodine during the desorption step.

to 3.2 Testing of the prototype filter.

8. Relation with Other Projects

Results of PNS 4114, part 2, may contribute to the program described here.

9. References

BITTER, K.; FURRER, J.; KAEMPFFER, R.: Alterung und Vergiftung von Jod-Sorptionsmaterialien. In: Projekt Nukleare Sicherheit. 2. Halbjahresbericht 1975. KFK 2262 (Juni 76), S. 123-26.

FURRER, J.; KAEMPFFER, R.; WILHELM, J.G.: Alterung und Vergiftung von Jod-Sorptionsmaterialien in Kernkraftwerken. Ageing and poisoning of iodine filters in nuclear power plants. Kerntechnik, 18 (1976), S. 313-17.

WILHELM, J.G.: Jodfilter in Kernkraftwerken. Kommission der Europäischen Gemeinschaften, V/1531/76-D (1976).

10. Reports and publications are available without restriction.

Classification: 5.3	
Title 1 (Original Language): Dosisabbau (RS 204 - II.6.1, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** KWU, Erlangen
Title 2 (English): Dose Reduction	**Project Leader:** Dr. Wille
Initiated (Date): 1. 4. 76 **Status:** Continuing **Completed (Date):** 30. 9. 78 **Last Updating (Date):** 31. 12. 76	

1. General Aim

Improvement of the currently used units and the development of advanced techniques to handle gaseous and liquid radioactive waste, its optimization and the application of experience both in the labs and operating plants 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 reactor shutdown will be reduced as much as possible by permanent 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, and to retain Tritium in appropriate systems.

2.3 Decontamination

Decontamination methods will be further developed to the point of applications 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, selection of the transportation container for noble gases, examination of oxygen- and hydrogen measuring apparatus and selction 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 or emergency systems); investigations of release mechanisms and Tritium extraction methods.

3.3 Decontamination of Units and Containers

Decontamination of large-scale containers 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

During this reporting period several endurance tests have been performed in the test facility, such as long-term operating test for 500 hrs. In addition, hydrogen tests have been conducted for the first time.

During the hydrogen-nitrogen separating tests it was apparent that one of the control valves needs to be improved in order to guarantee an automatic continuous operation.

6. Results

All control circuits of the test facility operate satisfactory except the hydrogen-nitrogen separation test, i.e. an uncontrolled continuous operation without adjustment is assured. Hydrogen-nitrogen separation was manually operated, whereby a 80 - 95 % hydrogen purity could be obtained.

7. Next Steps

After the hydrogen-nitrogen separating facility has been modified and improved and a diaphragm control unit for the two compressors installed, further passive continuous tests are planned to be conducted using noble gas additions.

8. Relation to Other Projects

Preparatory work has been performed within R & D Task RB 741/S 32.

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9. <u>References</u>

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

Classification: 5.3	
Title 1 (Original Language): Aktivierte Korrosionsprodukte in LWR-Kreisläufen (RS 209 - II.6.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Activated Corrosion Products in LWR Loops	Project Leader: Dr. Neeb
Initiated (Date): 1. 1. 76 — Completed (Date): 31. 12. 78 Status: Continuing — Last Updating (Date): 31. 12. 76	

1. General Aim

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. Particular Objectives

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. Research Program

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 labs.

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

Activity concentrations of some corrosion product-radionuclides in the primary coolant will be routinely measured in the NPP labs. For evaluation of specific relevance of these operating data, a comparative study of the ^{58}Co- and ^{60}Co-activity concentrations in the primary coolant of the four full load operating PWR plants KWO, KKS, KCB, and KWB-A as initiated. Initial investigations covered the time history of the two radionuclide concentrations in the four plants. For plant comparison the four operating light water PWR plants KWO, KKS, KCB and KWB-A were selected. The main data characteristic of these considerations have been compiled. The individual analyses available ^{58}Co and ^{60}Co activity concentrations in the primary coolant have been used for the calculation of average values of ^{58}Co and ^{60}Co activity concentrations in the primary coolant for each operating cycle. In order to avoid irregularities based on calculated average values, the following precautionary measures have been taken:

Due to specific conditions of the start-up operation, no values were taken from the first 120 d of each 1. operating cycle (KWB-A the first 150d) for calculation of average values.

The same refers to the first 30d each of each subsequent cycle as well as for the first 30d following extensive shutdown periods. Also, values obtained from shutdown corrosion product peaks, identified by the power history of the reactor, have been eliminated.

6. Results

Compilation of the characteristic plant data showed that a direct comparison of the activity concentrations of corrosion products obtained from the reference plants can be made without computational corrections, since basic plant data, such as the relationship between surface area and primary coolant volume and the thermal and fast neutron flux density approximately the same with all plants. Differences can be observed with the S.G. material (Ni and Co-content), coolant conditioning and primary circuit pretreatment. With this limited amount of criteria, one would expect to recognize and identify coolant activity relationship unless they do not exist or they are of too complex a nature.

The following picture on the operating history of the reference plants can be drawn from individual values as well as from calculated average values:

The introduction of a so-called decontamination- and protection layer operation starting with Fe-measurement during hot test operation in IBS-KCB has led to a distinctly reduced primary side metal erosion rate, and to a strongly adhesive oxyde layer, both phenomena contrary to those observed in KKS. These positive effects of the contamination- and protection layer operation will not be reflected in the ^{58}Co- and ^{60}Co activity concentrations. To the contrary, the three plants furnished with Incoloy steam generators showed that the flush system used for start-up of KKS reveal distinctly the lowest activity values, whereas the two plants subjected to the contamination- and protection layer operation showed considerably high activity concentrations.

In all reference plant ^{58}Co- and ^{60}Co-concentration values have already been observed during the first half of the 1. operating cycle. These do not systematically increase during subsequent operation, and, apart from a few exceptions, remain the same over the total reference period, showing little or no deviations (KWO 7 cycles, KKS 4 cycles, KCB 2 cycles, KWB-A 1 cycle).

This is especially surprising since with ^{60}Co, having a half life of 5,27a, a continuous increase was expected for the initial years. Therefore a long-lasting constant ^{58}Co- and ^{60}Co-inventory can be expected in the primary coolant for the contamination layer build-up.

7. Next Steps

The initiated comparison of PWR plant parameters in connection with ^{58}Co and ^{60}Co activity concentrations will be continued. Similar tests have been started for BWRs also.

For preparation of the analysis of loop component deposits etc., suitable analyzing methods will be investigated in order to determine specific activities.

8. Relation with Other Projects

Preparatory work has been performed under the scope of R & D task RB 74 I/S 27.

9. References

Dr. W. Brandt:
Abgabe- und Anlagerungsmechanismen von Korrosionsprodukten in Reaktorprimärkreisläufen
Abschlußbericht zum Förderungsvorhaben BMFT RB 74 I/S 27
Teil I Kontaminationsmodell (Okt. 1975)
Teil II Metallabgaberaten (Dez. 1975)

10. Degree of Availability

The reports are classified Company Confidential.

Classification: 5.3

Title 1 (Original Language): Spalt-Jod-Abscheidung in Kernkraftwerken und Wiederaufbereitungsanlagen: Abluftfilter an Reaktoren, Alterung und Vergiftung von Jod-Sorptionsmaterialien (PNS 4110/4114 - II.6.1., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Fission Product Iodine Removal in Nuclear Power Plants and Reprocessing Plants: Exhaust Air Filters at Nuclear Installations, Aging and Poisoning of Iodine Filters	Project Leader: J.G. Wilhelm H. Deuber
Initiated (Date): July 1971 Status: continuing	Completed (Date): 1976 Last Updating (Date): December 1975

1. General aim

Improvement in the operational behaviour and removal efficiency of exhaust air filters.

2. Particular objectives

Provisions to control the effects of solvent loading on the removal efficiency of iodine filters.

3. Experimental facilities and research program

3.1. Laboratory scale test rigs for the measurement of removal efficiencies of solvent loaded iodine sorption materials.
Test rigs in nuclear power plants.

3.2. Determination of poisoning compounds for iodine filters.
Measurement of retention times of poisoning compounds on iodine sorption materials.
Design of iodine filter assemblies with sufficient operational time.

4. Project status

4.1. Progress to date.

The influence of solvent loading on the removal efficiency of iodine filter charcoal was examined for the exhaust air filter in the shutoff rooms of an LWR nuclear power station. The first laboratory studies for the determination of the retention times of poisoning compounds were carried out.

4.2. Essential results

The loading of solvent in the impregnated charcoal of an exhaust air filter varied between 7.3 and 12.4 wt %. The removal efficiency for ^{131}I in form of CH_3I was substantially reduced by adsorption. The measurements with upstream activated filter charcoals in a nuclear power station showed that the solvent components xylene and toluene occurring in large quantities under the usual conditions of shutoff room exhaust air filtering cannot be retained sufficiently long. The high-boiling components of the petroleum-gasoline fraction, which also occur in the exhaust air and substantially contribute to iodine sorption filter loading through pollutants, were retained over a period of several months by relatively low-depth upstream filter beds.

5. Next steps

The measurement of retention times under controlled conditions of coadsorption of solvents and water will be continued, also prefilters will be tested in nuclear power plants.

Identification of Radioiodine Species in the Off-Gases of Nuclear Installations

1. General aim

Improvement in the assessment of the environmental impact of radioiodine in the off-gases of nuclear installations.

2. Particular objectives

Identification of the radioiodine species.

3. Experimental facilities and research program

3.1. Laboratory scale rigs for the generation of iodine species and testing of sorption materials.

3.2. Development of iodine samplers for the discrimination of elemental iodine, methyl iodide and possibly other iodine species as hypoiodous acid; application of discriminating iodine samplers in off-gases of nuclear installations.

4. Project status

4.1. Progress to date.

A discriminating iodine sampler for the application in off-gases of nuclear power stations is being developed. Different materials eligible as selective sorption materials for elemental iodine are tested with respect to their removal efficiencies for elemental iodine and methyl iodide. - Experiments on the volatilization and removal of hypoiodous acid have been made.

4.2. Essential results.

AC 6111, impregnated with potassium iodide, proved to be the most suitable of the investigated selective sorption materials for elemental iodine under certain conditions. When tested in the entire range of parameters, which is of significance for the examination of off-gases from nuclear power stations, it caused a negligible removal of methyl iodide. - An iodine species was volatilized from solutions of carrier-free iodide and elemental iodine which was difficult to remove with AC 6120. This species did not exhibit the properties expected for hypoiodous acid.

5. Next steps

The tests of AC 6111, impregnated with potassium iodide, will be continued with elemental iodine in the significant range of parameters. Measurements in the off-gases of nuclear power stations will be made. The development of a discriminating iodine sampler applicable in off-gases of reprocessing plants will be started.

7. Reference documents

Report KFK 2130 (1975) p. 96 (german with english abstracts)
Report KFK 2195 (1975) p. 113 (german with english abstracts)

8. Degree of availability

Unrestricted distribution

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

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
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.

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Title 1 (Original language)	Classification
Trattamento dei gas nobili radioattivi prodotti per fissione.	5.3

3. CURZIO G., GENTILI A.
Determinazione della densità granulare di materiali ad elevata porosità specifica. Il Giornale di Fisica, XIII, 4, 286, 1972.

4. CURZIO G., GENTILI A.
Noble Gas Adsorption Characteristics of Charcoal Bed: Van Deemter's Coefficient Evaluation. An. Chem. 44, 8, 1944 (1972).

5. CURZIO G. GENTILI A.
Libération des gas nobles par centres nucléaires: quelques remarques sur le fonctionement des filtres de charbon de bois. VI[e] Congrès International de la Société Française de Radioprotection: "Tendances Nouvelles en Radioprotection". Bordeaux, 27-30 mars 1972, p. 233, Montrouge 1972.

6. CURZIO G., GENTILI A.
The Effects of Decay Heat on Adsorption Characteristics of Charcoal Beds. Noble Gases Symposium, Las Vegas, 24-28 Sept. 1973.

7. CASTELLANI F., CURZIO G., GENTILI A.
Effects of Moisture on Krypton Adsorption Characteristics of Charcoal Beds. Kerntechnik, 17 (1975), n. 11, 486.

8. CURZIO G., GENTILI A.
Man-Rem Cost: a Reverse Evaluation. Intern. J. Environmental Studies, 1975, vol. 7, 287-288.

9. CASTELLANI F., CURZIO G., GENTILI A.
Krypton Diffusion in Granular Charcoal. An. Chem. 48, 3, 599-600 (1976).

10. CASTELLANI F., CURZIO G., GENTILI A.
Un problema di conduzione del calore in un mezzo cilindrico omogeneo con sorgente in movimento. To be published in "Termotecnica".

11. CASTELLANI F., CURZIO G., GENTILI A.
Facilities for Conditioning Radioactive Noble Gases Produced by Nuclear Power Reactors in West Europe. Kerntechnik 19(1), (1977).

12. CASTELLANI F., CURZIO G., GENTILI A.
Sistemi di trattamento dei gas nobili radioattivi prodotti per fissione negli impianti nucleari. RP 250(76).

13. CASTELLANI F., CURZIO G., GENTILI A.
Studio introduttivo di metodi di collaudo e controllo periodico di efficienza per un sistema di rilascio ritardato dei gas nobili radioattivi prodotti per fissione. RP 251(76).

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 radioattivi 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 PSIÇO 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).

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)	Classification
Il controllo dei sistemi filtranti installati negli impianti nucleari per la rimozione delle particelle e dello iodio.	5.3
Title 2 (English)	**Country** ITALY
Testing of the Eilter Systems used in Nuclear Plants for particle and iodine removal.	**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.

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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

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).

7. Reference documents

Report KFK 2130 (1975) p. 77 (german with english abstracts)

Report KFK 2195 (1975) p. 98 (german with english abstracts)

8. Degree of availability

Unrestricted distribution

Classification 5.3	
Title 1 GAS PHASE TRAPPING STUDIES (1)	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 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	COUNTRY UNITED KINGDOM
GAS PHASE TRAPPING STUDIES (2)	SPONSOR UKAEA
	ORGANIZATION RDL WINDSCALE
Title 2	Project Leader J J HILLARY
Initiated 1972 Completed :	Scientists:
Status : Last updating	

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.

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Classification: 5.4	
Title 1 (Original Language): Untersuchungen zur ^{129}J-Radioökologie (PNS 4132 - II.6.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: ORGANIZATION: PNS, GfK Karlsruhe
Title 2 (English): Investigations into the ^{129}I Radioecology	**Project Leader:** Dipl.-Ing. Schüttelkopf
Initiated (Date): 1974 **Status:** Continuing **Completed (Date):** 1978 **Last Updating (Date):** Dec. 1976	

1. General Aim

Determination of the Long-term Exposure of the Environment of a Reprocessing Plant by Long-lived ^{129}I

2. Particular Objectives

3. Research Program

3.1 Measurement of ^{129}I emission from WAK

3.2 Measurement of ^{129}I concentrations in the vicinity of WAK

3.3 Measurement of ^{127}I in the vicinity of nuclear facilities

4. Experimental Facilities, Computer Codes

5. Progress to Date

5.1 ^{129}I was determined in effluent waters and process solutions.

5.2 ^{129}I was determined in milk, thyroid and soil samples.

5.3 ^{127}I was measured in the environmental air of the Karlsruhe Nuclear Research Center by a method developed in 1976.

6. Results

6.1 The concentration of ^{129}I decreases in all effluent waters investigated. Investigations were terminated in early October.

6.2 The ^{129}I content in goat's milk has slowly decreased since an iodine filter was installed in the WAK off-gas system. The ^{129}I content of soil samples scatters over one order or magnitude and attains around 1 fCi/g.

6.3 The method developed for ^{127}I evaluation allows to measure both ^{127}I aerosols and elemental ^{127}I. The detection limit is about

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1 ng $^{127}I/m^3$ of air. Results obtained so far range from 1 to 10 ng $^{127}I/m^3$ for elemental iodine as well as for aerosol iodine.

7. Next Steps

7.1 Work has been completed

7.2 The determination will be carried on of ^{129}I in soil and milk samples taken in the zones principally exposed to the WAK plume.

7.3 It is planned to perform ^{127}I measurements in Northern Germany in the environmental air of nuclear facilities.

8. Relation with Other Projects

9. References

H. Schüttelkopf
Die Radioökologie von ^{129}J
KFK-Nachrichten No.4, 1975

H. Schüttelkopf
Zur ^{129}J-Emission aus einer Wiederaufarbeitungsanlage für einen Durchsatz von 1500 t/a
Internal Report of the Nuclear Safety Project,
No. PNS 80/76, August 1976 (not published)

H. Schüttelkopf
Die Dosisbelastung durch Radiojod und die Anwendung des spezifischen Aktivitätsmodells
Internal Report of the Health Physics Department,
No. ASS/330/04, October 1976 (not published)

H. Schüttelkopf, I. Schlager
Datum und Uhrzeit der Auflöseprozesse von Brennstoff in der Wiederaufarbeitungsanlage Karlsruhe 1971 bis 1976
Internal Report of the Health Physics Department
No. ASS/338/04, January 1977 (not published)

H. Schüttelkopf, I. Schlager
Vergleich der berechneten und gemessenen ^{129}J-Konzentrationen in Milchproben aus der Umgebung der WAK
Internal Report of the Health Physics Department,
No. ASS/342/04, January 1977 (not published)

10. Degree of Availability of the Reports

Reports are available from the author.

Classification: 5.4

Title 1 (Original Language): Theoretische und experimentelle Untersuchungen zur Ausbreitung radioaktiver Gase und Aerosole und der wahrscheinlichkeitsbewerteten Strahlendosen in der Umgebung nuklearer Anlagen nach Störfällen (PNS 4312 - II.6.4, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: ORGANIZATION: Projekt Nukleare Sicherheit GfK Karlsruhe
Title 2 (English): Theoretical and Experimental Investigations of the Atmospheric Dispersion of Radioactive Gases and Aerosols and of the Probabilistically Weighted Radiation Doses around Nuclear Installations due to Accidents	Project Leader: Dr. Hübschmann, Schüttelkopf
Initiated (Date): Januar 1969 Status: Continuing Completed (Date): 1980 Last Updating (Date): Dec. 1976	

1. General Aim

Improvement of the knowledge about the atmospheric dispersion of radioactive emissions

2. Particular Objectives

Short range atmospheric diffusion, long range atmospheric transport and diffusion, diffusion models for accidental releases, meteorological information system

3. Research Program

Tracer diffusion experiments are performed at the various stability categories, chemical tracer gases are emitted at heights between 50 and 200 m.

Theoretical evaluations of the measured data are performed in order to be able to predict the activity concentration pattern downwind of a power reactor after an accidental activity release.

4. Experimental Facilities

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 /5/. Field measurements are performed over areas of different surface structure, using a 15 m mast.

5. Progress to Date

The meteorological data measurement at the tower, the data registration and record have been continued during the report period. A small field measurement station has been installed. Nine diffusion experiments have been performed in 1976. The chemical tracers have been emitted at heights 60 and 100 m. In most experiments two tracers have been released simultaneously at both levels. Seven experiments have been performed at night during stable stratification. The sampling stations are automatically operated.

6. Results

The evaluation of the tracer experiments at stable diffusion categories resulted in a first survey of the diffusion under the local conditions, in particular the high surface roughness, around the research center. The diffusion experiments with 100 m emission height and their results have been comprehensively documented. The evaluation of the measured meteorological data comprises the statistics of diffusion parameters, of the net radiation and of particular meteorological situations, under which the normal atmospheric diffusion equation is not applicable (wind direction shear, stratification of turbulence, precipitation, fog and calms). As a preparation of a German Reactor Risk Analysis, the US-Reactor Safety Study, WASH-1400, has been analysed, in particular the atmospheric diffusion and the health effects model. Necessary modifications for the German Risk Analysis have been determined. A dynamic atmospheric diffusion and external dose model has been designed, which is appropriate to the activity transport over long distances (> 40 km). The computer code is being tested. As part of the accident analysis of AZUR (off gas clean-up of a fuel reprocessing plant) dose distribution calculations have been performed for accidental activity releases.

7. Next Steps

The diffusion experiments will be continued and concentrated on stable weather situations.

8. Relation with Other Projects

Relation exists to the German Reactor Risk Analysis (LRA/IRS), and to the CEA, France (diffusion experiments).

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

10. Availability of the Reports
Reports are available through Gesellschaft für Kernforschung, Karlsruhe, Zentralbücherei

Classification: 5.4	
Title 1 (Original Language): Wanderung langlebiger Transurane im Boden und im geologischen Untergrund (PNS 4412 - I.2.4, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (English): Migration of longlived Transuranium Isotopes in the Soil and in Geological Formations	Project Leader: Dr. Dippel, Dr. Jakubick
Initiated (Date): Jan 1973 — Completed (Date): 1977	
Status: Continuing — Last Updating (Date): December 1976	

1. General Aim

As a basis for safety evaluations for nuclear facilities quantitative data are worked out with respect to the retention capability of soils and geological formations for transuranium isotopes, to the migration of the transuranium isotopes in the soils and geological formations and to the solubility of these isotopes in natural waters.

2. Particular objectives

The migration of longlived transuranium isotopes from the soil surface into deeper layers is to describe by a mathematical model. To assure that the model represents the real migration behavior the main factors influencing the dispersion of plutonium under middle european climatic conditions is to be determined. The experimental results have to be compared with results of Pu-analyses of selected soil samples for rechecking the mathematical model.

3. Research Program

The problems involved should be solved by the following particular experiments:

To understand the processes of the fixation-or in reverse, the migration - of longlived transuranium isotopes partition coefficient are to be determined by static sorption experiments and the characteristic exchange times by kinetic experiments with the main soil constituents and undisturbed samples of selected natural soils. The latter are so called percolation experiments, which simulate the real condition in which Pu penetrates and moves through the soil layers.

4. Experimental facilities

As experimental facilities are available:
a liquid scintillation spectrometer for determination of low Pu-concentrations in the percolate samples, a high pressure column for the percolation experiments with a time-acceleration effects of 1:2000 with respect to natural condition and a glove box line ready for installation of the column and the other small type experiment for sample make up for the percolation experiments.

5. Progress to Date

The time history of Pu-Input into the soil from the past bomb-fallout was investigated for the Heidelberg area. Using a correlation between the measurements in Ispra and Heidelberg the course of Pu-concentration in the air was reconstructed. From this it was possible to obtain by use of an empirical relation the wet and dry fallout values.

For estimations of Pu-distribution in the soild suitable sites have been selected by specific criteria. Soil samples have been taken from these places. The thickness of the samples soil layers was 2 cm.

The laboratory preparation procedure consisted of drying, desintegration of soil aggregats, homogenization of plant rests. Crashing, grinding and homogenization of soil samples followed thereafter. The prepared samples were forwarded for Pu-analysis. Also estimations of soil physical properties were carried out for the purpose of characterization of soil profiles.

6. Results

Between 1954 and 1970 the plutonium input at the places selected shows an permanent increase. The yearly increase rates differ by a factor of 10 and more. They seem to follow cyclic variations.

The estimation of the soil physical properties yielded characteristic data. In a depth between 8 cm and 10 cm packing density and porosity were found as follows.

	Sandy soil	Silty soil	Org.clay soil
Packing density {g/cm^3}	1.75	1.20	1.08
Porosity {%}	33	53	59

The first measurements of Pu-239, 240 concentration in a sandy soil give:

Depth {cm}	Concentration {dpm/g}
0 - 2	0,029 $\pm$ 0,002
0 - 4	0,024 $\pm$ 0,001
0 - 6	0,011 $\pm$ 0,001
0 - 8	0,007 $\pm$ 0,001

An interpretation of the results of the field experiments will be carried out after all Pu-analyses are finished.

7. Next steps

Checking of the field results and of Pu-behavior under changed conditions will be proved in laboratory experiments. A glove-box line is under construction for this purpose.

The Pu-distribution in different soil types will be compared, the main residence time computed and the relation between concentration distribution and packing, porosity and specific surface area investigated.

Further soil sampling should reassure the reproducibility of the results.

8. Relation with Other Projects

This project is in relation to the project PNS 4132 "Investigation on the radio oecology of I-129" and PNS 4134 "Investigation on the long term radiologic environmental impact by local accumulation of nuclear facilities".

9. References

A. T. Jakubick
Migration of plutonium in natural soils
Proc. Symp. on Transur. Isotopes in the Env.
San Francisco 17.-21.11.75

10. Degree of Availability of the Reports

Publishing Section, Int. Atomic Energy Agency
Kärntner Ring 11
P.O. Box 590
A- 1011 W i e n, Austria

Classification: 5.4	
Title 1 (Original Language): Untersuchung der langfristigen radiologischen Umgebungsbelastung durch eine Anhäufung von nuklearen Anlagen (PNS 4134 - II.6.4, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: ORGANIZATION: PNS/GfK
Title 2 (English): Investigation of the Long-Term Radiological Environmental Impact Caused by an Accumulation of Nuclear Facilities	Project Leader: Dr. Bayer
Initiated (Date): Jan. 1973 Status: finished	Completed (Date): Dec. 1976 Last Updating (Date): December 1976

1. General Aim

The aim of this investigation is the estimation of the radiological impact on the population in regions where an agglomeration of nuclear facilities is expected.

2. Particular Objectives

The estimation of the radiological impact on the population of the Upper Rhine Region.

3. Research Program

The research program has been finalised.

4. Experimental Facilities, Computer Codes

The code "KIRMES" was evaluated for the calculations.

5. Progress to Date

After achieving partial results for the Upper Rhine Region, several improvements of the models describing the atmospheric and hydrospheric transport were made.

6. Results

Newer results have been obtained with the help of these modified models. By inserting some preparatory work for the German Reactor Safety Study (4330) a delay occured in the working plan. The documentation of these results will be issued in 1977.

7. Next Steps

This investigation will be finalised. Future sporadic studies in this field will be mentioned under the key number 4330.

8. Relation with Other Projects

The project has no relation with other projects.

9. References

Report KFK 2373 (1976) (German)
Report KFK 2435 (1977) (German)

10. Degree of Availability of the Reports

Unrestricted distribution.

Classification 5.4

Title 1 Konsekvenser af frigørelser af radioaktive stoffer til atmosfæren	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 Consequences of releases of radioactive substances to the atmosphere	Project leader: O. Walmod-Larsen
Initiated: 1972 Completed: Status: Progressing Last updating:	Scientists: S. Thykier-Nielsen P. Hedemann Jensen

1. General aim

Estimation of radiation doses to individuals and population, from releases of radioactive substances to the atmosphere under various environmental conditions.

2. Particular objectives

Development of models for calculation of

a. External gamma-doses from a cloud or plume of radioactive material.

b. Internal doses due to inhalation of radioactive material.

c. External gamma-doses from radioactive material peposited on the ground.

d. Population doses.

e. Global doses (both to individuals and population).

Furthermore the parameters in the models will be studied: Duration of release, atmospheric stability etc.

The consequences of different types of releases both accidental and normal will be investigated.

3. Experimental facilities and programme

None

4. Project status

A model has been developed for calculation of external gamma-doses, as well as internal doses due to inhalation of radioactive material. The Gaussion plume model is used assuming release from a point source.

Based on the model two computer programs has been written for the calculation of doses to individuals: GDOS (external gamma dosed) and INDOS (inhalation doses).

GDOS and INDOS is used as subroutines in the computer-program PLU48 which calculates population doses to a distance of 50 km from a point source.

5. Next steps

Further development as given in 2.

6. Relation with other projects

The models for calculation of external gamma doses and inhalation doses are described in the report:

a. "Modeller til beregning af eksterne gammadoser og inhalationsdoser fra frigørelser til atmosfæren af radioaktive stoffer", S. Thykier-Nielsen, Risø-M-1725.
   A comparison between GDOS and other models for calculation of external gamma doses are given in

b. "Sammenligning af matematiske modeller til beregning af eksterne gammadoser hidrørende fra radioaktivitetsfrigørelser til atmosfæren", Per Hedemann Jensen, Risø-M-1726.

8. Degree of availability

Available on an exchange basis.

Titre	Pays : FRANCE
Elaboration de codes de calcul décrivant les transferts physiques dans l'air, l'eau et le sol.	Organisme directeur : CEA/DSN
Titre (anglais) Computer code describing physical dispersion through air, ground an water.	Organisme exécuteur : CEA/DSN - SESSN Responsable : R.GERARD (SESSN)
Date de démarrage : 01/01/72 Date prévue d'achèvement : 31/12/79 Etat actuel : Etude en cours Dernière mise à jour : 19/11/76	Scientifiques : A.DOUMENC T.ROBERT

Objectif général :

Application pratique de certaines études de base (atmosphériques, hydrologiques, océaniques, hydrogéologiques) par élaboration de codes de calcul donnant des résultats directement utilisables pour l'évaluation des conséquences des rejets normaux ou accidentels dans les différents milieux physiques.

Objectifs particuliers :

Elaboration de codes de calcul décrivant, pour les différents types de sources susceptibles d'être rencontrées les transferts physiques dans l'atmosphère, les transferts physiques dans l'eau (eaux continentales et océans), les transferts physiques dans les sols.

Etat de l'étude :

1) Avancement à ce jour :

Codes de calcul concernant les transferts atmosphériques : émission ponctuelle instantanée, émission ponctuelle prolongée, émission cylindrique instantanée.
Codes de calcul concernant des transferts hydrogéologiques.
Répartition des concentrations radioactives dans un sol.
Modèle monodimensionnel.

2) Résultats essentiels :

Codes de calcul décrivant les transferts atmosphériques : EPOP A, TRANSAT, ECYIN A, TRAIR 1 et TRAIR 2 (sous programmes du code ALICE)
Codes de calcul concernant les transferts hydrogéologiques : FOCON, FACET, TRAMOS 1.

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Prochaines étapes :

Codes de calcul concernant les transferts océaniques, les transferts atmosphériques (sources multiples) et hydrologiques.
Abaque d'évaluation directe des transferts atmosphériques.

Relation avec d'autres études :

Prévisions quantitatives des transferts atmosphériques, hydrogéologiques, océaniques et hydrologiques.
Programme ALICE.

170-21-10/4171-10

Classification : 5-4 793

Titre	Pays :
Etude des transferts atmosphériques.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Studies of atmospheric diffusion and transport.	Organisme exécuteur : CEA/DSN-SESSN
	Responsable : J.P.MAIGNE (SESSN)
Date de démarrage : 01/01/72 Date prévue d'achèvement : 31/12/81 Etat actuel : Etude en cours Dernière mise à jour : 01/01/76	Scientifiques : G.DEVILLE-CAVELLIN

Objectif général :

Cette étude a pour objectif général la prévision quantitative des transferts, par l'atmosphère vers l'environnement, des effluents susceptibles d'être rejetés en permanence ou accidentellement par les installations.

Objectifs particuliers :

Développement et perfectionnement du modèle numérique des transferts par la prise en compte de nouveaux paramètres de la dispersion atmosphérique : variations spatiale et temporelle du vent, surélévation des nuages ou panaches, relief.
Qualification du modèle à partir des études expérimentales des transferts atmosphériques par simulation in situ ou sur maquette.
Développement et perfectionnement des méthodes, probabilités de prévision quantitative des transferts atmosphériques pour des conditions météorologiques non stationnaires (en particulier étude des persistances de situations météorologiques données).

Etat de l'étude :

1) Avancement à ce jour :

Qualification du modèle à partir des études expérimentales.
Réalisation des premières expériences "grands temps de transfert longue distance" (in situ et sur maquette hydraulique).
Réalisation des premières expériences "vents faibles" (in situ).
Mise au point d'une méthode probabiliste de prévision des transferts atmosphériques pour un rejet continu et des conditions météorologiques non stationnaires, à partir de statistiques météorologiques annuelles.

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2) Résultats essentiels :

Vérification expérimentale des prévisions et de la cohérence du modèle numérique de dispersion pour les "grands temps de transfert longue distance".
Application d'une méthode probabiliste de prévision de transferts atmosphériques à la probabilité de formation de brouillard au sol, dû à l'exploitation de tours de réfrigération.

Prochaines étapes :

Qualification du modèle numérique de dispersion dans des conditions météorologiques ou orographiques complexes telles que terrain bâti et vents faibles (brisés).
Mise au point d'une méthode probabiliste de prévision de transferts atmosphériques par situations météorologiques non stationnaires, pour un rejet de durée finie.

170-31-10/4173-10

Classification : 5-4 795

Titre Etude des transferts hydrogéologiques.	Pays : FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Studies of hydrogeological dispersion.	Organisme exécuteur : CEA/DSN-SESSN
	Responsable : J.DUCLOS (SESSN)
Date de démarrage : 01/01/76 Date prévue d'achèvement :31/12/81 Etat actuel : Etude en cours Dernière mise à jour : 01/01/76	Scientifiques : J.PEYRUS C.ESCANDE

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 de ces mêmes exutoires.

Objectifs particuliers :

Qualification de modèles mathématiques prévisionnels de transferts hydrogéologiques.
Mise au point de méthodes expérimentales permettant d'obtenir les données nécessaires à la vérification des modèles mathématiques prévisionnels et à leur adaptation à chaque cas particulier.
Mise au point de sous-programmes particuliers pour la transformation des mesures "terrain" brutes en données utilisables par les modèles (forage-concentration).

Installations expérimentales et programme :

Les installations expérimentales sont les suivantes :
Maquette métrique constituée par une cuve cyclindrique installée dans les locaux du laboratoire de radiogéologie de Bordeaux I;
Terrain d'expérimentation du Barp (33) qui sera utilisé à la simulation des transferts hydrogéologiques à l'échelle décamétrique.
Le programme d'utilisation sera :
Reprise des lois d'absorptions dans les sols (après mise au point d'une "sonde Tritium" utilisable en forage),
Mise au point d'une méthodologie de calcul in situ des coefficients de diffusion microscopique pour un milieu homogène donné (sables au Barp).
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 (FOCON).

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796 170-21-20/4171-10 Classification : 5-4

Titre	Pays :
Comparaison entre méthodes de simulation et transferts atmosphériques.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais)	Organisme exécuteur : CEA/DSN-SESSN
Comparison between atmospheric diffusion and transport simulation methods.	Responsable : J.P.MAIGNE (SESSN)
Date de démarrage : 01/01/77 Date prévue d'achèvement : 31/12/80 Etat actuel : Etude à lancer Dernière mise à jour : 19/11/76	Scientifiques : G.DEVILLE-CAVELLIN R.CRABOL

Objectif général :

Cette étude a pour objectif de comparer entre eux les résultats des trois méthodes de simulation des transferts atmosphériques :
- Simulation numérique au moyen du modèle prévisionnel mathématique sur le terrain.
- Simulation expérimentale en vraie grandeur par traceurs gazeux.
- Expérimentation sur un modèle physique réduit (maquette) en veine hydraulique.

Objectifs particuliers :

Il s'agira tout d'abord de vérifier qu'une étude sur maquette en veine hydraulique permet une bonne simulation des phénomènes de dispersion, en particulier par le respect des conditions de similitude. Ensuite on effectuera parallèlement des simulations sur le terrain par traceurs gazeux et sur maquette dans des conditions météorologiq et/ou orographiques complexes. Les résultats obtenus seront comparés entre eux et confrontés aux prévisions fournies par le modèle mathématique utilisé. La comparaison des divers résultats, en fonction de critères d'ordre pratique (facilités de mise en oeuvre, coûts,...) permettra alors de choisir la ou les méthodes les plus aptes à répondre à différents problèmes particuliers de transferts atmosphériques.

Installations expérimentales et programme :

Veine hydraulique SECURIPOL à EVIAN. Matériel d'expérimentation par simulation à l'aide de traceurs gazeux.

170-41-10/4175-20

Classification : 5-4

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Titre	Pays :
Etude des transferts océaniques et hydrologiques	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais)	Organisme exécuteur : CEA/DSN-SETSSR
Hydrologycal and sea environment diffusion	Responsable : J.PORTE (SETSSR)
Date de démarrage : 1/1/76 Date prévue d'achèvement 31/12/81 Etat actuel : en cours Dernière mise à jour : 1/4/77	Scientifiques :

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 et des voies d'eau, des effluents susceptibles d'être rejetés en permanence ou accidentellement par les installations.

Objectifs particuliers :

Qualification des modèles mathématiques prévisionnels de transferts dans les milieux liquides. Mise au point de méthodes expérimentales permettant d'obtenir les données nécessaires à la qualification des modèles mathématiques.

Prochaines étapes :

Acquisitions de données océaniques en vue des études théoriques et expérimentales des transferts océaniques et hydrologiques.

Documents de référence :

"Une méthode pratique pour la prévision numérique des pollutions océaniques" A.DOURY - C.BADIE - Rapport CEA 4512.

Etat de l'étude :

1) Avancement à ce jour :

La mise au point du modèle mathématique prévisionnel monodimensionnel est terminée (TRAMOS I) ainsi que le modèle mathématique avec détermination automatique des coefficients (TRAMOS II).
La maquette métrique est opérationnelle depuis le mois de septembre 76, les premières expérimentations ont été réalisées en octobre 1976, et doivent se poursuivre suivant le programme cité plus haut.
Le terrain d'expérimentation du Barp a été choisi, son installation est en cours.

2) Résultats essentiels :

Des évaluations mathématiques monodimensionnelles de transferts hydrogéologiques ont déjà réalisées avec succès à Saclay notamment.

Prochaines étapes :

Recherches des méthodologies et établissement d'abaques (définies dans le programme d'utilisation des installations expérimentales).
Mise au point d'un modèle mathématique prévisionnel bidimensionnel de transferts hydrogéologiques.

Relation avec d'autres études :

Interactions sol-installations. Expériences sur la sûreté des stockages des déchets (DSN-SESTR), Cadarache).

Documents de référence :

"Evaluation mathématique monodimensionnelle des transferts hydrogéologiques TRAMOS-1", C.ESCANDE - Rapport SESSN R6, novembre 1976.
"Campagne hydrogéologique de sûreté dans le corail pour le site de Mururoa", JC.PEYRUS, C.ESCANDE - Rapport SESSN R8, avril 1976.
"Evaluation mathématique d'une éventuelle pollution de la nappe des sables de Fontainebleau par le site de Saclay", Fiche Technique 76/287, avril 1976.
"Recherche des isothermes d'absorption par l'expérimentation", C.ESCANDE - Note Technique SESSN 644, 1976.
"Projet d'expérimentation hydrogéologique sur maquette pour la mise au point de modèles prévisionnels de transfert", JC.PEYRUS - Rapport SESSN R11, septembre 1976.

Titre	Pays : FRANCE
Etude des caractéristiques démographiques des sites sous l'angle de la sûreté.	Organisme directeur : CEA/DSN
Titre (anglais) Demographic studies of sites characteristics with respect of safety.	Organisme exécuteur : CEA/DSN-SESSN Responsable : R. GERARD
Date de démarrage : 01/01/72 Date prévue d'achèvement : 31/12/81 Etat actuel : en cours Dernière mise à jour : 19/11/76	Scientifiques : A.DROUET JP.MADOZ

<u>Objectif général</u> :

Acquisition automatique rapide de données démographiques sûres sous forme normalisée.
Détermination de critères quantitatifs permettant de prendre en compte les données démographiques pour l'examen des dossiers de sûreté des installations nucléaires.

<u>Objectifs particuliers</u> :

Etablissement et mise à jour permanente d'un fichier informatique susceptible de fournir à tout moment l'implantation des populations sédentaires, temporaires et prévisionnelles autour de tout site nucléaire métropolitain.
Etablissement et mise à jour permanente de dossiers d'information concernant l'urbanisme et le développement économique autour des sites nucléaires.
Etablissement et utilisation de critères permettant un jugement rapide et objectif des caractéristiques démographiques d'un site en liaison avec ses autres caractéristiques.

<u>Etat de l'étude</u> :

1) Avancement à ce jour :

La réalisation d'un premier fichier-programme "DISPO" susceptible de donner la répartition des populations sédentaires autour d'un site quelconque est achevée.
La réalisation du programme MERADE sous sa forme expérimentale permet l'utilisation comparative de divers critères de pondération des données météorologiques, radiologiques et démographiques.

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2) **Résultats essentiels :**

Le programme DISPO contribue à la réalisation des analyses de sûreté depuis août 1976.
Le programme MERADE a déjà permis quelques comparaisons de sites et va faire l'objet d'une utilisation permanente au fur et à mesure de l'acquisition de données démographiques et météorologiques complémentaires.

Prochaines étapes :

Classement sommaire de 18 sites nucléaires français en fonction de leurs caractéristiques démographiques et météorologiques.
Améliorations du fichier-programme DISPO (communes urbaines, régions frontalières).
Acquisition systématique des SDAU et des PRDE.
Etablissement du programme MERADE dans sa forme définitive.
Réalisation de fichiers-programme DISPO pour les populations temporaires et prévisibles.

Relation avec d'autres études :

Interaction site-installations.
Etudes météorologiques principalement dans le domaine de l'acquisition des données.

Documents de référence :

"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.
"Le chapitre "site" des rapports de sûreté - projet 2", Rapport SESSN R 02 , Rapport SESR R 32, mai 1975.
"Critères démographiques pour les sites s'installations nucléaires", R.GERARD - Rapport SESSN R 10, juin 1976.
"Distribution de la population sur un site métropolitain", A.DROUET - Note SESSN, août 1976.

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Titre Radioprotection dans les installations nucléaires.	Pays : FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Health physics in nuclear installations.	Organisme exécuteur : CEA/DSN-SETSSR
	Responsable : J.GUIRLET (SETSSR)
Date de démarrage : 1/1/76 Date prévue d'achèvement : 31/12/81 Etat actuel : En cours Dernière mise à jour : 15/11/76	Scientifiques :

Objectif général :

Conception de la radioprotection dans les installations nucléaires
Examen des moyens de mesures disponibles. Expérience de fonctionnement. Choix des appareils.
Développement de nouvelles générations de moyens de mesure
Protection du personnel.

Objectifs particuliers :

Définition des zones et accès.
Etablissement d'un catalogue d'appareils de mesure avec mise à jour permanente.
Recommandations à l'usage des projeteurs et constructeurs.
Guide pour l'analyse de sûreté.

Etat de l'étude :

Avancement à ce jour :

Définition du système de radioprotection d'ORPHEE.
Définition du mode d'exploitation du système de radioprotection de CABRI.

Prochaines étapes :

Choix d'appareils pour réalisation d'un catalogue
Classement - pour les conditions de fonctionnement normal
- pour les conditions d'accident
- pour la mesure des rejets.

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<u>Relation avec d'autres études</u> :

Etude de contamination de circuit primaire PWR et neutrons rapides.
Evaluation des doses (irradiation et contamination) dans les installations nucléaires.
Confinement, ventilation
Réglementation.

Title 1 (Original language)	Classification
Valutazione quantitativa del rilascio di sostanze radioattive naturali nell'ambiente	$\boxed{5.4}$ - 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

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) Studi sulla contaminazione del mare	Classification $\boxed{5.4}$ - 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

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.

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Title 1 (Original language)	Classification
Ricerche sui radionuclidi nell'ambiente	$\boxed{5.4}$ - 5.5 - 5.6
Title 2 (English)	**Country** ITALY
Researches on radionuclides in the environment	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) Researches on some stable elements in the environment	**Country** ITALY **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	$\boxed{5.4}$ -- 5.6
Title 2 (English) A mathematical model for the evaluation of the atmospheric diffusion concerning the radioactive gaseous effluents	Country: ITALY 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.

Classification 5.4	
Title 1 Gevolgen voor de omgeving van ongevallen bij kernenergiecentrales	Country The Netherlands Organization KEMA
Title 2 Environmental effects of nuclear power plants accidents	Projectleader B.Th. Eendebak

1. General aim

To analyse the risks of light water reactors on specific sites in the Netherlands.

2. Particular objectives

To study the effects of nuclear accidents as a function of site, population density, wheather conditions, etc.

3. Experimental facilities

Not applicable.

4. Project status

Computer code "MAKRO" is available.

5. Next steps

Not applicable.

6. Relation with other projects

This project was started by an order of the Minister of Economic Affairs to make a risk analysis of the fuel cycle in the Netherlands. This study was finished in June 1975.
See also the projects "Calculation of the quantities of radioactivity released as a result of a serious reactor accident" and "Failure analysis by application of event and fault trees".

7. Reference documents

Not available yet.

8. Degree of availability

Through the organization KEMA.

	Classification 5.4, 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.

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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.

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)	Classification
Ricerche sui radionuclidi nell'ambiente	$\boxed{5.4}$ - 5.5 - 5.6
Title 2 (English)	**Country** ITALY
Researches on radionuclides in the environment	Sponsor / Organisation: CNEN
Date initiated: January 1961 Date completed: In progress Last updating: April 1977	Project Leader Giorcelli

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Title 1 (Original language) Studi sulla contaminazione del mare	Classification $\boxed{5.4}$ - 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

Title 1 (Original language) Valutazione quantitativa del rilascio di sostanze radioattive naturali nell'ambiente	Classification $\boxed{5.4}$ - 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

Classification 5.5	
Title 1 Bepaling van het aantal lekke splijtstof-staven en de kernpositie tijdens het reactorbedrijf	Country The Netherlands Organization KEMA
Title 2 Determination of the number of leaking fuel rods on the core position during operation	Projectleader J. Hoekstra

1. General aim

To reduce the wet-sipping time.

2. Particular objectives

- An increase of the off-gas activity combined with a one step control-rod movement indicates the position of ruptured fuel
- The release of certain fission-products during reactor start-up is a measure for the number failed fuel rods.

3. Experimental facilities

Dodewaard nuclear power plant.

4. Project status

Still in progress.

5. Next steps

Not applicable.

6. Relation with other projects

None.

7. Reference documents

None.

8. Degree of availability

Through the organization KEMA.

171-11-10

Classification : 5-6

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Titre	Pays :
Code de calcul(ALICE) destiné à évaluer l'activité libérée par une installation nucléaire et les conséquences pour l'environnement en cas d'accident.	FRANCE Organisme directeur : CEA/DSN
Titre (anglais) Calculation Code (ALICE) for estimating the activity released by a nuclear plant and consequences for the environnement in the accident case.	Organisme exécuteur : CEA/DSN -SETSSR Responsable : D.MANESSE (SETSSR-BESR)
Date de démarrage : 01/09/75 Date prévue d'achèvement : 31/12/77 Etat actuel : en cours Dernière mise à jour : 02/02/76	Scientifiques :

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.

Objectifs particuliers :

Cette étude doit se traduire par des notes de synthèses exposant les raisons des choix et options, et par un programme de calcul de conception modulaire et évolutif. La réalisation de cette étude implique 3 grands domaines de recherche :

- l'intérieur de l'installation représentée comme un système à compartiments multiples où divers processus physiques et dispositifs peuvent influer sur le transfert des produits radioactifs.
- le transfert atmosphérique avec un modèle adapté aux différentes situations météorologiques possibles sur un site.
- les conséquences radiologiques sur l'homme induites par les différentes voies : irradiation externe par le panache, par les dépôts, contamination interne par inhalation et ingestion...

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 pouvant représenter les phénomènes de transfert atmosphérique.

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Etat de l'étude :

1) Avancement à ce jour :

L'étude est divisée en 2 grandes parties : l'intérieur et l'extérieur de l'installation. En ce qui concerne l'intérieur de l'installation, l'étude préliminaire est terminée, le mode de résolution des systèmes d'équation est défini, l'organigramme du programme est en cours d'écriture. Pour la seconde partie, le fonctionnement normal, ou rejet à débit constant est terminé, tant du point de vue transfert atmosphérique (utilisation d'abaques ou du modèle classique anglo-saxon) que du calcul des conséquences radiologiques, la programmation du modèle à bulles, qui offre plus de possibilités, est commencée.

2) Résultats essentiels :

La partie rejet à débit constant (transfert atmosphérique et conséquences radiologiques) a permis de calculer :
- l'appauvrissement du panache dû aux décroissances, filiations, dépôts secs et humides.
- les dépôts sur le sol avec décroissances et filiations
- la dose due aux dépôts avec un temps de séjour variable
- l'intégration des doses sur des régions compte tenu des distributions de type de temps et de la répartition de population.

Prochaines étapes :

1) Programmation et mise au point de toute la partie concernant l'intérieur de l'installation.

2) Programmation et mise au point du modèle à bulles pour pouvoir représenter, dans toute leur complexité, les rejets accidentels.

Relation avec d'autres études :

- Dépôts des aérosols dans les enceintes à la suite d'un feu de sodium
- Etudes thermodynamiques lors d'accidents.
- Perfectionnement du modèle à bulles (programme TRAIR 2)
- Etude des facteurs de conversion de dose pour l'inhalation
- Recherche de critères de site basés sur un calcul probabiliste de conséquences radiologiques.

Documents de référence :

Note SESR N 85
Note NovaTome ENU - TPD - 195 (version provisoire)
Note Technique SETSSR N 153

Titre	Pays : FRANCE
Evaluation de la répartition de la contamination éventuelle en cas d'accident dans l'environnement d'une installation nucléaire.	Organisme directeur CEA/DSN
Titre (anglais) Valuation of the distribution of possible contamination in case of accident in a nuclear installation environment.	Organisme exécuteur CEA/DSN - SETSSR Responsable : J.GUIRLET (SETSSR - BESR)
Date de démarrage : 1/1/76 Date prévue d'achèvement : 31/12/78 Etat actuel : en cours Dernière mise à jour : 30/3/77	Scientifiques : D.MANESSE F.LAVEISSIERE

Objectif général :

L'objectif général est d'évaluer les conséquences radiologiques estimées en doses et en dommages sur la population à la suite d'un accident grave. Une application est la recherche d'un critère de choix de sites basé sur un calcul probabiliste des conséquences. Une autre application est la préparation des mesures d'urgence à prendre à la suite d'un accident.

Objectifs particuliers :

Cette étude doit contribuer à proposer aux autorités de sûreté :
- Une adaption de l'installation au site choisi (par exemple : puissance, nombre de tranches, dispositifs de sauvegarde, type de confinement...)
- Un plan de développement contrôlé des activités économiques et de l'urbanisation.

Installations expérimentales et programme :

Utilisation du programme ALICE et des programmes annexes de traitement des données (météorologie, population...)

Etat de l'étude :

1) Avancement à ce jour :

Une première étude de critères de site (notes BESR 20, 21, 25), basée sur la catégorie de rejet PWR 2 du WASH 1400 a été appliquée à 15 sites français (anciens et nouveaux) et à 4 sites américains. Le seul dommage considéré pour cet accident maximal étant la mort dans les 60 jours à la suite d'une irradiation de la moëlle.

2) Résultats essentiels :

Un premier classement des sites a pu être proposé.

.../...

Prochaines étapes :

- Détermination d'un spectre représentatif d'accidents avec les probabilités associées.
- Prise en compte de plusieurs types de dommages
- Poursuites des études de plans d'intervention pour les PWR.

Relation avec d'autres étapes :

- Sélection d'accidents représentatifs, scénarios de ces accidents. (PWR et NR)
- Connaissances précises des statistiques météorologiques et des répartitions de population.
- Relations dose-dommage, effets à moyen et long terme.

Documents de référence :

-Rapport WASH 1400 pour les PWR

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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: 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 **Status** :	**Completed** : 1976 **Last updating**	**Scientists:**

6. FAULTS AND ACCIDENT COMBINATIONS

Classification 6

Title Analysis of reportable incidents from light water reactors	COUNTRY Denmark SPONSOR ORGANIZATION
	Project leaders J.R. Taylor H. Kongsø
Initiated 1973 Status In progress	

General aim

The general aim of the project is to keep a running analysis of the safety related incidents and component failures occurring in light water reactors, based on USNRC incident reports. Detailed analysis of causes and structure of failure events have been made, for selected incidents, and for selected reactors over their complete reported record.

Next steps

Further steps will be to keep this record up to date. A computer based recording of failures classified by cause, has been started.

Reference documents

Two reports have so far been published:
J.R. Taylor, Design Errors in Nuclear Power Plant, Risø-M-1742, September 1974 and J.R. Taylor, A Study of Abnormal Occurrence Reports, Risø-M-1837, September 1975.

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Availability

Reports are available on request.

Classification 6

Title Common cause failure	COUNTRY Denmark SPONSOR ORGANIZATION
	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 Reliability of computer based control	COUNTRY Denmark SPONSOR ORGANIZATION
	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.

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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.

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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

Formal methods are established in order to achieve the determination of minimal cut sets from fault trees. Methods are based upon the segmentation of the tree, construction of minimal cut sets of subtrees and subsequent expansion into the minimal cut sets of the original tree. The method as compared with the traditional combinatorial techniques has the advantage of consenting (i) determination of the minimal cut sets of any order it may be required, (ii) hand calculations and interactive programming, (iii) direct or built-in sensitivity analyses.

TITLE 1 (original language) Fault analysis of the conventional island in a LWR nuclear power plant.	Classification 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.