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European Community Water reactor Safety Research Projects

VOLUME I

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INTRODUCTION

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

The following guidelines were used in compilation of the index which is 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 assembled in the following order of country : Belgium
   Germany
   Denmark
   France
   Ireland
   Italy
   JRC Ispra
   Luxembourg
   Netherlands
   United Kingdom

4) Updated formats will be inserted in the relevant replacement position. When additional pages have to be inserted they will be numbered with the proceeding page number plus an oblique and an extra number (for example page 53/1 will be inserted following page 53).

5) Formats for new projects will normally be inserted following the last format of the relevant country within that class (chapter).

6) If a project is entered under more than one class (chapter), the full format is given only once in the most important position.

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 Seismic 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 computorized 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

20. Other topics

1. BLOWDOWN AND EMERGENCY CORE COOLING

Classification: 1

Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: NORCOOL. A Model for Analysis of a BWR under LOCA Conditions.	Organization: Risø National Laboratory
Initiated date: September 1976 Completed date: Status: under development	Scientists: O. Rathmann P.S. Andersen P. Astrup N. Bech O. Rosdahl
	P. Albråten P. Hansen J. Kotakorpi M. Gregory

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 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 either on the one-dimensional Fourier equation with two-dimensional conduction at quenching fronts represented through correlations, or on the two-dimensional Fourier equation assuming rotational symmetry.

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. In NORCOOL-II, however, an arbitrary number of parallel fuel elements in the core and the whole primary system inside the vessel may be represented as a network of coupled heated or unheated one-dimensional flow channels.

3. Experimental facilities and programme

4. Project status

   1. Progress to date

      NORCOOL-I is in the completion 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 NORA for reactor systems under development at IFA, Norway.
c) The Danish transient subchannel computer program TINA, the combined transient subchannel and 3-dimensional nodal neutronics program ANTI and the one-dimensional blow down code RISQUE under development at Risø.
d) Updating of COBRA 3-C and RELAP 4 by STF, Finland and Studsvik Energiteknik, Sweden.
e) A 64-rod (electrically heated) core heat-up experiment by Studsvik Energiteknik, Sweden.

7. Reference documents

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

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

8. Degree of availability.

145-1-01		1
TITRE Accident de perte de caloporteur dans les réacteurs à eau pressurisée : codes 1ère génération.		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) LOCA and ECCS studies on PWR : first generation codes.		Organisme exécuteur CEA/DSN-SRS/SEAREL Responsable
Date de démarrage 01/01/71	Etat actuel en cours	Scientifiques
Date d'achèvement	Dernière mise à jour 01/79	

1. OBJECTIF GENERAL

Utiliser les codes de 1ère génération (essentiellement la famille RELAP 4) pour faire des calculs d'interprétation ou prévisionnels d'expériences françaises et étrangères. Ce travail devrait permettre de déterminer les possibilités de ces codes, leur limitation et les meilleures options à utiliser lorsque seront faits les calculs pour l'évaluation de sûreté des réacteurs de puissance.

2. OBJECTIFS PARTICULIERS :

1) Etude des options des codes RELAP 4 mod 5 et mod 6.
2) Qualification des codes sur expériences OMEGA et problèmes standard CSNI.
3) Précalculs de l'expérience EURATOM LOBI.
4) Calculs préliminaires PHEBUS.
5) Calculs de sensibilité pour les réacteurs de puissance.

3. ETAT DE L'ETUDE

1) Avancement à ce jour

- Les calculs préliminaires PHEBUS sont terminés (voir fiche PHEBUS) ; ils ont permis de mesurer la sensibilité des différents paramètres essentiels à la conduite des essais ; ces études servent de base à l'établissement de la grille des essais.
- Des études de renoyage relatives à PHEBUS ont été réalisées avec les codes CERES et RELAP FLOOD. ./.

- Le problème standard CSNI n° 5 (dépressurisation sans chauffage de LOFT avec injection de sécurité) a été calculé au moyen du code RELAP 4 mod 5 en tant que calcul prévisionnel de l'expérience. Le problème standard n° 7 qui est une expérience de renoyage ERSEC est en cours de calcul au moyen du code RELAP 4 mod 6. Le problème standard n° 8 (expérience de dépressurisation et de renoyage SEMISCALE préparant un essai LOFT) est en cours de préparation au moyen du code RELAP 4 mod 5.
- L'interprétation des essais OMEGA est en cours. On a surtout fait porter l'effort sur un essai sans chauffage et avec chauffage.
- LOBI : les premiers calculs prévisionnels LOBI définis dans la grille d'essais proposés par la France et adoptés par le Groupe d'ISPRA ont été effectués au moyen du code RELAP 4 mod 5.

  Ces calculs correspondent à la partie "grosses brèches" du programme et leur interprétation est en cours par confrontation des calculs des différents participants.
- Le 1er calcul concernant la phase de dépressurisation de l'ADR d'un réacteur de puissance a été réalisé. Il s'agit d'un calcul "physique", certains modèles d'évaluation de la version du mod 5 utilisée, ayant présenté des erreurs. Une mise à jour du code vient d'être effectuée.

2) <u>Résultats essentiels</u>

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

  Les calculs ont montré la nécessité de bien prendre en compte les fuites thermiques dans toute la boucle. Enfin un certain nombre de paramètres, propres à la boucle, se sont révélés comme très importants : citons principalement la perte de charge de la vanne VA EP 14 qui permet de by-passer le circuit de charge au début du transitoire et la perte de charge des creusets en débit direct et inverse.
- Les calculs de renoyage avec RELAP 4-FLOOD ont permis de déterminer les conditions de fonctionnement de ce code et ont montré la nécessité de bien contrôler les conditions initiales du renoyage dans PHEBUS du fait de la petitesse du volume coeur (3 dm3) par rapport au volume du plenum inférieur (56 dm3) étant donné qu'une dilatation de l'eau contenue dans ce dernier volume pourrait accélérer le remplissage du coeur.
- Les premiers calculs OMEGA ont été effectués avec la version RELAP 4 mod 3. Il s'est avéré que pour bien rendre compte des essais sans chauffage il était nécessaire :

  1°/ d'utiliser le débit à la brèche de MOODY avec un facteur $C_D = 0.6$

  2°/ d'utiliser le modèle de séparation de phases dans le plenum opposé à la brèche.

  Les calculs ont été ensuite repris avec la version améliorée RELAP 4 mod 5.

  Il faut avec cette version :

  1°/ utiliser le modèle homogène équilibré à la brèche.

  2°/ utiliser le modèle de glissement entre phases dans la section d'essais et dans les plenums.

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Les résultats de cette version semblent plus proches de l'expérience que ceux de la version mod 3 particulièrement en ce qui concerne les grandeurs à variation lente (P, T_{fl}, M).

L'interprétation des essais avec chauffage a montré des déficiences quant au calcul des transferts de chaleur, les causes pouvant être soit au niveau des corrélations elles-mêmes, soit au niveau des conditions locales du fluide intervenant dans les corrélations.

- L'étude des phases de dépressurisation et de début de remplissage au cours de l'ADR d'un réacteur de 900 MW a été réalisée. Il s'agit d'un premier calcul avec les modèles dits "physiques" du code. Le réacteur a été modélisé par 49 volumes et 65 jonctions. Le coeur est représenté par un canal chaud, un canal moyen et une zone de "by-pass". Le combustible est découpé en 7 zones dans le canal moyen et en 9 zones dans le canal chaud.
  Le cacul a été poursuivi jusqu'à 28 s. de temps réel d'accident. L'injection des accumulateurs a fonctionné à 12,5 s. Le temps total de calcul est de 3 h. sur IBM, la plus grande partie de ce temps correspondant à la phase de remplissage.
- La calcul du problème standard n° 5 a montré la nécessité de modéliser la zone annulaire par deux volumes en parallèle, l'un relié à la branche rompue et l'autre à la branche intacte, afin de permettre à la vapeur de s'échapper durant la phase d'injection de l'eau de secours. Les proportions entre ces 2 volumes sont 1/3 et 2/3.

4 - PROCHAINES ETAPES

- Un calcul d'accident réacteur avec les modèles d'évaluation est en cours. Il pourra servir de base à une étude de sensibilité ultérieure. Des études de scénarios particuliers d'accidents pourront être réalisés.
- OMEGA : poursuite d'interprétation des expériences en grappes et expériences avec une meilleure instrumentation.
  Utilisation du code RELAP 4 mod 6 dès que les tracés de courbes liés à ce code seront opérationnels.
- LOBI : Adaptation de certains coefficients physiques (telle que perte charge) après les premiers essais physiques préliminaires réalisés sur la boucle.
- Poursuite des calculs des 2 problèmes standards en cours : N° 7 et 8.

5 - RELATIONS AVEC D'AUTRES ETUDES

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

7 - DOCUMENTS DE REFERENCE

- "OMEGA : dépressurisation adiabatique. Analyse par le code RELAP 4". M. MEZZA, Note Technique SETSSR 77-193.
- "Boucle Blowdown (LOBI) - ISPRA. Calcul de référence réalisé au DSN à l'aide du code RELAP 4 mod 3" A. SONNET, Note Technique SETSSR 77-209.

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- Publication PHEBUS (voir fiche PHEBUS 147-1 - 01).

- Premier calcul de l'accident de référence d'un réacteur de 900 MW avec le code RELAP 4 mod 5.
  R. POCHARD, H. TARTU (Note Technique SEAREL 78/02)

- Utilisation du code RELAP 4-FLOOD pour l'étude de la phase renoyage dans PHEBUS.
  N. TELLIER et H. TARTU (Note Technique SEAREL 78/01).

145-1-02		1
TITRE : DEVELOPPEMENT DE MODELES ET CODES DE CALCUL DE 2è GENERATION POUR L'ETUDE DE L'ACCIDENT DE PERTE DE CALOPORTEUR DANS LES REACTEURS A EAU PRESSURISEE : Programme général coordonné.		Pays FRANCE
		Organisme Directeur CEA-DSN EDF-SEPTEN
TITLE (Anglais) DEVELOPMENT OF ADVANCED MODELS AND ADVANCED CODES FOR THE STUDY OF THE LOCA IN PWR : Coordinated general program.		Organisme exécuteur CEA - EDF
		Responsable
Date de démarrage 01/01/77	Etat actuel en cours	Scientifiques
Date d'achèvement 31/12/81	Dernière mise à jour 1/79	

1. OBJECTIF GENERAL

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

2. OBJECTIFS PARTICULIERS

2.1 - Ecriture de modèles physiques

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

Les modèles mis au point pour CLYSTERE sont en partie utilisés

- Ecoulement diphasique 1 D
- Singularités géométriques
- Brèche 1 D
- Piquage pressuriseur
- Injection de secours
- Ecoulement et transfert de chaleur en dépressurisation
- Ecoulement et transfert de chaleur en amont du front de trempe
- Ecoulement et transfert de chaleur en aval du front de trempe
- Conduction 2 D - Modèle de front de trempe

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- Volume 0 D
- Downcomer
- Volume 3 D
- Piquage sur volume
- Thermomécanique des crayons
- Singularités complexes
- Cross flow
- Brèche 3 D
- Neutronique
- Ecoulement air eau vapeur
- Séparation sur les structures
- Modèle de flooding
- Modèle de pompe

2.2. Validation des modèles physiques

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

2.3. Elaboration d'un système informatique (POSEIDON)

Cette élaboration comprend les tâches suivantes :

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

Ce système doit conduire à un code de type modulaire.

2.4. Méthodes numériques

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

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

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2.5. Ecriture des modules

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

Les modules recensés sont les suivants :

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

Confinement : - écoulement entre casemates
- enceinte

2.6. Réalisation d'assemblages

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

OMEGA et CISE
MARVIKEN
AQUITAINE
ERSEC système

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PKL
SEMISCALE renoyage
PHEBUS
LOFT - SEMISCALE
LOBI

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

3 - INSTALLATIONS EXPERIMENTALES ET PROGRAMME

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

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

4 - ETAT DE L'ETUDE

4.1. Avancement à ce jour

4.1.1. Elaboration du système informatique

ESOPE - OTOMAT

De nombreux programmes sont actuellement écrits en langage ESOPE qui est ainsi testé de façon pratique.

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Fonctions de l'eau

Le jeu de fonctions qui avait été réalisé a été progressivement amélioré au cours des utilisations. Un découpage des tables a été réalisé.

Banques - bibliothèques

Une première version des banques et bibliothèques est actuellement opérationnelle. Elle permet le traitement de composants ne comportant que deux connections (entrée-sortie).

Moniteurs

Une première version des moniteurs est actuellement opérationnelle. Cette version est effectuée avec l'itérateur ITSOR basé sur la méthode numérique de couplage SOR (voir paragraphe suivant).
Ces moniteurs permettent le traitement de composants ne comportant que deux connections (entrée-sortie).

Mini-systèmes

Un assemblage à 3 composants (tuyauteries) a été testé avec les versions des banques, bibliothèques et moniteurs décrits aux paragraphes précédents.

4.1.2. Méthodes numériques

Méthodes numériques d'assemblage

Des tests ont été effectués sur la méthode par fonction de transfert sur des assemblages de tuyauteries décrits par les codes SERINGUE puis HEXECO.

La méthode SOR a été introduite dans le système POSEIDON.

Optimisation des méthodes numériques

Le temps de calcul du module écoulement a été amélioré en particulier par l'introduction de dérivées analytiques en remplacement des dérivées numériques. Des temps de calcul comparables entre le module à 6 équations et le module SERINGUE ont été obtenus.

Un travail de comparaison de méthodes numériques a été démarré. Ce travail consiste en l'exécution d'un premier problème test français (calcul CANON). Il comprend également la participation au problème standard numérique de l'OCDE/CSNI.

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

Module écoulement

Le module d'écoulement HEXECO 001 a été introduit dans POSEIDON.

L'interprétation des essais MOBY DICK eau-air a été reprise. Une forme améliorée des termes de transfert de quantité de mouvement en a été déduite.

Des identifications avec d'autres modèles d'écoulement ont été réalisées pour élargir la gamme de validité du modèle d'écoulement.

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Module singularités

Ce module a été écrit sous forme d'équations de saut à partir de la formulation à 6 équations HEXECO.

Module tuyau chauffant en dépressurisation

Le traitement de parois chauffantes a été ajouté au module écoulement. Le module tuyau chauffant en dépressurisation ainsi constitué (HEXECO 004) a été testé et est actuellement utilisé pour des premiers dépouillements partiels (limités au tube chauffant) des essais OMEGA. Ce module comprend les améliorations numériques indiquées au paragraphe 4.2.

Module tuyau chauffant en renoyage

- Code FLIRA

Différents modèles ont été incorporés au code FLIRA :

. modèle de glissement algébrique.

. modèle B de coefficient d'échanges au niveau du front de trempe.

Des tests de méthodes numériques ont été également effectués.
Des comparaisons de calculs FLIRA et de calculs effectués avec le code PSCHIT ont été faits pour vérifier le traitement du front de trempe.

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

  Le calcul de l'écoulement a été réalisé avec le modèle drift flux. Les résultats de ce calcul ont été comparés avec les résultats expérimentaux.

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

  La mise au point d'un jeu de modèles ayant reçu une première validation sur les essais ERSEC est en cours.

- Modèle de front de trempe.

  La corrélation de coefficient d'échange au niveau du front de trempe (modèle B) mise au point avec l'aide du code PSCHIT a été étendue aux pressions allant jusqu'à 5 bars.
  Des tests de sensibilité lorsque l'on transpose les calculs au combustible ont été poursuivis. Des tests numériques de sensibilité au maillage ont été également effectués.

Module pompe

Les ajustements du modèle en écoulement monophasique dans les 4 quadrants ont donné des résultats en assez bon accord avec les résultats expérimentaux, sauf dans le 2ème quadrant.

Des calculs en écoulement diphasique ont été faits sur les essais EVA et EPOPEE. Des difficultés de prévision des phénomènes de dégradation des caractéristiques sont rencontrées dans l'interprétation des essais EVA. Le calcul des essais EPOPEE indiquent des blocages soniques (limitation du débit). Ces derniers résultats doivent être vérifiés par de nouvelles analyses.

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Module tuyau avec piquage injection de secours

Un premier modèle d'oscillations a été écrit et permet de retrouver qualitativement les résultats expérimentaux. Ce modèle est en cours d'amélioration.

Module crayon combustible

L'écriture de ce module a été démarrée.

4.1.4 - Assemblages

Les éléments nécessaires pour l'assemblage OMEGA ont été mis au point et écrits en langage ESOPE. L'assemblage lui-même est en cours de réalisation.

4.2 - Résultats essentiels

- Le modèle de déséquilibre mécanique a été amélioré et donne une prévision assez bonne des essais MOBY DICK eau-air.

- Le modèle B de coefficient d'échange au niveau du front de trempe a été validé sur l'ensemble des essais ERSEC tube.

- Les interprétations des essais EVA et EPOPEE avec le module pompe ont permis de retrouver qualitativement certains phénomènes et ont contribué à une certaine compréhension physique.

5 - PROCHAINES ETAPES

5.1 - Elaboration du système informatique

Banques - Bibliothèque - Moniteurs

Amélioration et extension aux composants à nombre d'entrées-sorties supérieur à 2.

Système

Tests d'assemblage sur des circuits complexes.

5.2 - Méthodes numériques

Méthodes numériques d'assemblage

Etude de synthèse sur les méthodes SOR et fonction de transfert.

Poursuite des tests.

Optimisation des méthodes numériques module

Suite du travail de comparaison (problème standard OCDE/CSNI et problème test français).

./.

5.3 - Ecriture des modules - Développement et validation des modèles physiques

Module écoulement

- Reprise des dépouillements MOBY DICK eau-vapeur avec les corrélations améliorées de déséquilibre mécanique.
- Calculs d'écoulements dans des gammes différentes de paramètres.
- Traitement des apparitions de niveau.

Module singularités

- Etude de sensibilité aux modèles et vérification sur l'expérience.

Module tuyau chauffant en dépressurisation

- Validation sur les essais OMEGA tubes.

Module tuyau chauffant en renoyage

- Améliorations du code complet et notamment traitement du début de renoyage.
- Suite de la mise au point d'un jeu de modèles d'écoulement et de coefficients d'échange en amont et en aval du front de trempe. Validation sur les essais ERSEC.

Module pompe

- Poursuite des interprétations de l'expérience EPOPEE.
- Examen de l'introduction du modèle d'écoulement à 6 équations dans le module pompe.

Module tuyau avec piquage injection de secours

Poursuite de l'amélioration du modèle d'oscillations.

Modules volume OD

Démarrage de la modélisation.

Module crayon combustible

Poursuite de l'écriture de ce module.

Modules composants

L'écriture d'un certain nombre de modules composants devrait être démarrée en prenant comme base le module d'écoulement HEXECO.

5.4 - Assemblages

Les assemblages MARVIKEN, AQUITAINE, PKL..... devraient être effectués au fur et à mesure de la disponibilité des modules.

./.

6 - RELATIONS AVEC D'AUTRES ETUDES

- Voir paragraphe 3.

7 - DOCUMENTS DE REFERENCE

G. HOUDAYER, G. LE COQ, B. PINET, M. REOCREUX, J.C. ROUSSEAU
Modeling of two phase flow with thermal and mechanical non equilibrium.
Rapport DSN 166 e - Présentation at the 5 th Water Reactor Safety
Research Information Meeting - Washington 1977.

P. CLEMENT, R. DERUAZ, J.P. L'HERITEAU, P. RAYMOND, P. REGNIER, M. REOCREUX
Development of reflood code FLIRA and PSCHIT. Physical modeling and interpretation of ERSEC experiments.
Rapport DSN 167 e - Presentation at the 5 th Water Reactor Safety
Research Information Meeting - Washington 1977.

M. REOCREUX, H. SUREAU, J. THIBAUDEAU, M. CHABRILLAC, M. COURTAUD, M. GOMOLINSKI,
French thermo-hydraulic studies for the development of safety advanced code for PWR .
ENS/AMS International Topical meeting on Nuclear Power Reactor Safety
BRUXELLES (16-19 Oct. 1978) .

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

1. General aim

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

2. Particular objectives

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

B Forces on internals in the vessel during blow down

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

3. Experimental facilities and programme

- Participation in the Marviken project
- Main computer programmes:

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

4. Project status

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

5. Next steps

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

6. Relation with other projects

Not applicable.

7. Reference documents

Internal KEMA reports.

8. Degree of availability

Free on basis of exchange with other programmes and results.

N.V. KEMA	CLASSIFICATION : 1. 7.1. 7.2
TITLE :	COUNTRY: THE NETHERLANDS
	SPONSOR : KEMA ORGANIZATION : KEMA
TITLE (ENGLISH LANGUAGE): Calculations of the consequences of pipe breaks in reactor systems	PROJECTLEADER : R.M. vam Kuijk
INITIATED : - LAST UPDATING : 1978 STATUS : progressing COMPLETED : 1980	SCIENTISTS : Kloeg, Oppentocht, Talens

General aim

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

Particular objectives

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

B Forces on internals in the vessel during blow down

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

Experimental facilities and programme

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

Project status

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

Next steps

- Calculation of Marviken II and III results.

Relation with other projects

Not applicable.

Reference documents

Internal KEMA reports.

Degree of availability

Free on basis of exchange with other programmes and results.

CLASSIFICATION 1.1

Classification: 1.1

Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: DINO - Core heat-up during blow down	Organization: Risø National Laboratory
Initial date: February 1971 — Completed date: September 1972 — Status:	Scientists: H. Abel-Larsen, M. Lolk Larsen

1. General aim

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

2. Particular objectives

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

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

3. Experimental facilities and programme

4. Project status

Completed.

5. Next steps

6. Relation with other projects

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

7. Reference documents

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

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

8. Degree of availability

Available.

Classification: 1.1

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

1. General aim

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

2. Particular objectives

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

3. Experimental facilities and programme

4. Project status

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

5. Next steps

Completion of program layout and documentation.

6. Relation with other projects

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

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

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

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

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

7. Reference documents

8. Degree of availability

Available on exchange basis when completed.

		CLASSIFICATION: LWR 1.1 & 1.2
TITLE (ORIGINAL LANGUAGE): Einfluß der DWR-Umwälzschleifen auf den Blowdown (LOBI Projekt)		COUNTRY: FRG C.E.C.
		SPONSOR: FRGMRT C.E.C.
TITLE (ENGLISH LANGUAGE): Influence of the PWR loops on the blowdown (LOBI project)		ORGANISATION: C.E.C. J.R.C.-Ispra
		PROJECT LEADER: W. L. Riebold
INITIATED: Dec. 1, 1973	COMPLETED: Nov. 3o, 1981	SCIENTISTS: Eder, Fortescue, Fritz, Kolar, Larsen, Mörk-Mörkenstein, Ohlmer, Piplies, Städtke
STATUS: Completion of shake-down tests	LAST UPDATING: May 1979	

1. General Aim

Experimental investigation of the thermohydraulic behaviour of a primary cooling system (PCS) of pressurized water reactors (PWR) during the blowdown of a loss-of-coolant accident (LOCA). The experimental results will be used to check and improve existing blowdown computer codes applied for the safety analysis of LWRs.

2. Particular Objectives

Experimental investigation of the LOBI pump characteristics under two-phase flow conditions.

Design and construction of a large scale 2-loop blowdown test facility for performing loss-of-coolant experiments (LOCE) by simulating pipe ruptures of different sizes at various locations within the PCS of PWRs.

Experimental investigation of the influence of the thermohydraulic behaviour of the individual components of the PCS of PWRs on the course of a LOCA (blowdown) by measuring the significant thermohydraulic quantities, in particular those influencing the core cooling.

3. Experimental Facilities and Programme

The objectives of the investigation require a test facility with a
9.507 thermohydraulic behaviour during blowdown which is as close as

possible a reactor-typical one, Fig. 1.

The LOBI test facility is an approximately 1/7oo scale model of a 4-loop 13oo MWe PWR primary cooling system (PCS). It consists of a 5 MW electrically heated test section (reactor model) with two primary cooling loops, one with three times the capacity (water volume and mass flow) of the other, representing the four primary loops of a PWR cooling system.Both loops are active loops each containing a pump and a steam generator. Tube ruptures of various sizes (double ended to small leak) will be simulated at three different locations (hot leg, cold leg, loop seal) within the single or broken loop. Heat is removed from the primary loops by the active secondary cooling circuit by means of two condensers (representing the turbines) and a cooler. From the emergency core cooling system (ECCS) for the time being only the intermediate pressure or accumulator system is represented providing ECC water for both, separate and combined cold and hot leg injection into both loops; the high pressure injection system (HPIS) is planned to be added in the near future.

The scaling factor of 712 has been applied to

- power input: 5 MW to 64 heater rod bundle
- coolant mass flow: 21 kg/s and 7 kg/s for the intact and broken loop respectively
- coolant volume: o,83 m^3 within both primary loops, pressurizer included.

For the test facility design the

- power to volume ratio
- pressure drop and temperature distribution along flow paths
- ratio of component volumes to each other

have been preserved.

The height, and relative heights of components are scaled 1 : 1, thus preserving gravitational heads. The heat transfer surfaces (rod bundle and steam generators) are full length.

The nominal operation conditions are 155 bar and 323° C pressure and temperature respectively and a core mass flow rate of 28 kg/s.

Control facilities allow simulation of

- pump hydraulic behaviour by speed control
- fuel decay and stored heat by power control to the heater rod bundle
- containment back pressure.

An instrumentation system is being installed which provides for the measurement of all relevant thermohydraulic quantities under transient conditions at the boundaries (inlet and outlet) of each individual loop component and within the reactor model.

The signals from about 400 measuring channels are recorded by a specially tailored data acquisition system using PCM techniques and analog magnetic tape.

Test facility design calculations are performed by the "programme and analysis" group of the LOBI project staff with RELAP4-MOD2 and -MOD5 code, by the J.R.C. "design office" using BERSAFE and STRUDL-II codes, and by the GRS-Garching (FRG) with BRUCH-D, DAPSY and ZOCO-VI codes.

According to the LOBI R&D-contract between the Federal Minister for Research and Technology of the Federal Republic of Germany (FRGMRT) and the Commission of the European Communities (C.E.C.), two different experimental programmes, A and B, are to be executed with the LOBI test facility after having completed the preliminary tests aiming at the determination of the various thermohydraulic and mechanical characteristics of the facility.

Test programme A consists of 60 tests to determine the influence of seven relevant parameters on the blowdown: rupture location and size, downcomer gap width, pump operation conditions, heating power input, steam generator secondary pressure, pressurizer connection, ECC water injection.

These 60 tests have been subdivided into two groups, A1 and A2, of 30 tests each:

Test programme A1 (tests 1 - 30) will be defined by the FRGMRT exclusively, and the results are at the exclusive disposal of the FRGMRT. The first test A1-01 will be used for a blind post-test pre-prediction exercise (LOBI PREX) with international participation.

Test programme A2 (tests 31 - 6o) will again be defined by the FRGMRT, however, taking into account suggestions from the member countries of the European Communities, and the results will be freely available to all member countries of the E.C.

Test programme B (tests 61 - 9o) will bedefined by experts from the member countries of the E.C., and the results will be freely available to all member countries of the E.C.. This programme is mainly aimed to perform component studies after having applied appropriate component modifications.

4. Project Status

4.1 Progress to Date

- Completion of mounting and thermal insulation work, execution of commissioning and shake-down tests
- Performance of first blowdown test SD-o1 (shake-down test 1)
- Repair and modification of test facility during an interrupt period of the shake-down test programme
- Preparation of Specifications for LOBI PREX
- Execution and completion of LOBI pump tests in two-phase flow conditions
- Start of contract work for LOBI discharge nozzle calibration tests in two-phase flow conditions
- Completion of three-dimensional analysis of dynamic loadings on the LOBI test facility structures; application of supplementary shock absorbers
- Preparation LOBI test results documentation procedures
- Performance of pre-test prediction calculations
- Revision of test matrix A
- Discussion of results of survey calculations for test matrix B
- Preparation of computer codes for the results evaluation and presentation
- Design studies on LOBI test facility extensions (HPIS, secondary loop modification, auxiliary feed water system)

- Acceptance and calibration tests of the various measurement chains
- Completion of measurement instrumentation system of LOBI test facility
- Preparation and testing of mini-computer programs for data acquisition, processing and representation, and for process control.

4.2 Essential Results

After some commissioning tests, the first blowdown test SD-01 has been performed successfully on December 13, 1978, in the framework of the shake-down test programme. The initial conditions of this double-ended 200 % hot leg break test without ECC water injection have been 50 bar and 260° C pressure and temperature, respectively, with 1,6 MW heating power and nominal core mass flow rate of 28 kg/s. The aim of this test was a first check of the mechanical and thermohydraulic behaviour of the facility under reduced absolute and subcooling pressure conditions.

During the preparation phase of the subsequent test SD-02, failures of the commissioning heater rod bundle occurred, and the blowdown could not be initiated due to a blockage of the quick-opening flap valves used to simulate the rupture.

The heater rod bundle failure was due to tube ruptures close to the upper end of the bundle which have been identified as hot ruptures typical for Nickel under high temperature and minimum mechanical stress condition. Remedial measures have been taken by venting the upper region (upper plenum between hot leg nozzles and upper power connecting plate) of the heater rod bundle and by avoiding stagnant flow conditions in this region allowing a small bypass flow from the upper downcomer region.

The failure of the quick-opening flap valves operation was caused by too small fabrication tolerances and could therefore easily be overcome.

Unexpected electrolytical corrosion of the metallic (VACON) parts of the distance pieces between the heater rod bundle and the core barrel tube (in the unheated region) was the reason for preventing the rod bundle from thermal expansion and caused so mechanical stresses. The distance pieces have been replaced by those made totally from ceramics.

The shake-down tests left, SD-o2, -o3, -o4, will be performed

- with the first experimental heater rod bundle (cos-shaped heat flux, fully instrumented) which has been mounted to replace the damaged commissioning bundle
- as cold leg rupture test, because the interrupt of the shake-down test programme has been used also to displace already now the rupture device from the hot leg position to the cold leg position as foreseen for the first official test of programme A1.

The shake-down test programme has been continued with test SD-o2 performed successfully on May 13, 1979, from 125 bar and 3oo^{o} C pressure and temperature, respectively, and 6o % = 3,1 MW heating power and nominal core mass flow rate of 28 kg/s as initial conditions.

The remaining test SD-o3 and -o4 will then be executed with nominal initial conditions with respect to pressure (155 bar), temperature (323^{o} C), heating power (5,3 MW) and core mass flow rate (28 kg/s), aiming so at a consistency and reproducibility check of the measured results.

On proposal of the USNRC, Washington, the FRGMRT-Bonn agreed on using the first official LOBI test A1-o1 for blind post-test prediction calculations in the framework of an international exercise (LOBI PREX). The particular interest in this calculation exercise is given by the fact that never before any such blowdown code could be adapted to the special transient characteristics of the integral system test facility("virginity" aspect). In total, there will be 14 participants from the European Community (4 member countries) and the USA. The LOBI test facility specifications and all further documentation required to prepare the respective code calculations are contained in [1] and have been submitted to the participants.

The LOBI pump tests at Westinghouse Canada Ltd. (WCL), Hamilton/Canada, aimed at investigating the two-phase flow pump characteristics have been completed. The evaluation of the test results is presently under way; first results are shown in Fig. 2.

Theoretical considerations have shown that the transient pump behaviour can be sufficiently described by the steady-state pump characteristics. Therefore, and due to very expensive tests, no transient LOBI pump tests have been performed at WCL-Hamilton. Additional steady-state tests in single-phase flow conditions have confirmed the results from previous tests at WCL-Hamilton and at the pump manufacturer ASTRÖ-Graz, Austria.

An R&D contract has been concluded with WCL-Hamilton for performing LOBI discharge nozzle calibration tests in two-phase flow conditions. The tests will be started in August 1979.

Feasibility studies have been initiated for the design of the high pressure injection system (HPIS) to be added to the LOBI facility, and for modifications of the LOBI secondary loop in order to allow more reactor-typical operation for all types of break simulation, especially under small leak LOCA conditions.

For the purpose of the LOBI test documentation by "Quick Look Reports" (QLR) and "Experimental Data Reports" (EDR), a QLR prototype has been prepared; its practical applicability will be checked during the shake-down tests.

Results from test calculations with the recently implemented RELAP4/MOD6 code show good agreement with those contained on the NEA tape.

The interactive version of the retrieval program is completed and allows the graphical display of all quantities of the data base.

The ASTEM program package has been completed and allows the calculation and display of about 3o different physical properties of water/vapor for given input data like pressure/temperature, density/temperature or saturation quantities.

The "Working Group on LOBI Programme A" has defined the final test matrix A consisting of 6o tests distinguished in base line tests and variation tests. The whole test matrix has been subdivided in two parts, A1 and A2, of 3o tests each.

This WG has established two heating power time curves to be used for controlling the heating power input to the rod bundle during blowdown: an upper limit curve for the pre-DNB range and a lower limit curve for the post-DNB range.

The "Working Group on LOBI Programme B" has discussed and compared the results from survey calculations of different members. An interesting result was that for a pipe rupture on the pump suction side the core temperature was much higher in case of pump rotor blocked than in case of continued pump operation at constant speed.

Further calculations for large breaks and all three rupture locations are aimed to determine the influence of downcomer gap width on core temperature and mass flow and on break mass flow.

The French and German Experts of this WG have presented the recently developed special versions of their RELAP4 codes for calculating small leak LOCAs and shown first results. The "WG LOBI B" will have strongly to rely on the availability of these code versions, or at least on the contribution of these members, for the future survey calculations for small leak LOCAs.

The stay of a LOBI staff member at EG&G Idaho Inc., USA, led to the participation in development work of RELAP4/MOD7, [2], [3].

Based on results from DAPSY code (GRS-Munich) and STRUDL II code (Design Office of J.R.C.-Ispra) calculations concerning dynamic loads on the LOBI test facility structure during blowdown, several supplementary shock absorbers have to be installed in various locations to further support the respective components of the test facility. These calculation results had been qualitatively confirmed by some acceleration measurements performed during the first blowdown test SD-o1.

During the last commissioning test, pressure drop measurements within the primary loops of the LOBI facility, and steam generator calibrations for steady-state mass flow measurements have been performed in single-phase flow conditions and at different mass flow rates: the measured Δp-values have been smaller than those from the design calculations [4].

Failures of the measurement instrumentation system which occurred during the commissioning and the shake-down tests have been analyzed and removed.

During the interrupt period of the shake-down test programme, the measurement instrumentation at the various locations of the test facility has been finally completed and is shown in Fig. 3 and 4. After mounting of the first experimental heater rod bundle, control measurements of the heater rod thermocouple isolation resistance has been performed and shown that 183 out of 188 TCs are still operational.

The signal transfer which operated well until test SD-o1 with reduced instrumentation, has shown some interference problems during test SD-o2 with complete instrumentation.

The mini-computer program for an interactive data evaluation has been tested and allows

- the one- und multi-dimensional graphical display of measured results
- the overlay of several curves
- the digital filtering and the production of histograms.

Data processing and formatting programs allow

- the conversion of data from PCM format into engineering units and their evaluation and representation
- the input of the measured results into the data bank.

Process control programs (heating power, pump speed) have been developed and tested.

A schamtical representation of the LOBI data acquisition and process control system (DAPCS) is shown in Fig. 5.

5. Next Steps

- Completion of test facility shake-down tests (SD-03, -04) ending up with a consistency and reproducibility check of the test results
- Starting the official LOBI experimental programme with test A1-01, the LOBI PREX test
- Performing pre-test prediction calculations, test results evaluation and documentation
- Performing LOBI PREX calculations and results comparisons by producing overlay plots; PREX 2nd Workshop meeting at Ispra
- Evaluation of LOBI pump test results and preparation of two-phase flow pump characteristics
- Starting the LOBI discharge nozzle calibration tests.

6. Relation to Other Projects

- Semiscale MOD3 project of USNRC at INEL, Idaho Falls, USA
- LOFT project of USNRC at INEL, Idaho Falls, USA
- PKL project of KWU at Erlangen, FRG
- ROSA II project of JAERI at Tokai-Mura, Japan.

7. Reference Documents

- Quarterly Reports of 1978 and 1979, GRS-FRG
- J.R.C.-Ispra Establishment, Programme Progress Reports Reactor Safety, January - June 1978, July - December 1978, January - June 1979
- [1] W. Riebold, W. Kolar, M. Larsen, E. Ohlmer, L. Piplies, H. Städtke:

  Specifications - LOBI-Pre-Prediction Exercise - Influence of PWR Primary Loops on Blowdown (LOBI)

  Technical Note Nr. I.06.01.79.25, February 1979

- [2] S.R. Fisher, H. Chow, G. Van Arsdall (EG&G Idaho Inc.), H. Städtke (J.R.C.-Ispra):

Development of a Non-Equilibrium ECC Mixing Model for Use in RELAP4/MOD7

Second Multi-Phase Flow and Heat Transfer Symposium Workshop, University of Miami, Miami Beach, April 16-18, 1979

- [3] S.R. Fisher, H. Chow, G. Van Arsdall (EG&G Idaho Inc.), H. Städtke (J.R.C.-Ispra):

Implementation of a Non-Equilibrium Model in RELAP4/MOD7

ANS Topical Meeting: Computional Methods in Nuclear Engineering, April 23-25, 1979

- [4] E. Ohlmer, W. Schulze:

Druckverlustmessungen an der LOBI-Anlage

Technical Note Nr. I.06.01.79.04, January 1979

8. Degree of Availability

- Quarterly Reports: from GRS-Köln, Glockengasse 2, 5 Köln 1
- Programme Progress Reports: restricted distribution list
- Conference Papers: from authors
- Technical Notes: restricted distribution list.

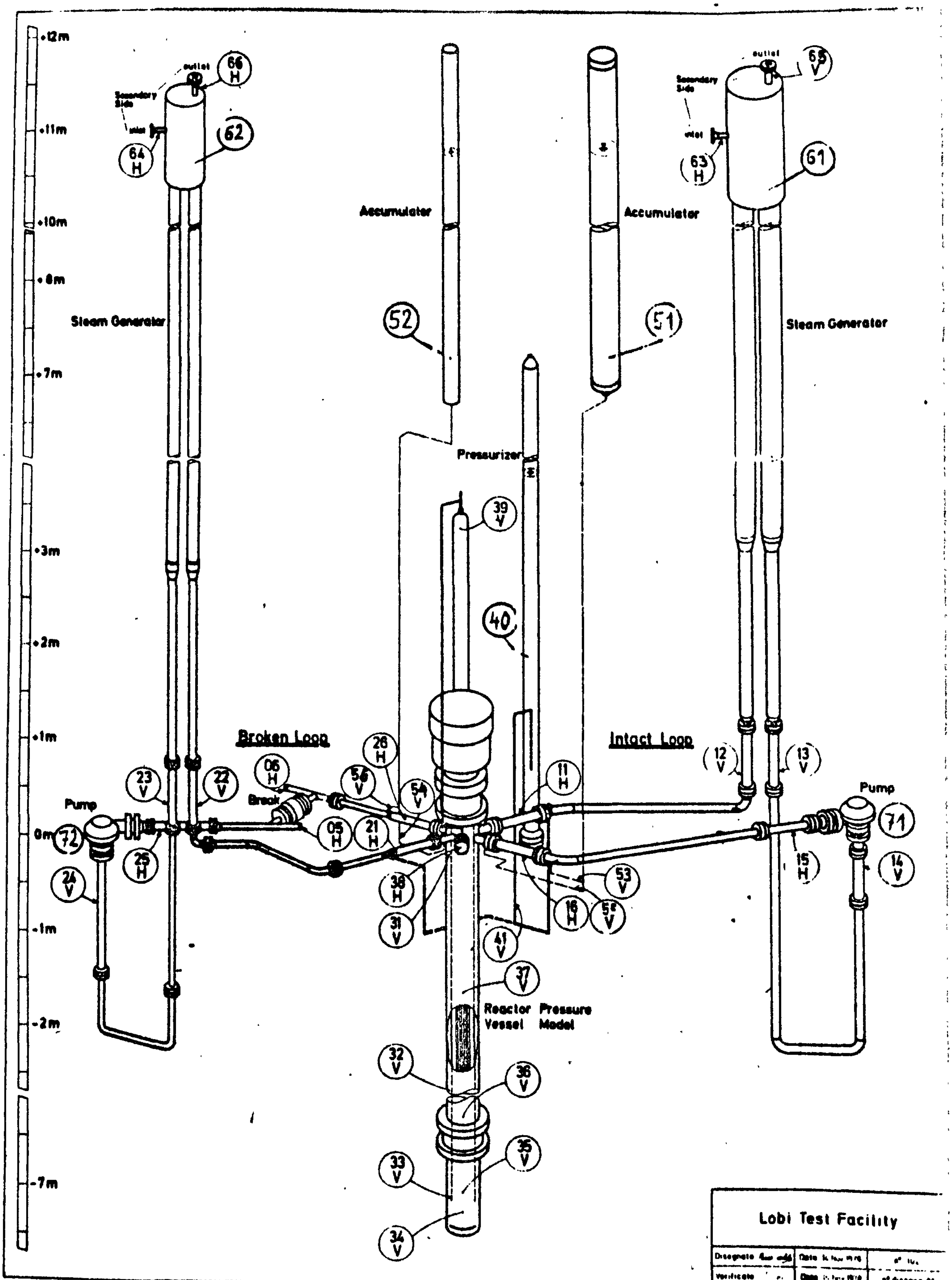
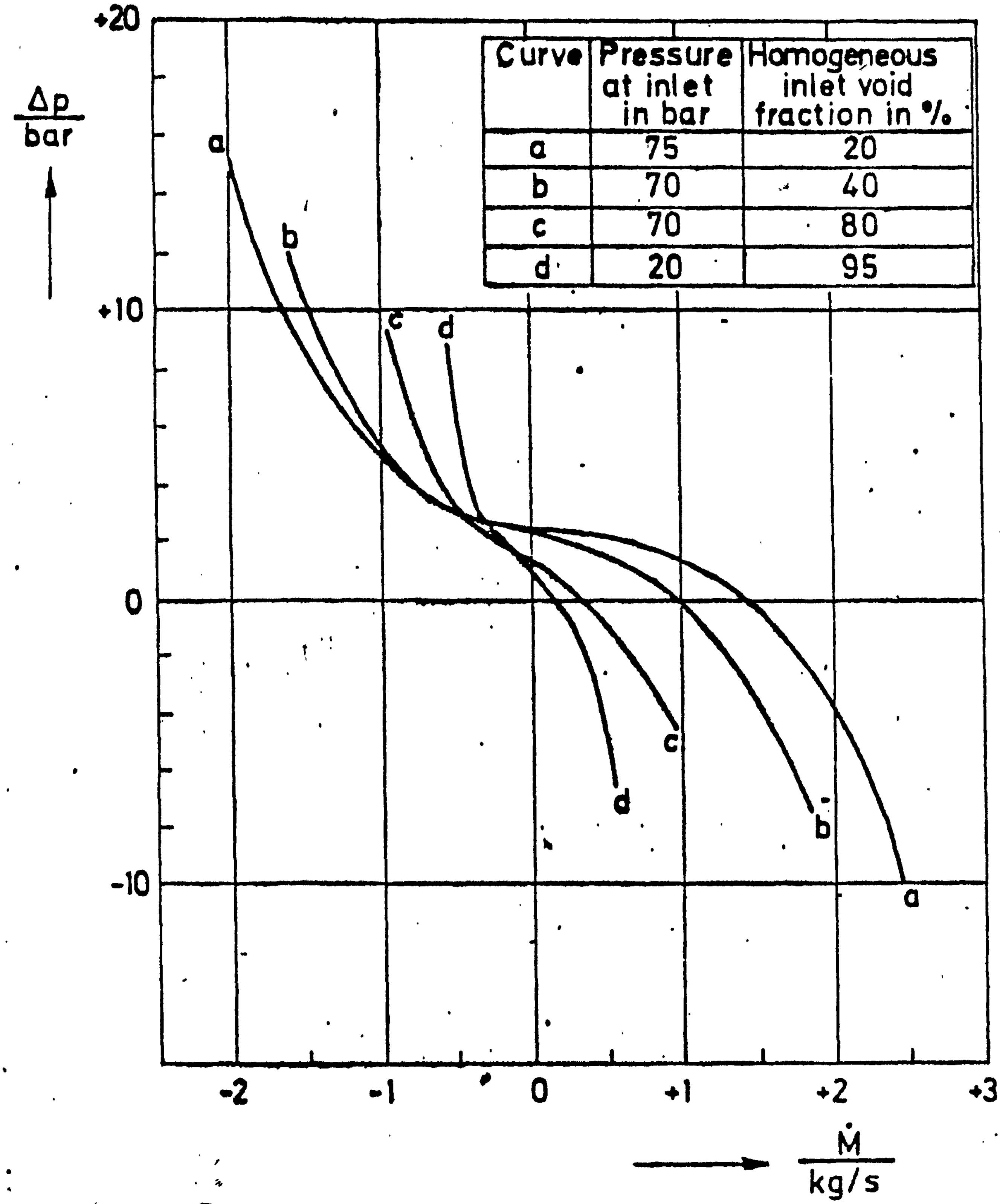
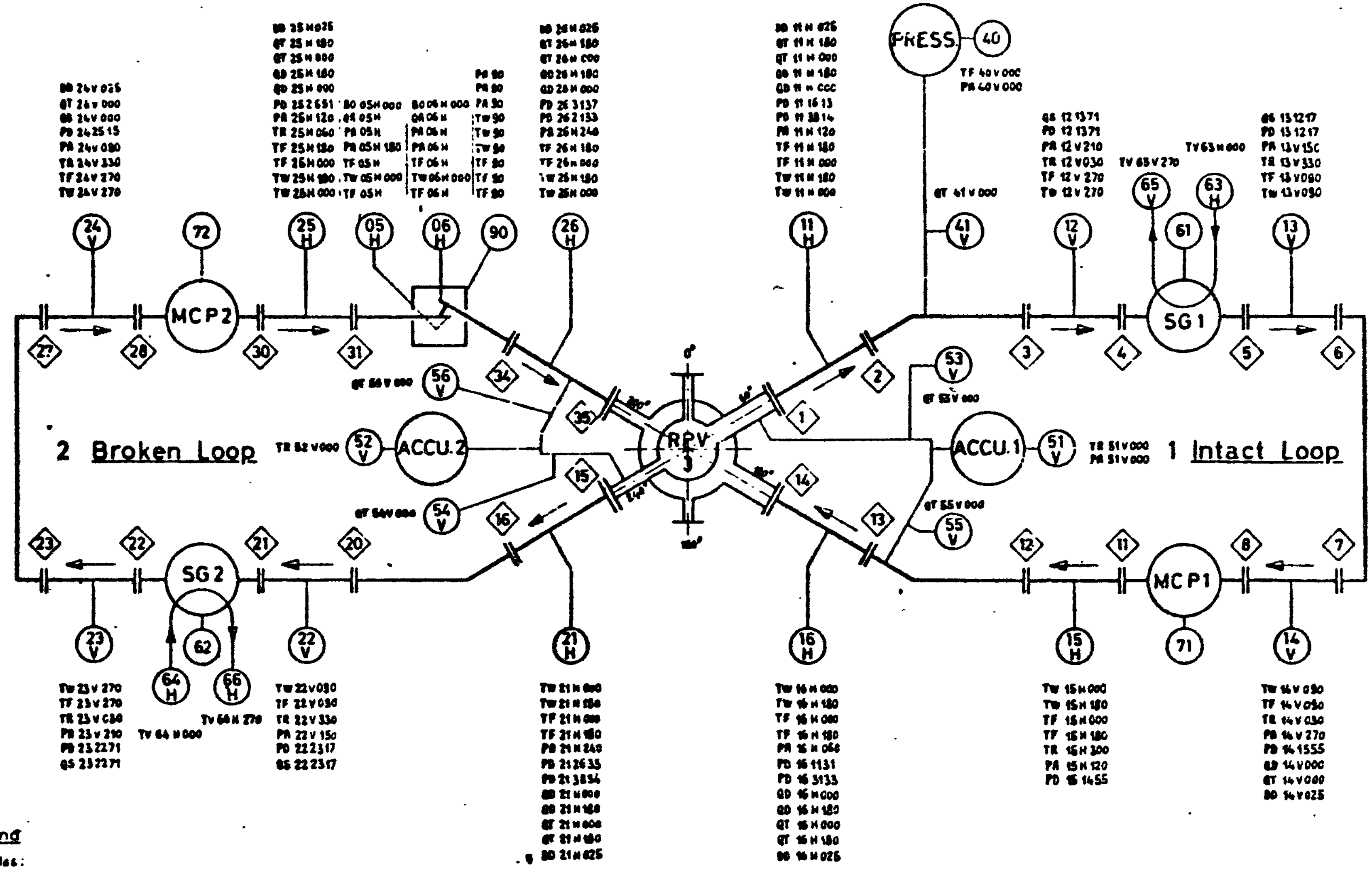


Fig. 1 : LOBI Test Facility for Cold Leg Break Configuration with Measurement Locations



Curve	Pressure at inlet in bar	Homogeneous inlet void fraction in %
a	75	20
b	70	40
c	70	80
d	20	95

Fig. 2 : Δp across the Lobi Pump at different inlet conditions for a constant speed of $n = +3250\ min^{-1}$



Legend

Examples:

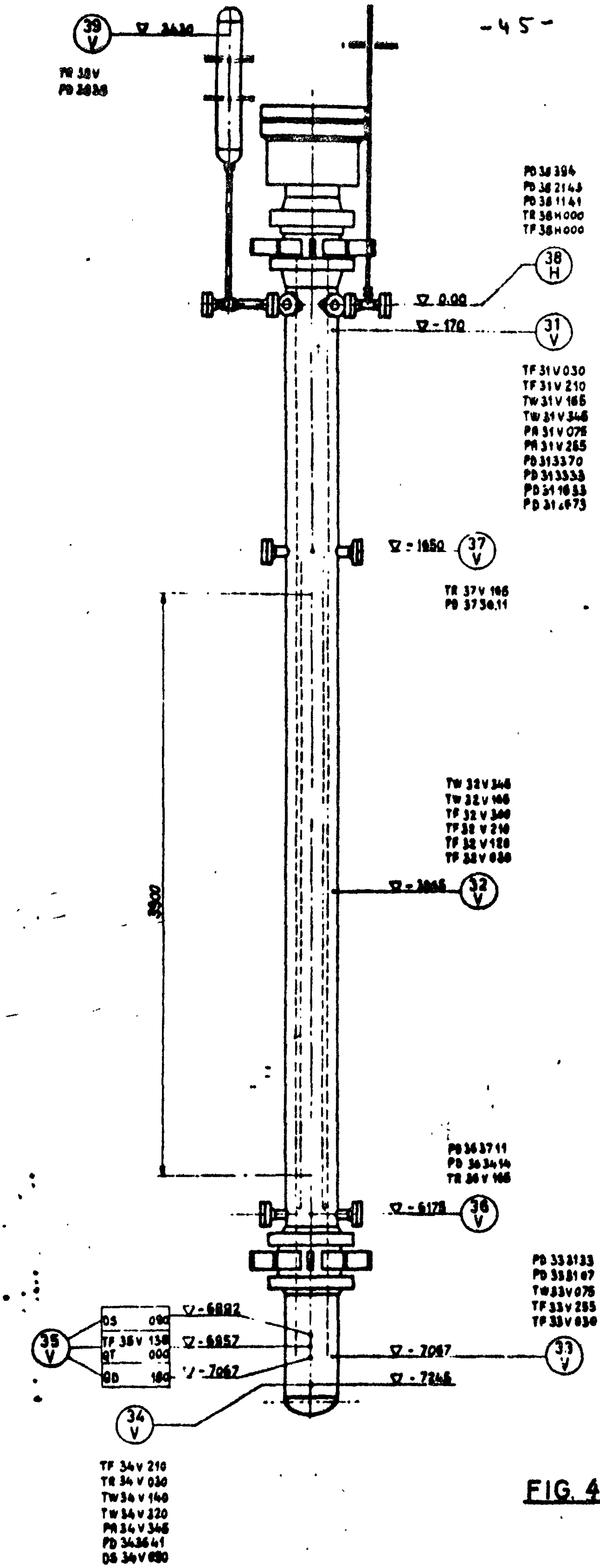
(21 H) measuring position 21 (measurement insert Nr 21)
orientation of measurement insert (horizontal or vertical)

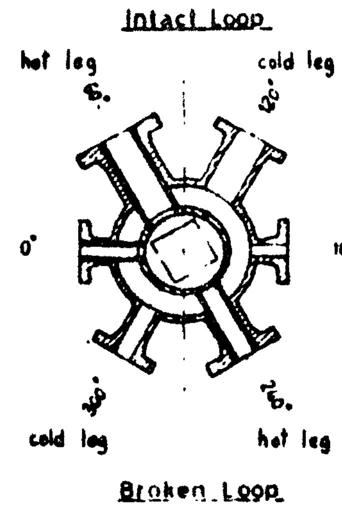
TW 16 H 180
Quantity Meas insert Angle

EXPLANATION OF SYMBOLS

TW	TEMPERATURE, WALL	QD	FLOW, DRAG BODY
TF	TEMPERATURE, FLUID (TC)	QT	" TURBINEMETER
TR	TEMPERATURE (Res Th)	QA	" APERTURE DIFFERENT. PRESSURE
PA	PRESSURE, ABSOLUTE	QS	FLOW, SG-DIFFERENT PRESSURE
PD	PRESSURE, DIFFERENT	DD	DENSITY 2 BEAM

FIG. 3 LOBI
Instrumentation of Intact and Broken Loop





EXPLANATION OF SYMBOLS

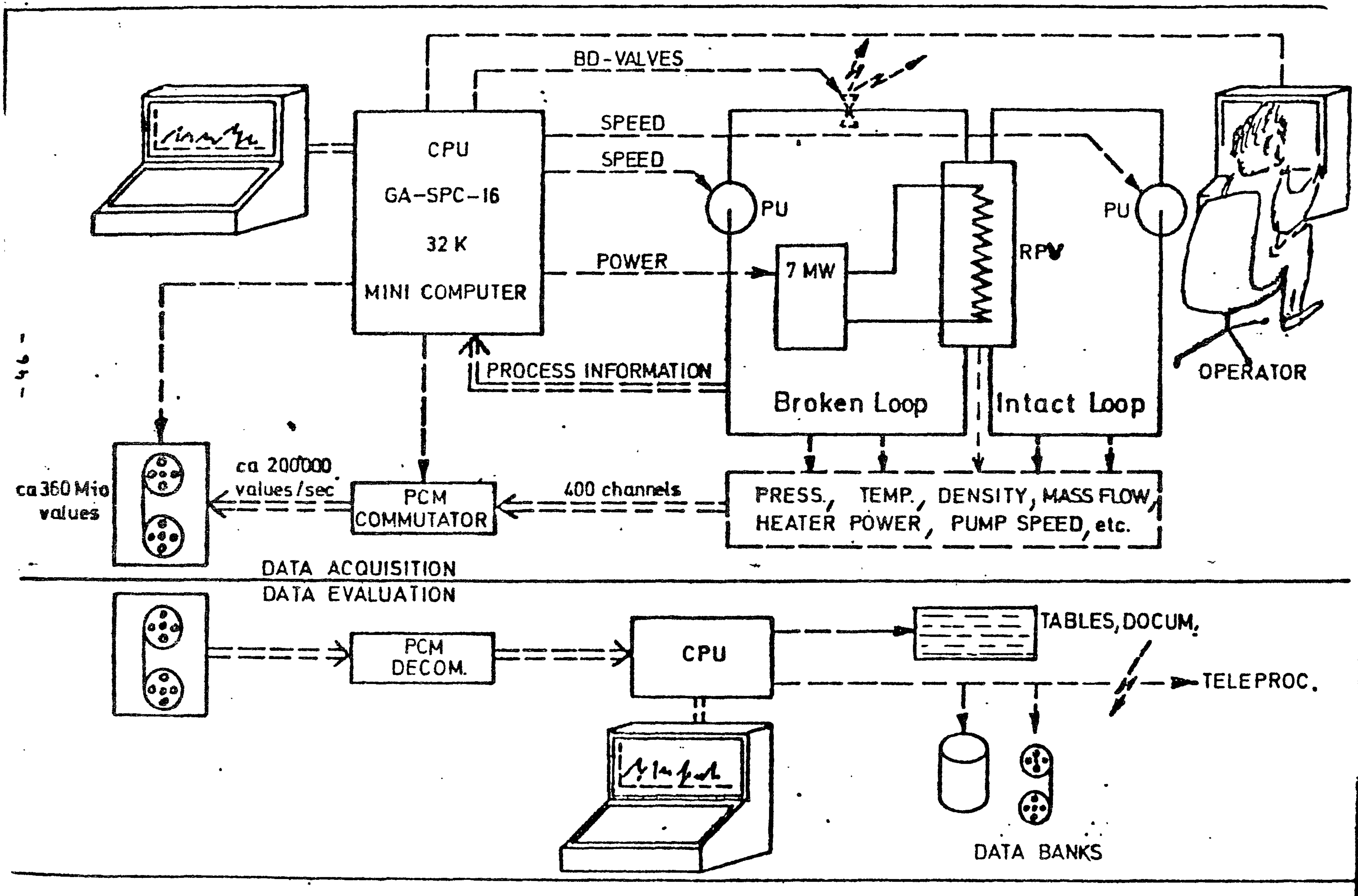
TW . TEMPERATURE WALL
TF · TEMPERATURE FLUID (THERMOCOUPLE)
TR : TEMPERATURE FLUID (RESISTANCE THERMOMETER)
PA . PRESSURE ABSOLUTE
PD . PRESSURE DIFFERENTIAL
GD : FLOW , DRAG - BODY
GT · FLOW TURBINEMETER
DS · DENSITY 1 BEAM

Legend
Examples·

31 V — measuring position 31 (measurement insert Nc 31)
orientation of measurement insert (vertical or horizontal)

TW	31 V	210
Quantity	Meas Insert	Angle

FIG. 4 LOBI Test Facility RPVM Instrumentation



BD-VALVES
CPU
GA-SPC-16
32 K
MINI COMPUTER
SPEED
SPEED
POWER
PU
7 MW
RPV
PU
OPERATOR
PROCESS INFORMATION
Broken Loop
Intact Loop
ca 360 Mio values
ca 200000 values/sec
PCM COMMUTATOR
400 channels
PRESS., TEMP., DENSITY, MASS FLOW, HEATER POWER, PUMP SPEED, etc.
DATA ACQUISITION
DATA EVALUATION
PCM DECOM.
CPU
TABLES, DOCUM.
TELEPROC.
DATA BANKS

Classification 1.1, 1.1.1, 1.1.2

Title 1	Two-phase four-quadrant pump behaviour	Country	U.K.
		Sponsor	C.E.G.B.
Title 2		Organisation	C.E.R.L.
Initiated 1977	Completed	Project Leaders	
Status Continuing	Last updating Aug 1978		

1. General Aim

To examine the behaviour of a pump under single and two-phase conditions to provide experimental support for pump models to be used in LOCA calculations.

2. Particular Objectives

To determine head and torque characteristics in all modes of pump operation (forward and reverse flow and impeller direction) for a full range of void fractions.

3. Experimental Facilities and Programme

An experimental pump of specific speed approximately 1000 (U.S. units) is being tested using Freon 12 as a working fluid. The rated conditions are flow of 5 ℓs^{-1} at a head of 20 m. Impeller diameter is 180 mm. Inlet and exit voidage, pressure and temperature, mass flow, and pump power and speed will be measured. Pressures will range up to 10 bar (equivalent to $\sim$ 60 bar in water).

4. Project Status

Single-phase characteristics have been measured. Two-phase testing has been carried out for a limited range of void fractions.

CLASSIFICATION 1.1.1

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	1.1.1	RS 016 B
Vorhaben/Project Title Untersuchung der Vorgänge bei der Druckentlastung wassergekühlter Reaktoren. Modellversuche mit einem 11,2 m hohen Stahlbehälter mit Einbauten. Investigation into the Phenomena Involved in the Depressurization of Water-Cooled Reactors. Experiments Using a Steel Vessel 11.2 m in Height with Internals.		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor BATTELLE-INSTITUT E.V. Frankfurt am Main
Arbeitsbeginn/Initiated July 15, 1972	Arbeitsende/Completed April 30, 1979	Leiter des Vorhabens/Project Leader Dr. T.F. Kanzleiter
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

The experimental blowdown program is aimed at integral large-scale experimental simulations of loss-of-coolant accidents in water-cooled reactors of PWR and BWR type. All experimental results are to be compared with the results of model calculations to show the applicability of the computer codes used and, if possible, to improve them.

2. Particular Objectives

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

3. Research Program

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

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

3.3. BWR experiments with internals of BWR type.

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

4. Experimental Facilities, Computer Codes

The experimental facilities consist essentially of

- a pressure vessel (5.2 m^3, 140 bar, 300 °C) with an electric heater (600 kW max.)
- PWR internals with flexible and inflexible components
- BWR internals
- a loop with piping, pumps and heat exchangers to realize the same differences in enthalpy as inside an original vessel
- measuring instruments (approx. 110 channels for pressure, differential pressure, temperature, density, mass flow, force, strain, acceleration, displacement and water level
- a data collection and processing system with 120 channels and a threshold frequency of 5 kHz.

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

5. Progress to Date

Ad 3.1 Four preliminary BWR and PWR experiments without internals (Nos. SWR 1R, SWR 2R, DWR 1R, DWR 2R) were carried out in 1974 and 1976. Experiment SWR 2R has been chosen as an OECD standard problem (No. 6) for an international comparison with theoretical results.

Ad 3.2 Seven PWR experiments with flexible internals (Nos. DWR 1 to 5, DWR 2L and DWR 5A) were carried out in 1975. Thermal-hydraulic and structural-dynamic evaluation of these experiments is still in progress.

Ad 3.3 Eleven BWR experiments with internals were planned and prepared in 1976 through 1978. Attempts were made to improve the differential pressure transducers that had previously shown unsatisfactory dynamic characteristics (s. Annual Report 1977).

6. Results

Ad 3.3 The further development of differential pressure transducers for direct installation inside the pressure vessel did not lead to acceptable dynamic characteristics of the transducer. Alternative types of differential pressure transducers, also suitable for direct installation inside the vessel are not available. Therefore, the specified differential pressure measuring system had to be replaced by a combination of externally installed differential pressure transducers with pressure lines leading to the measuring points inside the pressure vessel (low system eigenfrequency but good accuracy) and absolute pressure transducers (low accuracy of pressure differentials but good dynamic response). As a result of this replacement, the pressure vessel internals have to be remodelled and the experimental program has to be revised. At present in the experimental BWR program major emphasis is placed on the investigation of the phenomena in the case of steam line breaks, such as

- liquid level behavior
- void distribution in the rising two phase mixture
- thermodynamic nonequilibrium and flashing effects
- mass and enthalpy flow at the site of rupture
- forces acting on the internals.

Additional objectives in the case of feed water line breaks are

- pressure wave and pressure pulsation effects.

The experimental program comprises experiments involving different initial conditions (subcooling, steam quality) and different break areas and break locations (simulating BWR feed water line breaks, BWR steam line breaks and PWR steam generator steam line breaks).

The experimental parameters have to be specified in detail on the basis of preliminary calculations carried out by KWU and Battelle (Project RS 312).

7. Next Steps

Ad 3.2 - Completion of the evaluation of PWR experiments, Part I

Ad 3.3 - Measurement of the maximum initial subcooling and steam quality in the pressure vessel attainable by means of an improved external loop

- Final specification of the experimental program
- Preparation and performance of the BWR experiments

Ad 3.4 - Preparation of additional PWR experiments

8. Relation to Other Projects

RS 132: Pre- and Post-Test Analysis of the BWR-series Experiments of Research Project RS 016 B

9. References

(1-4) Quarterly Reports in the Series "GRS-Fortschrittsbericht. Bericht über die vom Bundesministerium für Forschung und Technologie geförderten Forschungsvorhaben auf dem Gebiet der Reaktorsicherheit" (in German)

- January to March 1978
- April to June 1978
- July to September 1978
- October to December 1978

(5) BF-RS16B-40-4-3, "Revidiertes SWR-Versuchsprogramm" April 1978

(6) B. Hölzer, T. Kanzleiter, F. Steinhoff, "Specification of OECD Standard Problem No. 6. Determination of Water Level and Phase Separation Effects During the Initial Blowdown Phase". GRS, Garching, February 1977

(7) CSNI-Report No. 30
W. Winkler, "Comparison Report on OECD-CSNI Standard Problem No. 6". GRS, Garching, August 1978

10. Degree of Availability of the Reports

Reports are available through GRS-FB.

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

Berichtszeitraum/Period 1.1.78 - 31.12.78	Klassifikation/Class 1.1.1	Kennzeichen/Project Number RS 312
Vorhaben/Project Title Begleitende Theoretische Arbeiten zu den SWR-Versuchen des Forschungsvorhabens RS 0016B Pre and Post Test Analysis of the BWR-Series Experiments of Research Project RS 0016B		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor BATTELLE-INSTITUT E.V. Frankfurt am Main Abt. Thermische Verfahrenstechnik
Arbeitsbeginn/Initiated 1.11.77	Arbeitsende/Completed 31.10.79	Leiter des Vorhabens/Project Leader M. Müller
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

The realization of the experiments RS 0016B BWR-series has to be supported by theoretical analysis (test predictions and post-test analysis). Especially experimental problems, arising during execution, are beeing analysed in detail to influence the planning of further experiments.

2. Particular Objectives

Analysis of the results of the RS0016B BWR-series experiments to evaluate experimental data for the verification of the computer code RELAP4-MOD5.

3. Research Program

3.1 Test prediction: checking of instrumentation requirements and of the measurement ranges by calculating the transient temperature-, pressure-, and massflow lapse.

3.2 Quick-Look: Immediately after an experiment a post-test calculation is done using the initial and boundary conditions measured, to check wether the experimental aim is fullfilled and which modifications are required for the following experiments.

3.3 Detailed analysis: Disagreement of experiment and calculation are considered in detail by variation of parameters and models. Mainly the following aspects are examined:
- Break flow model
- Phase separation model
- Flashing effects and Liquid Level
- Nodalisation

The results will be published in a final report.

4. Experimental Facilities, Computer Codes

To evaluate the experimental data the US-NRC Code RELAP4-MOD5 was implemented.

5. Progress to Date

Parameter and sensibility studies were performed to fix the experiment matrix and to establish the measurement ranges depending an initial- and boundary conditions.

6. Results

Initial subcooling in the downcomer and void in the core region causes a sharper pressure gradient at the beginning of blow-down compared with only saturation conditions and void zero.

A sieve plate in the steam dome (resistance like steam dryers) does not affect much the integral blow-down.

The system loop to establish initial temperature distribution has little influence on transient blow-down behaviour.

If heat transfer from the walls to the fluid is considered, pressure lapse is about 10 % higher than without, with respect to the energy added. The mixture level stagnates below the break location, when break diameters of about 28-35 mm are chosen. The unvertainty is caused by the inability of RELAP4-MOD5 to calculate mixture level behaviour of stacked volumes in connection with phase separation. Using the co-countercurrent flow option of MOD5 does not solve the stacking problem.

7. Next Steps

The results calculated with RELAP4-MOD5 will be compared with LAMB calculations from KWU

Modifications of RELAP4-MOD5 are planned to improve the ability of liquid level calculations.

8. Relation to other Projects

RS 0016B SWR

9. References

Planned: RELAP4-MOD5 Misture level calculation

10. Degree of Availability of the Reports

Available through GRS-FB

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	1.1.1	RS 263/2
Vorhaben/Project Title Analytische Tätigkeiten der GRS im Rahmen des Reaktorsicherheitsforschungsprogramms des BMFT Kräfte auf Kerneinbauten Analytical activities of the GRS in the frame of the BMFT research program on reactor safety Forces on Core Internals		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH
Arbeitsbeginn/Initiated 1.2.1977	Arbeitsende/Completed 31.12.1979	Leiter des Vorhabens/Project Leader DI T. Grillenberger
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 31.12.1978	Bewilligte Mittel/Funds

1. General Aim

Parallel and complemantary analytical work within the experimental research programs promoted by the BMFT, and further development of computer codes in the field of reactor safety.

2. Particular Objectives

Analytical investigations to accompany the research project RS 16B "Forces on Core Internals".

3. Research Program

3.1 Post-test calculation of the BWR reference test.

3.2 Evaluation of the experimental results of the PWR quick program.

3.3 Pre-test calculation of the BWR specific test with internals.

3.4 Post-test calculations and evaluations of the BWR tests with internals.

3.5 Pre-investigations concerning the PWR specific tests with internals.

3.6 Post-test calculations and evaluations of the PWR tests.

4. Test facilities, computer codes

DAPSY, simulation of multidimensional pressure wave propagation. DRUFAN, a lumped parameter model for the simulation of blowdown.

5. Progress to Date

The evaluation of the pressure wave phenomena in the PWR quick program has been finished. The blowdown phase of the test DWR5 has been calculated with DRUFAN. The results show a good agreement with the experimental data for pressure and temperature as well as for the pattern of the mass flow rate. Only the amplitude of the mass flow differs significantly. A check of the total mass shows that the measured values are too large. Since the results in the very beginning of the blowdown agree well, it can be assumed that the mass flow is overestimated by the drag body in the case of high void fraction and large Mach number. Therefore a corrective function depending on these parameters has been developed, which improves the interpretation of the measured data.

6. Results

The evaluation of the DWR5 test in the pressure wave phase and in the blowdown phase is documented in two reports /1, 2/, dealing with the post-test calculations with DAPSY and DRUFAN. It is shown that the test targets are reached sufficiently and the results are very useful for the code verification.

8. Relation to other Projects

The RS 16 DWR5 calculations have been done in close connection with the DRUFAN development, so that the results of this work can also be seen as a verification of DRUFAN.

9. References

/1/ M. Hrubisko: Nachrechnung und Auswertung der Kurzzeitphase des Battelle-Blowdown-Versuchs DWR5 (RS 16/B) mit dem Programm DAPSY. GRS-A-245, Dezember 1978

/2/ M.J. Burwell, M. Hrubisko: Nachrechnung und Auswertung der Langzeitphase des Battelle-Blowdown-Versuchs DWR5 (RS 16/B) mit dem Programm DRUFAN. GRS-A-246, Dezember 1978

10. Degree of Availability of the Reports

GRS-A-... reports are available through the Gesellschaft für Reaktorsicherheit (GRS) mbH, Forschungsbetreuung, Glockengasse 2, D-5000 Köln 1.

Berichtszeitraum/Period	Klassifikation/Category	Kennzeichen/Project Number
1.1.78-31.12.78	1.1.1	06.01.01/01A (PNS 4115)

Vorhaben/Project Title	Land/Country: FRG; Fördernde Institution/Sponsor: BMFT; Auftragnehmer/Contractor: Kernforschungszentrum Karlsruhe, Projekt Nukleare Sicherheit IRE
Auslegung, Vorausberechnung und Auswertung der HDR-Blowdown-Experimente zur dynamischen Belastung und Beanspruchung von Reaktordruckbehältereinbauten. Design, precomputation and evaluation of the HDR-blowdown experiments on dynamic loadings and deformations of reactor-pressure-vessel internals.	

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1978	1982	Dr.Krieg/Dr.Schumann
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

Remark: The previous project PNS 4221/PNS 4223 has been restructured. Since Jan. 1st, 1978, it is continued by the projects PNS 4115, PNS 4116, PNS 4125, and PNS 4126.

1. General Aim

Development and verification of computer codes for analysis of the dynamic response of reactor vessel internals.

2. Particular Objectives

Design, precomputation and evaluation of the HDR-blowdown-experiments on dynamical loadings and deformations of reactor pressure-vessel internals.

3. Research Program

Conception of the experimental program, design of the test facility, computational simulation prior to and after the tests.

4. Experimental Facilities, Computer Codes

Experiments carried through at the former HDR-reactor:
Computer codes as developed in PNS 4125 and the codes HDRNA and TURBIT-3 for analysis of the temperature distribution; hydraulic mechanism for snap-back tests.

5.+6. Progress to Date and Results

- The experiments on the temperature distribution have been performed and analyzed. The main results are: 1) The temperature distribution is spatially stable; 2) the temperature can be reproduced within ± 3 K; 3) after switching off the heating circuit the temperature distribution changes quickly so that the blowdown must be triggered within 30 s, at least, thereafter.

- By means of the code TURBIT-3 the turbulent decay of temperature differences in the radial direction in the downcomer has been described accounting for buoyancy forces. It was found that any disturbances of the temperature distribution are damped out very fast.

- Using the 3D-code FLUX2 (compressible fluid) the maximum accelerations, deformations and pressure differences as to be expected in the blowdown-experiments have been precomputed. The maximum values for the reference case are: pressure difference 13 bar, deformation 1 mm, acceleration 1200 m/s^2.

- The snapback-experiments have been further prepared. The hydráulic deflection device has been reconstructed. It can be used at two positions. The instrumentation has been ordered.

- By means of the 2-1/2 dimensional code STRUYA the consequences of small asymmetries in the core barrel foundation and disturbances produced by water sucking-off devices in the downcomer have been investigated. It has been found that these disturbances have small effects on the blowdown pressure field.

7. Next Steps

The snapback-tests and a preliminary blowdown-test are due for 1979. The snapback-tests will be documented so that precomputations can be prepared by different institutions and compared.

8. Relation with other Projects

The project is closely related to PNS 4116, 4125, 4126. Further, it is coordinated with all other projects of the HDR blowdown program.

9. References

GRÖTZBACH, G.: Numerische Untersuchung der Quervermischung bei auftriebsbeeinflußter turbulenter Konvektion in einem vertikalen Kanal. KFK 2648 (1978)

SCHUMANN, U.; MÖSINGER, M.; SCHNAUDER, H.: Ergebnisse des Temperaturschichtversuches als Vorversuch zu den Experimenten mit RDB-Einbauten. 2. Statusbericht des Projektes HDR-Sicherheitsprogramm des Kernforschungszentrums Karlsruhe, Karlsruhe, 24. Oktober 1978. S.3/58-3/82.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
Jan.1,1978-Dec.31,1978	1.1.1	06.01.01/02A (PNS 4116)
Vorhaben/Project Title Meßtechnische Erfassung und Auswertung des dynamischen Verhaltens der Versuchseinbauten im Reaktordruckbehälter (RDB) des HDR im Rahmen der HDR-Blowdown-Versuche Experimental Data Acquisition and Processing of the Dynamic Behavior of the Pressure Vessel Test Internals in the HDR-Blowdown-Experiments		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe GmbH Projekt Nukleare Sicherheit IRE
Arbeitsbeginn/Initiated Jan.1,1975	Arbeitsende/Completed 1979	Leiter des Vorhabens/Project Leader K.D. Appelt
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

The presently available methods and computer codes allow to design the LWRs such that an accident caused by pipe rupture in the primary circuit can be safely managed. However, the extent of exploitation of the structural load carrying capability is not yet known. To be able to make a detailed determination of existing safety reserves, it is necessary to replace the conservative computer models by realistic models and computer codes which, however, have to be verified by relevant experiments.

2. Particular Objectives

Recording the dynamic deformations and stresses of the pressure vessel internals is the particular objective of the anticipated investigations. To get realistic results, the phenomenon of fluid-structure interaction has to be taken into account, i.e. the feedback of structural flexibility upon the pressure field imposed. Consequently, both the variables of state in the fluid and the dynamic response of the loaded structure must be recorded simultaneously by measurement technology. It is the objective of this task to measure the structure response of reactor pressure vessel internals (deflections, accelerations, strains) occurring during a blowdown and to evaluate the experimental data.

Experience gathered from other similar projects has shown that the instrumentation used must satisfy very stringent quality requirements if intolerable failures and falsifications of experimental results are to be avoided.

3. Research Program

The experimental program includes the development, testing and provision of the instrumentation for the planned large-scale experiments as well as the measurements and evaluation. It is divided as following:

3.1 Laboratory basic tests of prototype transducers (recording of characteristic data).

3.2 Thermohydraulic and mechanical tests of the prototype transducers in the autoclave system.

3.3 Tests on the dynamic behavior of the prototype transducers (with different transducer supports).

3.4 Dynamic calibration of the prototype measurement chains (recording of the transfer function).

3.5 Qualification and ordering of instrumentation for the HDR.

3.6 Recording the characteristic data for the transducers by laboratory basic tests and dynamic calibration of the original measurement chains prior to their installation in the reactor pressure vessel of the HDR.

3.7 Installation of the instrumentation in the reactor pressure vessel of HDR including performance tests and acceptance.

3.8 Conduct of tests and acquisition of measuring data.

3.9 Evaluation of measured data.

4. Experimental Facilities, Computer Codes

True-scale experiments not connected with problems of modeling belong to the most significant studies indicated above. Therefore, the (decommissioned) HDR reactor, modified into an almost true-scale PWR, has been used as the suitable experimental facility. Details can be found in the Annual Report 1976 of the PNS 4221 task.

To develop and study the instrumentation a "facility for the dynamic study of transducers under blowdown conditions in the reactor pressure vessel" was erected which consists of an electrodynamic shaker, an autoclav system, and an auxiliary device allowing dynamic investigations of the transducers in air and water at ambient temperature. Besides, small auxiliary devices have been used for various special investigations.

5. Work Completed

Qualification tests of nine accelerometer prototypes, two displacement transducer prototypes /and five pressure transducer prototypes were perfomed in the autoclave-shaker test facility. Development of facilities for high-temperature dynamic calibration of accelerometers and pressure transducers was completed and the facilities have started operation. The primary reports on the qualification tests of one accelerometer prototype and one displacement transducer prototype were completed.

6. Essential Results

Transducer defects found in previous qualification tests have been widely eliminated. Two accelerometer models and two displacement transducer models qualified as suitable for the performance of the RDB-E-blowdown tests. So far, neither the absolute pressure transducer prototypes nor the differential pressure transducer prototypes qualified as suitable.

7. Plans for the Near Future

Ordering of accelerometers, displacement transducers and strain gauges for the RDB-E-blowdown tests; provision of corresponding calibration facilities. Conclusion of qualification tests of pressure transducer prototypes and possibly ordering of absolute and differential pressure transducers for the RDB-E-blowdown tests.

8. Relation with Other Projects

The data on structural response of the reactor pressure vessel test internals under blowdown conditions, which are measured and recorded in the experiments, serve as the input for the code developments made under the PNS 4223 task to treat the dynamic load of reactor internals, taking into consideration the backfeeding of structure and fluid.

9. Literature

PNS-Halbjahresbericht 1978/I, KFK 2700, Sept. 1978

10. Degree of Availability

Unrestricted distribution, Kernforschungszentrum Karlsruhe, Literaturabteilung

Berichtszeitraum/Period	Klassifikation/Cl.	Kennzeichen/Project Number
1.1.78 - 31.12.78	1.1.1	06.01.02/01A (PNS 4125)

Vorhaben/Project Title	
Weiterentwicklung und Verifizierung fluid-strukturdynamischer Codes zur Beanspruchung von RDB-Einbauten beim Blowdown. Development and Verification of Codes in Coupled Fluid-Structural Dynamics for Stress Analysis of Reactor Vessel Internals Under Blowdown Loading.	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Projekt Nukleare Sicherheit Kernforschungszentrum Karlsruhe IRE

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1974	1982	Dr. Krieg/Dr. Schlechtendahl
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	Dec. 78	

Remark: The previous project PNS 4221/PNS 4223 has been restructured. Since Jan. 1st, 1978, it is continued by the projects PNS 4115, PNS 4125 and PNS 4126.

1. General Aim

Development and verification of computer codes for analysis of the dynamic response of reactor vessel internals.

2. Particular Objectives

Simulation of the fluid- and structural dynamics inside the pressure vessel under a postulated failure of the primary coolant circuit close to the inlet nozzle. Experimental investigation by using the facility of the former HDR-reactor.

3. Research Program

Development of analytical methods and computer codes for fluid-structural dynamics.

4. Computer Codes

Codes in production:

YAQUIR, STRUYA	(2D fluid dynamics)
DRIX-2D	(2D non equilibrium two phase flow)
CYLDY2	(shell dynamics for the core barrel)
STRUYA/CYLDY2	(2D fluid dynamics coupled with CYLDY2)
FLUX 1	(3D fluid dynamics coupled with CYLDY2, incompressible Fluid)
FLUX 2	(3D fluid dynamics coupled with CYLDY2, compressible Fluid)
SING-S	(Boundary integral equation code for 3D fluid structure coupling)
FLUST	(fluid dynamics system code with 2D models)

Codes in development:

CYLDY3	(improved shell dynamics for the core barrel)
FLUST	(coupling with DRIX-2D and FLUX)

5.+6. Progress to Date and Results

After a number of model enhancements the two phase code DRIX-2D was used to investigate the local mass flow distribution in the blowdown nozzle. In a cooperative effort with the university of Karlsruhe the influence of model dimensionality in fluid dynamics codes has been studied and its importance has been assessed. Predictive calculations with the code FLUST for HDR blowdown experiments have been completed. Work has begun on implementing FLUX in FLUST.

With the code CYLDY3 which was recently developed for shell dynamics of the core barrel, eigenvalues (eigenfrequencies and mode shapes) of high accuracy has been obtained. Comparison with other results showed that, using finite elements, an adequate description of the shell bending close to the boundaries as well as a sufficient representation of high order mode shapes would be either extremely costly in terms of the number of elements or even impracticable.

With the STRUYA code the influence of numerical integration parameters was studied. Furthermore, the equipment built into the HDR downcomer for flow and temperature control purposes was found to have a negligible influence on blowdown. The model basis of the FLUX code was successfully checked against one-dimensional test cases for which analytical solutions can be obtained. Comparisons between FLUX-calculations and the RS16/2-experiments were satisfying, although a certain sensitivity against damping parameters was found. Work has begun on modeling of two fluid phases in FLUX. For a DWR case a strong influence upon the maximum stresses in the core barrel was found. Due to the differences in computer effort and stability behavior for the codes STRUYA and FLUX the conclusion was made, that for the analysis of the pressure vessel internals primarily the code FLUX will be used.

With the code SING-S for coupled fluid structural dynamics in any three-dimensional geometry parameter calculations has been carried through for the oscillations of the spherical containment of a boiling water reactor with pressure suppression system. For different collapsing steam bubbles in the water pool of the pressure suppression system the maximum shell elongations have been calculated. Furthermore, the influence of an additional not collapsing steam bubble in the water pool could be determined.

-3-

7. Next Steps

The drift flux theory will reach a stage of completion with documentation. Work on integral balance equations for fluid dynamics will continue.

On the basis of the high accuracy CYLDY3 results for the eigenvalues of the core barrel dynamics also the deformations and stresses versus time will be calculated. Coupling of FLUX and FLUST will be achieved. Efforts will be shifted from code development to HDR experiments (06.01.01/01A).

The stresses, resulting from the oscillations of the spherical boiling water reactor containment will be determined. Maximum bending stresses are expected at the interconnection between the spherical containment and the bottom cone of the condensation chamber.

8. Relations with other Projects

The project is closely related to PNS 4115, 4116 and 4126. Further, it is coordinated with all other projects of the HDR blowdown program.

9. References

U. Schumann: Dynamic of a nuclear-reactor shell-structure in an incompressible fluid. To appear in: U. Müller, K.G. Roesner, B. Schmidt (ed.): RECENT DEVELOPMENTS IN THEORETICAL AND EXPERIMENTAL FLUID MECHANICS-COMPRESSIBLE AND INCOMPRESSIBLE FLOWS. Springer Verlag (1978)

U. Schumann: Effektive Berechnung dreidimensionaler Fluid-Struktur-Wechselwirkung beim Kühlmittelverluststörfall eines Druckwasserreaktors - FLUX. KFK 2645, in Vorbereitung (1978)

H. Mösinger: Assessment of a drift-flux approximation for a strongly transient two-phase flow. Specialist's Meeting on Transient Two-phase Flow, Paris (1978)

R. Krieg: Coupled Problems in Transient Fluid and Structural Dynamics in Nuclear Engineering. Part I: Safety Problems by Flow Singularity Methods. Appl. Math. Modelling 2 (1978), 81-89

B. Göller, G. Hailfinger, R. Krieg: Vibrations of the Pressure Suppression System of a Boiling Water Reactor. Int. Conf. Vibration in Nucl. Plant, May 78, Keswick,UK

B. Göller, G. Hailfinger, R. Krieg: Schwingungen im Druckunterdrückungssystem von Siedewasserreaktoren mit flexiblem, wasserbeaufschlagten Kugel-Containment. Reaktortagung 1978, Hannover

G. Enderle, F. Katz, H. Mösinger, E.G. Schlechtendahl, K. Stölting: Belastung eines DWR-Kernmantels nach dem Bruch der Hauptkühlmittelleitung. Reaktortagung, Hannover, 4.-7. Apr. 1978, Deutsches Atomforum e.V., Kerntechnische Ges. im Dt. Atomforum e.V. Leopoldshafen 1978: ZAED, S. 222-25

G. Enderle, F. Katz, H. Mösinger, E.G. Schlechtendahl, K. Stölting: Pressure field and core barrel loadings during PWR-blowdown. Internat. Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, Oct. 16-19, 1978

F. Kedziur, N. Moussiopoulos, U. Schumann, K. Stölting: PWR-depressurization and its hydraulic analogy. Nucl. Engineering and Design, 47(1978, S. 25-34

R. Krieg, G. Hailfinger: SING 1 - Ein Computercode zur Berechnung transienter dreidimensionaler, inkompressibler Potentialströmungen nach einem Singularitätenverfahren. KFK 2505 (März 78)

R. Krieg: Coupled fluid structural dynamics in blowdown suppression systems: numerical schemes and applications. Lecture Notes for Advanced Course on Structural Dynamics, Ispra, Oct. 12, 1978

A. Ludwig, U. Schumann: Eigenschwingungen eines Druckwasser-Reaktor-Kernmantels in Vakuum und Wasser. Reaktortagung, Hannover, 4.-7. April 1978, Deutsches Atomforum e.V., Kerntechnische Ges. im Dt. Atomforum e.V. Leopoldshafen 1978: ZAED S. 130-33

E.G. Schlechtendahl: Status of code development in the Federal Republic of Germany concerning fluid-structural dynamic coupling during reactor transients. 6. Information Meeting on Water Reactor Safety Research, Gaithersburg, Md., Nov. 6-9, 1978

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.7.78 - 31.12.78	1.1.1	06.01.02/02A (PNS 4126)
Vorhaben/Project Title: Laborversuche zur Abstützung von fluid-strukturdynamischen Rechenprogrammen zur Beschreibung der Anfangsphase bei Kühlmittelverluststörfällen. Laboratory Experiments for validation and enhancement of fluid/structure dynamics codes to initial phases of LOCA.		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Projekt Nukleare Sicherheit, Kernforschungszentrum Karlsruhe IRE
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1978	1982	Dr. Krieg
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	December 1978	

1. General Aim

Development and verification of computer codes for analysis of the dynamic response of reactor vessel internals.

2. Particular Objectives

Design, performance and interpretation of laboratory experiments for validation of models and computer codes developed under PNS 4125.

3. Research Program

3.1 Nozzle experiments for investigation of two-phase flow models

3.2 Laboratory experiments of blowdown type supplementary to the HDR-blowdown tests

3.3 Experiments of oscillation type for investigation of fluid-structural coupling

4. Experimental Facilities, Computer Codes

Ad 3.1 IRB-water-steam-circuit
Nozzle test section
Computer code DUESE
Computer codes from PNS 4125

Ad 3.2 Container with setup for fast pressure release
Computercodes from PNS 4125

Ad 3.3 Thin-walled circular cylinder, empty as well as filled with water
Computer codes from PNS 4125 (SING-S, FLUX)

5.+6. Progress to Date and Results

Ad 3.1 The first series of 100 two-phase flow tests with the converging nozzle have been performed. Analysis of the 34 signals per tests have begun. First results have been obtained. They show good signal quality and indicate fairly good predictability by the drift flux model.

Ad 3.2 Investigations on the constructive details of the pressure vessel internals have been intensified. It turned out, that simulation of these internals during experiments of blowdown type depends strongly on the theoretical models which will be used.

Ad 3.3 For experiments with periodic excitation the required equipment is available or has been ordered. Calculations parallel to these experiments (fluid-structural-coupling) are in preparation.

7. Next Steps

Ad 3.1 Interpretation of the first series of two-phase flow tests in collaboration with LIT will be finished. A second series of tests is planned with some modifications of the test setup an(the trancducers.

Ad 3.2 Before carrying through the experiments of blowdown type, mor(knowledge is required about the theoretical models which are going to be verified.

Ad 3.3 The experiments with periodic excitation will be carried through for type a and prepared for additional types. Also th(calculations will go on in the same time scale.

8. Relations to other Projects

The project is closely related to PNS 4115 and PNS 4125. It is coordinated with all other projects of the HDR blowdown program.

9. References

[1] F. Kedziur
Investigation of Strongly Accelerated Two-Phase Flow;
ICHMT-Conf. on Momentum, Heat & Mass Transfer in Two-Phase
Energy and Chemical Systems, Dubrovnik, Jugoslavia,
2.-9. Sept. 78

[2] F. Kedziur, H. Mösinger
Vergleich zwischen ein- und zweidimensionaler Berechnung
einer Wasser-Dampf-Düsenströmung; KFK 2623, Okt. 1978

		CLASSIFICATION: 1.1.1
TITLE (ORIGINAL LANGUAGE): Critical and unbounded flow characteristics		COUNTRY: ITALY
		SPONSOR: C.N.E.N.
TITLE (ENGLISH LANGUAGE):		ORGANISATION: C.N.E.N.
		PROJECT LEADER: G. FARELLO
INITIATED: March 1978	COMPLETED:	SCIENTISTS:
STATUS: In progress	LAST UPDATING: June 1979	

1) General aim — Measure of blowdown parameters and jet flow characteristics.

2) Particular objectives — Measure of jet pressure distribution and two-phase flow parameters in the discharge channels (pressure, temperature, flowrate).

3) Experimental facility — Small laboratory loops with visualized test sections.

4) Project status — Data and parameters in the first test section have already been analyzed.

5) Next steps — A second test section and loop are in progress.

6) Reports — Centi, Cumo, Farello, Lumini, "Preliminary remarks on jet impingement and jet flow configurations", Cagliari 10-14 Sept. 1979, XVIII International IAHR Congress.

7) Contact person — G. Farello, CNEN, CSN, Casaccia, C.P. 2400, I-00100 Roma

		CLASSIFICATION: 1.1.1, 1.1.2
TITLE (ORIGINAL LANGUAGE): Programma P.I.Pe.R. - Esperienze di blow-down da un recipiente in pressione dotato di strutture interne.		COUNTRY: ITALY
		SPONSOR: CNEN and CNR
TITLE (ENGLISH LANGUAGE): Blow-down experiments by PIPER apparatus with internal structures.		ORGANISATION: University of Pisa
		PROJECT LEADER: Piero VIGNI
INITIATED: (second phase) 1975	COMPLETED: (second phase) 1980	SCIENTISTS: Umberto ROSA, Francesco D'AURIA
STATUS: in progress	LAST UPDATING: May 1979	

1. <u>General aim</u>
   To investigate some basic blow down problems for analyzing the causes of possible disagreements between the experimental results and the RELAP calculations, with particular reference to the scaling effects.

2. <u>Particular objectives</u>
   To reproduce experimentally, by a small-scale model, the pressure and mass transients following a L.O.C.A. in a B.W.R. and to evaluate the mechanical effects of the thermohydraulic transients on the internals and on supporting structures of the test vessel.

3. <u>Experimental facilities</u>
   P.I.P.E.R. apparatus is a pressure vessel, 3 m height, equipped with an electrical heating device, rupture disk assembly and instrumentation for pressure, temperature and liquid level measurements. The dynamic loads on the vessel supporting structures on the internals and on external targets can be also measured.
   The main design features of the vessel are:
   - pressure: 100 kg_p/cm^2
   - temperature: 310 °C
   - volume: 90 l
   - blow-down nozzles: two equal, at different height, with a diameter of 50 mm and a length of 400 mm.

4. <u>Project status</u>
   The program of blow-down experiments with internal structures was completed. Some tests at initial pressure of 70 ate, with low vapor mass fraction and a break position under the initial water level were performed; some geometrical parameters were varied (particularly the number and hydraulic diameter of jet pumps) for evaluating their effects on the pressure history.
   The dynamic loads on grid models and on supporting vessel structures were also measured.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Blow-down experiments by PIPER apparatus with internal structures.	1.1.1, 1.1.2

5. Next steps
Blow-down experiments are in progress to investigate the pressure and temperature histories at different stations in outflow nozzle, for full--opening breaks of different areas, up or under the initial water-level.

6. Relation to other projects and codes
SOPRE 1 research on the behaviour of pressure suppression containment system after LOCA - Project Leader: M. Mazzini.
Analysis of thermal and hydraulic transients following a LOCA - Project Leader: N. Cerullo.

7. Reference documents

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

2- N. CERULLO et al.
"Blow-down activity performed at Scalbatraio Center of 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, 1974).

3- P. VIGNI et al.
"Theoretical and experimental investigation on the model-laws for the vessel depressurization of a nuclear BWR without internals"
XVIII Congresso IAHR, Cagliari 1979.

4- P. VIGNI et al.
"Experimental and theoretical investigation on the depressurization of a vessel with internals"
ENS/ANS Topical Meeting on Nuclear Power Reactor Safety, Brussels 1978.

5- P. VIGNI
"Esperienze di efflusso rapido da un recipiente in pressione con strutture interne"
Istituto di Impianti Nucleari di Pisa, RL 317(78).

8. Degree of availability
The references 1-4 are free, the reference 5 may be available with the authorization of the CNEN.

(Istituto Impianti Nucleari, Università, V. Diotisalvi 2, I-56100 Pisa)

	CLASSIFICATION: 1.1.1,1.1.2,1.1.4,1.2,7.2
TITLE (ORIGINAL LANGUAGE): Analisi dei transitori termici ed idraulici a seguito di LOCA nei reattori ad acqua leggera e nei sistemi di contenimento associati.	COUNTRY: ITALY SPONSOR: CNEN and CNR
TITLE (ENGLISH LANGUAGE): Analysis of thermal and hydraulic transients following a LOCA in L.W. reactors and in associated containment systems.	ORGANISATION: University of Pisa PROJECT LEADER: N. CERULLO

INITIATED:	COMPLETED:	SCIENTISTS:
1974	1980	F. ORIOLO G. MEI G. CECCARELLI G. BITETTI
STATUS: in progress	LAST UPDATING: May 1979	

1. General aim
The program has the purpose of investigating thermal and hydraulic transients following LOCAs on light water reactors. To achieve a keener understanding of some aspects of the blow-down phenomena and to improve some features of both blow-down and containment codes checking their calculation with experimental data obtained at the safety facilities of the Scalbatraio Center, University of Pisa.

2. Particular objectives
Qualification of a "Best-Estimate" computer code in the analysis and design of experimental tests; the experimental results will be applied to check and improve the blow-down computer code capabilities.

3. Facilities
IBM 370/160 computer belonging to the CNUCE-CNR, Pisa. The experimental small scale facilities PIPER and SOPRE of the Scalbatraio Center, University of Pisa.

4. Project status
Extensive work has been carried on and continues regarding the WREM codes and the blow-down, heat-up and containment codes.
These codes have been used to analyze:
- LOCA Standard Problems proposed by NEA/CSNI;
- OECD-CNSI Containment Standard Problem 1: some results obtained will be presented at the OECD-CSNI workshop;
- results of experimental programs performed by the "Istituto di Impianti Nucleari" at the Scalbatraio Center, University of Pisa.

5. Next steps
- Application of the RELAP4-Mod. 5 and Mod. 6 computer programs to the experimental test on blow-down transients to implement the numerical models.
- Use of these codes in the analysis of the power plant LOCA transients.
- Use of the CONTEMPT-LT 026 computer program and its application to MARK III containment analysis.

45.897

TITLE (ENGLISH LANGUAGE): Analysis of thermal and Hydraulic transients following a LOCA in L.W. reactors and in associated containment systems.	CLASSIFICATION: 1.1.1,1.1.2,1.1.4,1.2,7.2

6. Relation to other projects
   - Blow-down tests by PIPER Apparatus - Project Leader P. VIGNI.
   - SOPRE 1 - Research on the behaviour of pressure suppression containment system after LOCA - Project Leader M. MAZZINI.

7. Reference documents

1- 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 program for the analysis of certain problems in thermal reactors safety - NEA C.P.L., Ispra 23-24-25 Ottobre 1974.

2- N. CERULLO et alii
"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 December 1976, on LOCA Standard Problems.

3- N. CERULLO et alii
"Experimental Investigation on the Behaviour of Pressure Suppression Containment Systems by the SOPRE 1 Facility"
Paper presented at the SMIRT 4 Conference, S. Francisco, California (USA), 15-19 August 1977.

4- N. CERULLO et alii
"Research on the behaviour of pressure suppression containment systems carried out at the University of Pisa: comparison between experimental results and CONTEMPT-LT calculations"
Istituto di Impianti Nucleari, Università di Pisa, RP 290(77).

5- G. BITETTI et alii
"OECD-CSNI Containment Standard Problem n° 1: Calculation Results by RELAP4-Mod. 5 and CONTEMPT LT-026 Computer Codes"
Paper presented at OECD-CSNI workshop, held in Frankfurt on September 10th, 1978.

6- P. VIGNI, F. ORIOLO, U. ROSA
"Experimental and Theoretical Investigation on the Depressurization of a Vessel with Internals"
ENS/ANS Topical Meeting on Nuclear Power Reactor Safety, Brussels (Belgium), October 16-19, 1978.

8. Degree of availability
   All references are available.

(Istituto Impianti Nucleari, Università, V.Diotisalvi 2, I-56100 Pisa)

Netherlands Energy Research Foundation (ECN)		CLASSIFICATION: 1.1.1./1.1.2.
TITLE: Mechanisch gedrag van het reactorbinnenwerk tijdens grote ongelukssituaties		COUNTRY: THE NETHERLANDS
		SPONSOR: ECN ORGANIZATION: ECN
TITLE (ENGLISH LANGUAGE): Mechanical behaviour of reactor internals during major accident situations.		PROJECTLEADER: L.H. Vons
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 postulated design accidents such as Loss Of Coolant Accidents.
- The deformation in the reactor internals immediately following a LOCA.

Experimental facilities and program: -

Project status

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

Next steps

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

Relations to other projects

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

Reference documents: not yet available.

Degree of availability: N/A.

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

Personnel: 1978 + 1979 : 0,6 manyear.

Classification 1.1.1/1.1.2	
Title 1 Blowdown Analysis	**Country** UK **Sponsor** CEGB **Organisation** Berkeley Nuclear Laboratories, Berkeley, Glos.
Title 2 Critical Two-Phase Flow	
Initiated 1975 **Completed** **Status** Continuing **Last Updating**	**Project Leaders** K.H. Ardron S.J. Board

1. General Aim

To improve capability of calculating critical steam water flow-rates in reactor size pipes.

2. Particular Objectives

(i) To develop a two-fluid model of non-equilibrium steam-water flow.

(ii) To test this model against measurements of flow-rate, pressure drop and density variations in pipes carrying critical steam-water flows.

3. Experimental Facilities

A pressure vessel with a capacity of 500 kg is being installed to discharge subcooled or saturated liquid at up to 35 bar through pipes of diameter 35-100 mm. Initially measurements will be made of mass velocity, subcooling and axial pressure and density gradients. Later experiments will provide data on bubble numbers in the discharge pipes.

4. References

Ardron, K.H. and Furness, R.A. - Studies of the Critical Flow Models Used in Reactor Blowdown Analysis, Nucl. Eng. Res., 39, 257.

Ardron, K.H., Duffey, R.B. and Hall, P.C., 1977, - Studies of Physical Models Used in Transient Two-Phase Flow Analysis. Presented at I. Mech. Engrs. Conf. on Heat and Fluid Flow in LWR Safety, Manchester.

Ardron, K.H., 1977 - A One-Dimensional Two Fluid Theory for the Critical Flow of Saturated or Subcooled Water in a Pipe. CEGB Report RD/B/N3967.

Ardron, K.H. and Ackerman, M.C. 1978 - Studies of the Critical Flow of Subcooled Water in a Pipe. Paper presented at Second CSNI Specialist Meeting on Transient Two-Phase Flow, Paris, June 1978.

CLASSIFICATION 1.1.2

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) STATUS Progressing	COMPLETED End 1976 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.

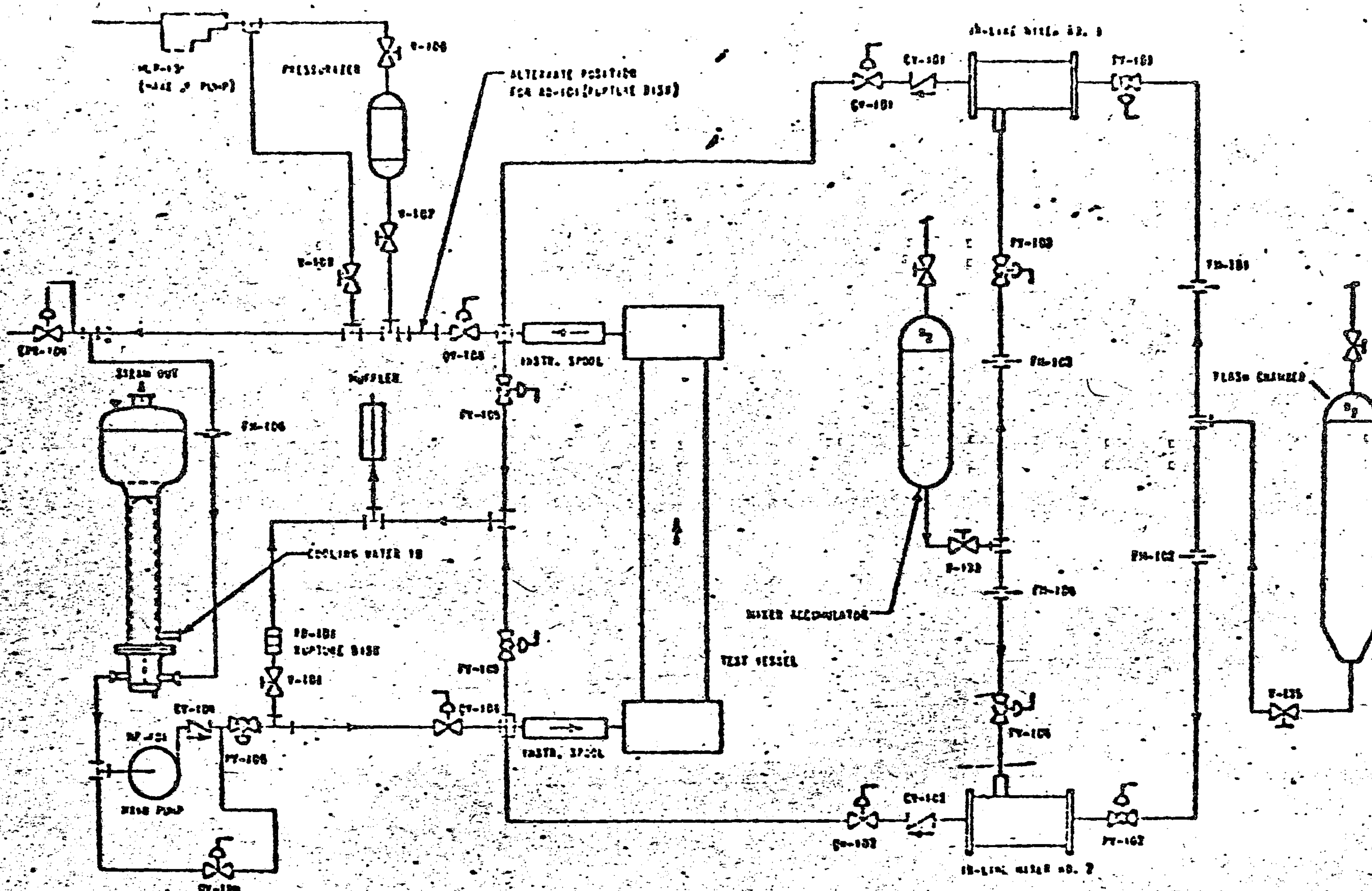


FIGURE 1 : BLOWDOWN HEAT TRANSFER TEST LOOP SCHEMATIC

	CLASSIFICATION: 1.1.2.
TITLE (ORIGINAL LANGUAGE): Critical Heat Flux in stagnant Water with a near-by cooled wall.	COUNTRY: BELGIUM
	SPONSOR: CEN/SCK
TITLE (ENGLISH LANGUAGE):	ORGANISATION: CEN/SCK
	PROJECT LEADER: Mr. HEBEL, W.

INITIATED: 1975	COMPLETED:	SCIENTISTS: Mr. DECRETON, M.
STATUS: Continuing	LAST UPDATING: May 1979	

1. General Aim

Investigating the mechanisms responsible for the Critical Heat Flux (burnout) limit in a particular geometrical configuration where a stagnant water layer is confined by opposite heated and cooled walls.

2. Particular Objectives

Providing design criteria for irradiation test devices containing an LWR fuel rod in a water filled pressure tube that is cooled from outside.

3. Experimental Facilities

A CHF test installation is available where electrically heated rods can be loaded into pressure tubes of different dimensions and of different materials. The rod diameters are usually those of PWR and BWR fuel pins and the spacing of the stagnant water gaps may vary between about 2 and 15 mm. The pressure on the water gap can be set from 1 to 240 bar. The test section is connected to a cooling water loop with variable and controlled entrance temperature. Various instrumentation equipment is provided for measuring wall temperatures and temperature gradients in the gap, for pressure and power control. Onset of CHF is detected by controlling the heating rod dilation by means of a linear motion transducer.

4. Project Status

Numerous testing runs have been performed for evaluating systematically the role of the different characteristic parameters as water pressure, rod diameter, gap spacing, cooling wall temperature, etc.. A theoretical approach has been developed for describing the observed phenomena and for predicting CHF occurrence.

5. Next Steps

Continuation of the experimental and theoretical work for a detailed quantification of the relevant heat transfer mechanisms as vapour production at the hot wall, heat and mass transfer through the gap and vapour condensation related to the cold wall.

6. Relation with other Projects

A complementary research program (called DNB) is being performed at CEN for extending the investigations towards the hydrodynamics of CHF in a forced flow of water.

7. Reference Documents

(1) W. HEBEL, P. DIRVEN : "Critical Heat Flux through a stagnant Water Annulus" KERNTECHNIK 17 (1975), 12, P.546-549

(2) W. HEBEL, M. DECRETON : "Mechanisms of Critical Heat Flux in a stagnant Water Annulus", 2nd MULTI-PHASE FLOW AND HEAT TRANSFER SYMPOSIUM, Miami Beach (USA), April 1979.

Critical Heat Flux in stagnant Water with a near-by cooled wall.

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

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

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

1. General Aim

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

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

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

2. Particular Objectives

2.1 Method of Radiotracer Velocity Measurement

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

2.2 Gamma Absorption Density Measurement Technique

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

3. Research Program

The experimental program consits of:

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

4. Experimental Facilities

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

5. Progress to Date

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

6. Results

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

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

7. Next Steps

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

8. Relation with Other Projects

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

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

RS 123 Safety Investigations performed at the decommissioned HDR plant

1.7. - 31.12.1977

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

9. Reference Documents

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

10. Degree of Availability of the Reports

Unrestricted distribution.

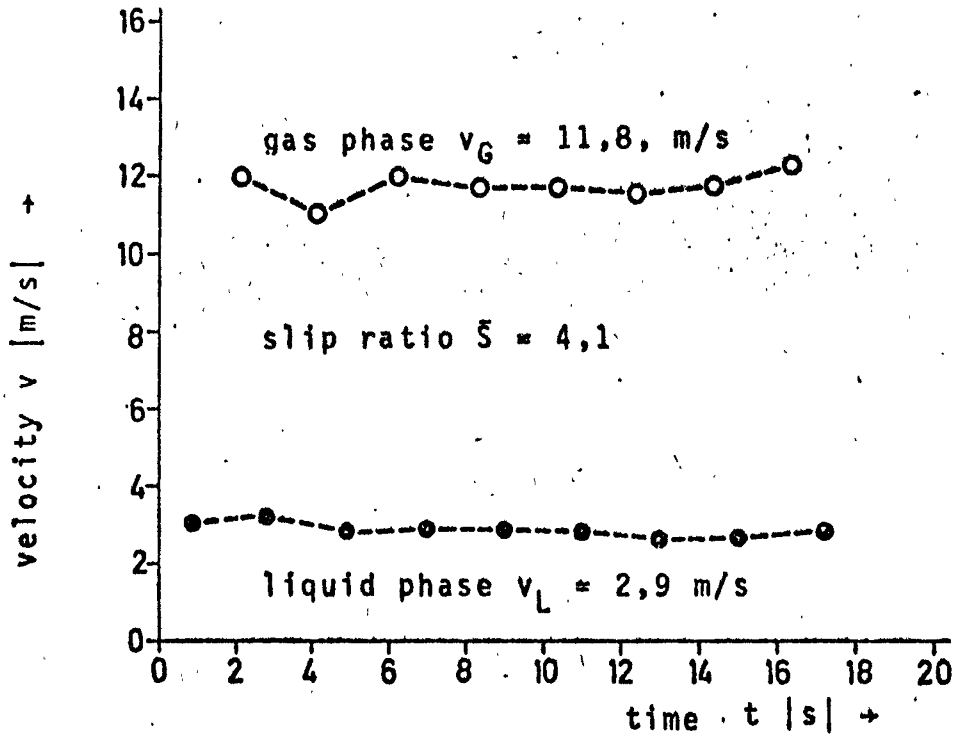

fig. 1: Velocity of gas and liquid phase (experiment 11)

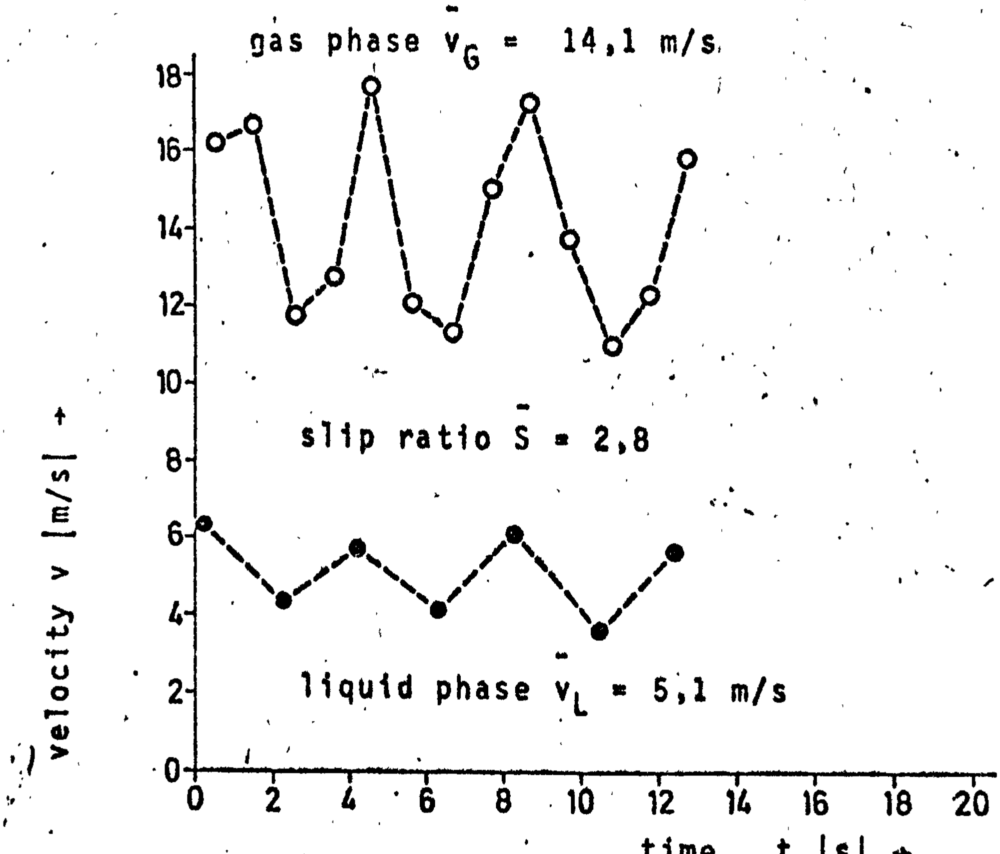

fig. 2: Velocity of gas and liquid phase (experiment 12)

Berichtszeitraum/Period 1.1.78 - 31.12.78	Klassifikation/Classification 1.1.2	Kennzeichen/Project Number RS 109
Vorhaben/Project Title Einfluß d. DWR-Umwälzschleifen a. d. Blowdown (LOBI Projekt) einschl. Erweiterung um LOBI-Pumpenversuche z. Ermittlung d. Zweiphasenströmungskennfelder (WCL-Hamilton, Aströ-Graz) Influence of PWR loops on the blowdown (LOBI project) incl. extension by LOBI pump tests to establish two phase flow characteristics (WCL-Hamilton, Aströ-Graz)		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kommission d. E.G. G.F.S.-Ispra Commission of the E.C. J.R.C. Ispra Establishment
Arbeitsbeginn/Initiated 1.12.73	Arbeitsende/Completed 30.11.81	Leiter des Vorhabens/Project Leader W. L. Riebold
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating Dec. 1978	Bewilligte Mittel/Funds

1. General Aim

Experimental investigation of the thermohydraulic behaviour of a primary cooling system (PCS) of pressurized water reactors (PWR) during a loss-of-coolant accident (LOCA) (blowdown) caused by a rupture of the main cooling pipe.

2. Particular Ojectives

Design and construction of a large scale 2-loop blowdown test facility.

Experimental investigation of the LOBI pump characteristics under two-phase flow conditions.

Performance of loss-of-coolant experiments (LOCE) by simulating pipe ruptures of different sizes at various locations within the PCS of PWRs.

Experimental investigation of the influence of the thermohydraulic behaviour of the individual components of the PCS of PWRs on the course of a LOCA (blowdown) by measuring the significant thermohydraulic quantities, in particular those influencing the core cooling.

Application of the experimental results to check and improve blowdown computer codes and associated theories used for the safety analysis of LWRs.

3. Research Programme

The LOBI R&D-contract between the BMFT-Bonn and the Commission of the European Communities (E.C.) foresees the execution of two experimental programmes, A and B, which have to be started after completion of the preliminary tests.

3.1 Execution of p r e l i m i n a r y t e s t s to determin the mechanical and thermohydraulic characteristics of the LOBI test facility, to check the measurement instrumentation, data acquisition and process control system, and to check the measured results on physical consistency and reproducibility.

3.2 E x p e r i m e n t a l P r o g r a m m e A consists of 6o tests in total aiming at the investigation of the influence of seven important parameters on the blowdown. These 6o tests are subdivided into two parts, A 1 and A 2, of 3o tests each.

3.2.1 E x p e r i m e n t a l P r o g r a m m e A 1 (tests 1 - 3o) is to be defined by the BMFT-Bonn exclusively, and the test results too, will be exclusively available to the BMFT-Bonn. The first test of programme A 1 ("virgin" blowdown test) has been chosen for a "blind" Standard Problem.

3.2.2 E x p e r i m e n t a l P r o g r a m m e A 2 (tests 31 - 6o) again is to be defined by the BMFT-Bonn taking into account suggestions from the Member Countries of the E.C.; the test results will be freely available to all Member Countries of the E.C.

3.3 E x p e r i m e n t a l P r o g r a m m e B (test 61 - 9o) is to be defined jointly by all Member Countries of the E.C. and comprises for the time being 3o tests; the test results will be freely available to all Member Countries of the E.C.. These tests are aiming at the investigation of the influence of the geometrical shape and the elevation of individual components on the blowdown.

4. Experimental Facilities, Computer Codes

The objectives of the investigation require a test facility the thermohydraulic behaviour of which during blowdown is as close as possible a reactor-typical one.

The LOBI test facility is an approximately 1/7oo scale model of a 4-loop 1300 MWe PWR primary cooling system (PCS). It consists of a 5 MW electrically heated test section (reactor model) with two primary cooling loops, one with three times the capacity (water volume and mass flow) of the other, representing the four primary loops of a PWR cooling system. Both loops are active loops each containing a pump and a steam generator, and designed for 160 bar and 325° C operation pressure and temperature respectively. Tube ruptures of various sizes (double ended to small leak) will be simulated at three different locations (hot leg, cold leg, loop seal) within the single or broken loop. Heat is removed from the primary loops by the active secondary cooling circuit (54 bar and 270° C operating pressure and temperature respectively) by means of two condensers (representing the turbines) and a cooler. From the emergency core cooling system (ECCS) for the time being only the intermediate pressure (up to 60 bar) or accumulator system is represented providing ECC water for both, separate and combined cold and hot leg injection into both loops; the high pressure injection system (HPIS) is planned to be added in the near future. Fig. 1 shows a schematic representation of the test facility.

The scaling factor of 712 has been applied to

- power input: 5 MW to 64 heater rod bundle
- coolant mass flow: 21 kg/s and 7 kg/s for the intact and broken loop respectively
- coolant volume: 0,83 m^3 within the primary loop system.

For the test facility design, the ratio of (1) power to volume, (2) pressure drop and temperature destribution along the flow paths and of (3) the components volume to each other has been preserved.

The height, and relative heights of components are scaled 1 : 1, thus preserving gravitational heads. The heat transfer surfaces (core rod bundle, steam generators) are full lengths.

An instrumentation system has been installed which provides for the measurement of all relevant thermohydraulic quantities under LOCA typical transient conditions at the boundaries (inlet and outlet) of each individual loop component (Fig. 2) and within the reactor model (Fig. 3 and 4).

The signals from about 4oo channels are recorded by a specially tailored data acquisition system using PCM techniques and analog magnetic tape, Fig. 5.

A process control system allows simulation of
- pump hydraulic behaviour by speed control
- fuel decay and stored heat by power control to the heater rod bundle during blowdown, Fig. 5.

Test facility design calculations have been and will be performed by the LOBI project staff with the RELAP4-MOD2 and -MOD5 and with the BERSAFE and STRUDL-II codes, and by the GRS-Munich (FRG) with the BRUCH-D, DAPSY and ZOCO-VI codes.

Pre-prediction and results evaluation calculations will be performed by the LOBI project staff with RELAP4-MOD5, -MOD6 and -MOD7 codes and by the GRS-Munich (FRG) with the same codes and with the DRUFAN and TRAC codes.

Special R&D contracts have been concluded by the Commission of the E.C. covering
- the "LOBI pump characteristics investigation in two-phase flow conditions"
- the calibration of the LOBI discharge nozzles under two-phase flow conditions; those convergent-divergent nozzles with various throat diameters will be inserted into the outlet branches of the LOBI rupture device for simulating various rupture sizes; they will be used for break mass flow measurements.

5. Progress to Date

During the report period the operation tests of individual elements and components have been completed, commissioning tests of the loop system have been executed and the preliminary or shake down (SD) tests have been started in the framework of which the first blowdown test (SD-o1) has been performed. The works executed in detail have been:

- Theoretical work has been concerned with
  - the development of several small computer programmes for test data handling and results evaluation and representation purposes
  - the preparation of an input data set and the performance of blowdown computer code calculations for the LOBI test facility
  - the analysis of thermohydraulic and mechanic test facility characteristics
  - design calculations for modifications and extensions of the test facility
  - the preparation of test programmes, results evaluation and documentation procedures
- The tests for the LOBI pump investigations in two-phase flow conditions have been executed in the laboratories of Westinghouse Canada Ltd. (WCL), Hamilton/Canada, in the framework of an R&D-sub-contract from the Commission of the E.C.; the tests are nearly completed, the results evaluation is under way
- For the LOBI discharge nozzles calibration tests to be performed in the laboratories of WCL-Hamilton/Canada, the respective experimental and work programme has been prepared and an R&D-contract from the Commission of the E.C. has been concluded.
- Completion of electrical wiring and connecting work.
- Completion of operation tests with the various elements and components of the mechanical and electrical system of the test facility.
- Performance of commissioning tests with the test facility
- Application of thermal insulation to the test facility
- Performance of the first blowdown experiment (SD-01) in the framework of combined commissioning and preliminary or shake down (SD) tests
- Completion of the preparation and of various connections for the final installation of the measurement instrumentation system

- Reduced instrumentation of the test facility for the commissioning and preliminary tests
- Measurement of several test facility characteristics
- Specification of an improved and extended measurement instrumentation system
- Mounting and commissioning of several calibration and test facilities for the various measuring transducers; performance of transducer calibrations
- Development, testing and extension of mini-computer programmes for measured data acquisition, handling and representation and for process control purposes
- Hardware extension of the LOBI data acquisition and process control system (DAPCS) for adaptation on supplementary requirements.

6. Results

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

6.1 Programm and Analysis

- A computer program package developed for generating a LOBI data base allows the storage of experimental and calculated results.

  The retrieval program developed allows the performance of basic mathematical operations (e.g. calculations of computed parameters etc.) with the data stored in the data base, the two- and three-dimensional representation and the listing and punching of those data. See e.g. Fig. 6.
- A "pump evaluation program package" has been developed for the results evaluation of the LOBI pump tests at WCL. The package consists of three parts: the "data processing procedure" for converting measured data into thermohydraulic and pump specific quantities, the "interpolation/extrapolation procedure" for determining matrix points, and the

"evaluation procedure" for final determination of the pump two-phase flow characteristics.

- A basis input data set has been prepared [3] for the survey calculations for the LOBI facility in the framework of the "ad-hoc Specialists WG on LOBI Programme B" activities. Survey calculations with REALP4/MOD5 have been performed by the LOBI staff for the tests nr. 4, 5 and 6 of the preliminary test matrix for programme B proposed by french experts. The results [4] have been discussed together with those from survey calculations of experts from various Member Countries. Physically contradictory calculation results have been obtained with respect to the core mass flow behaviour in case of a hot leg rupture which require still an explanation before being attributed to specific test facility features. A further interesting result was that in case of a rupture on the pump suction side (loop seal) the core temperatures were considerably higher for the case of the broken loop pump blocked than for continued operation of this pump at constant speed.

  Special RELAP4 code versions for the calculation of small leak accidents have been developed and presented by both the french (Framatome) and the german (GRS-Köln) experts during the last meeting of the "ad-hoc Specialists Working Group on LOBI Programme B".

- A special study was concerned with the initialization problem for the steam generator in RELAP4/MOD5 [5] and has shown, that

  -- the input data for the primary side temperatures are almost independent from the secondary side temperature and from the number of control volumes on the primary side

  -- the heat transfer coefficients on the secondary side are not constant but depend on the choice of the secondary side temperature.

- Theoretical considerations have been dealing with special characteristics of the test facility and with the experimental programme:

-- the difference between the temperature measured within the heater rods wall and the outer surface in steady-state conditions ranges between 10 and 14^{o} C; according to results from Semiscale heat transfer tests these values may be exceeded by up to 150 % during the initial blowdown phase.

-- the heat losses of the LOBI primary loop system have been estimated to about 300 kW and 20 kW for the non-insulated and insulated loop system respectively.

-- due to the emphasis shift in the LOBI project objectives from large to small rupture size experiments new operation requirements have to be met by the LOBI secondary cooling circuit for which the dynamic behaviour of this circuit is important and has therefore to be determined. For this purpose a computer program DYNES [6] has been developed modeling the circuit by 5 volumes, and the axial temperature profile of each volume. An experimental programme has been elaborated to establish the thermohydraulic characteristics of the secondary cooling circuit. [7]

-- as a further consequence of the increased interest in small leak tests, the emergency core cooling system (ECCS) simulation in the LOBI facility has to be extended by adding the high pressure injection system (HPIS); the required design calculations and components specifications work is at present under way.

-- BRUCH-D and ZOCO-VI code calculations (performed by the GRS-Munich (FRG) have shown, that the LOBI containment pressure history during blowdown can be adequately simulated by an appropriate control of vent valves to be installed at the outlet of the LOBI containment discharge pipes; a detailed discussion of the results obtained for double-ended 200 % and for single-ended 10 % breaks has revealed the need for further code calculations for at least one intermediate break size (25 % or 50 %) to back up the design and control specifications of the required vent valves. The simulation of the containment pressure during the LOBI blowdown experiments has been considered necessary for all break sizes from 200 % down to 10 %, although the break mass flow is critical two-phase flow during nearly the whole experiment duration; only for break sizes less than 10 % the containment pressure simulation has not been considered necessary because of critical break mass flow occurring during the main part of the refill phase.

— the results from DAPSY code calculations by the GRS-Munich/FRG yielding the hydraulic forces occurring within the LOBI loop system during the initial blowdown phase, have been used as input data for STRUDL-II code calculations by the LOBI project staff to perform a three-dimensional analysis of the dynamic loads on the loop structures. The results obtained at 47 nodes of the broken loop have shown that at several locations within this loop considerable forces (up to 50000 N) and/or displacements have to be expected during the first 125 ms of blowdown from a double-ended 200 % cold leg break. Based on these results and on a few acceleration measurements performed at the pump and the steam generator of the broken loop during the first blowdown test SD-01, it has been decided to install additional "shock arrestors" on the pump and steam generator of both loops. [8]

N.B. It has to be observed that in the broken loop of the Semiscale MOD 3 test facility of the USNRC in the INEL (Idaho National Engineering Laboratory) at Idaho Falls only 5 small "shock arrestors" have been mounted being designed against earth quake vibrations.

— in the framework of comparison analysis of the two similar projects Semiscale and LOBI, an analysis of the Semiscale test S-02-4 has shown, that the results evaluation efforts have by far not yet reached the desirable extent; therefore many problems and contradictory experimental results are left open and unexplained as e.g. the measured flow behaviour within the intact loop hot leg: drag disk and turbine flow meter show opposite flow direction during the 11 - 16 s time interval. Further analysis and

and interpretation results are reported in [9].

A comparison analysis of the Semiscale MOD 3 facility design with that of the Semiscale MOD 1 and the LOBI facility has been performed and is reported in [10].

-- the revision of the test matrix of the experimental programme A, and its subdivision into part A 1 and A 2, has been dealt with in several meetings of the "Working Group LOBI-GRS"; several new and additional parameters and the emphasis shift towards small break tests have given rise to detailed discussions of basic simulation problems and test facility features in the light of previous and new project objectives. A new test matrix of 73 tests has been proposed as a basis for final discussions and considerations; due to the large number of test parameters, and in view of achieving a reasonable transparency of the experimental programme, it has revealed necessary to define a reference test as basis for the parameter variation and to arrange several test groups.

-- an experimental programme for the commissioning and the preliminary tests had been prepared including the dynamic calibration of the heater rod thermocouples, the measurement of the primary loops heat losses, the determination of the primary loops natural convection characteristics and of the secondary circuit characteristics, the measurement of the pressure drop distribution within the primary loops and of the pumps coast-down behaviour, the mass flow calibration of the steam generators, and several blowdown tests representing shake down tests for determining the test facility mechanical and thermohydraulic behaviour, for the measurement instrumentation system, for the data acquisition and process control system and for the tests evaluation and documentation procedures.

This programme had finally been modified by postponing some of these tasks and combining commissioning and preliminary tests; in view of the first LOBI Standard Problem more interest had to be paid to check the reproducibility and physical consistency of measured results.

- according to a proposal from the USNRC, the first official LOBI test of programme A 1, which may be considered the "virgin" LOBI blowdown test with respect to blowdown code pre-prediction calculations for LOBI experiments, has been chosen for a blind Standard Problem exercise open to one USNRC contractor, BMFT contractors and the Member Countries of European Community.
- proposals have been prepared for (1) the documentation procedures of the LOBI tests by "Quick Look Reports" (QLR) and "Experimental Data Reports" (EDR) and for (2) the evaluation procedures for (a) validating the individual tests and (b) the respective results representation in the before mentioned reports.
- according to a proposal put forward by the USNRC the opportunity should be considered to extend the objective of the LOBI project from blowdown tests to integral LOCA tests covering the blowdown-refill-reflood period, which would first of all require to add also the low pressure injection system (LPIS) to the LOBI ECCS simulation.

- For the experimental investigation of the LOBI pump under two-phase flow conditions the geometrical configuration and the measurement instrumentation system of the test facility in the laboratory of WCL-Hamilton/Canada have been adapted to the special requirements from the tests objective.

  The experimental programme has been subdivided into two parts of 20 % and 80 % of the total number of tests. The test matrix of the second part has been defined on the basis of the immediately evaluated results from the tests of the first part which yielded the influence of the volumetric vapour quality α_V on the pump pressure head and torque, and the preliminary complete pump characteristics at 70 bar pressure and 40 % volumetric vapour quality.

  The tests of the second part of the programme have been completed, too; they yielded the data required to establish four further complete pump characteristics at 20, 70 and 75 bar pressure, and 95, 80 and 20 % volumetric vapour quality, respectively. A preliminary evaluation by means of the "pump evaluation program package" has shown the physical consistency of the measured results; the final evaluation is under way. An example is given in Fig. 7.

Pump tests under transient conditions have been cancelled since theoretical considerations have shown that the transient pump behaviour may be satisfactorily described by the aid of the steady state pump characteristics.

- Four different rupture sizes have finally been agreed upon for being considered in the LOBI experimental programmes A and B; they will be simulated by specially designed convergent-divergent discharge nozzles of four different throat diameters.

  These nozzles being intended at the same time to be used for break mass flow measurements have to be calibrated under two-phase flow conditions.

  A special computer program for calculating the critical nozzle flow has been extended to cover subcritical flow conditions and to include possible shocks within the divergent nozzle part.

  A computer program for results evaluation from the LOBI discharge nozzle calibration calculates the flow through the nozzle for critical and subcritical flow conditinns from the measured data, assuming homogeneous equilibrium two-phase flow.
  A special study contract to perform these LOBI discharge nozzle tests has been concluded with WCL-Hamilton/Canada.

6.2 Mechanical Components & Systems

- After completion of the operation tests of valves and pumps of the secondary and tertiary system, the whole test facility has successfully passed the pressure tests in ambient temperature conditions.

- The hot commissioning tests started subsequently have been performed including the various components in a stepwise procedure.
  During the interrupt period caused by a failure and the required repair within the secondary cooling circuit, the reactor model has been dismounted for a first inspection
- Application of thermal insulation to the whole test facility
- After restart of the loop operation, the remaining commissioning tests have been combined with the preliminary or shake down tests to recover part of the time lost due to strong time delay in thermal insulation work.
- The first LOBI blowdown test SD-01 (shake down test no. 01) has been successfully performed on December 13, 1978 in the framework of the preliminary tests programme.
  The initial data have been: p_R = 50 bar, T_R = 260° C fluid pressure and temperature at the reactor model outlet
  P_o = 1,9 MW heating power input leading at a core temperature increase of T = 14 °C at the nominal core mass flow of $\dot{m}$ = 28 kg/s.
  No failure or disturbance within the mechanical and electrical system of the test facility has occurred!

6.3 Electrical Components & Systems

- Completion of all electrical wiring and connection work within loop regulation and control system and within the electrical power supply system
- Installation and testing of an electrical loop safeguard system to automatically protect the various components in case of failures.
- Commissioning of the loop regulation and control system and of the 5,5 MW rectifier system together with the mechanical loop components and systems.

6.4 Measurement Techniques and Signal Analysis

- Completion of installation and connecting works of the various differential pressure transmission lines, of the various transducer cooling lines and of all signal transmission lines.
- Testing and calibration of wall and fluid temperature transducers, testing of differential pressure transducers and turbine flow meters.
- The measuring range of the dragbody probes has been finally determined on the basis of blowdown computer code calculations.
- Interferences in some amplifier units of the temperature measuring chains could be attributed to an erroneous ground wiring and could be removed.
- Special filters have been installed to eliminate disturbances in the heater rod temperature signals caused by the 5o Hz carrier frequency of the heater rod bundle isolation control.
- Special measurements with the LOBI γ-densitometers on a Plexiglass mock-up simulating typical two-phase flow patterns have shown that using a calibrated densitometer, the error in determining the density with relatively simple evaluation procedures remains within acceptable limits.
- Test measurements with the first γ-densitometer delivered have revealed serious errors. In close cooperation with the manufacturer the design specifications for the remaining 6 γ-densitometers to be ordered have been modified. Repeated test measurements with the modified devices have shown quite satisfactory results with respect to longterm drift and random drift; the cooling water temperature has to be measured to account for the influence of temperature fluctuations.
- Comprehensive checks and discussions have resulted in the final decision to use the radio-tracer method for phase

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velocity and mass flow measurements in the LOBI experiments; the activation and transport of tracers are ensured; a price offer from the KfK-Karlsruhe is still pending and has therefore prevented from placing already the respective order.

- Several high precision calibration facilities with computer controlled operation and data output for calibrating temperature, absolute and differential pressure transducers have become available.

- An air-water loop for test and calibration measurements under two-phase flow conditions has been constructed.

- Pressure drop distribution measurements and steam generator calibration for steady-state single-phase mass flow measurement have been performed in the framework of preliminary tests with the LOBI facility.

  The results reported in [11] show

  - the measured pressure drops are smaller than the calculated ones
  - there is a Re-number dependence due to very smooth tubes which requires
  - the necessity to repeat those measurements at different temperatures.

- A special automatic DNB control device has become available allowing the on-line display of up to 8o heater rod temperatures.

- The commissioning of the measurement instrumentation installed in a reduced extent for the first blowdown test SD-o1 has shown

  - unobjectionable signal transmission, no impairing interferences between control and switching signals (heating power, pump speed) and measurement signals
  - the failure of some heat rod thermocouples, resistance thermometers, turbine flow meters (bearings blocked) and of one dragbody (strain gage failure)

  The results analysis is still under way.

6.5 Data Acquisition and Process Control System

- The mini-computer program package for both the data acquisition, handling, evaluation and display system and the process control system has been further extended to meet special requirements from the experiments allowing
  - the graphical display of all measuring channels and of the respective calibration curves, and the taking of hard-copies
  - the compensation of offset and amplifier drift in converting raw data into those in engineering units
  - the graphical display of selected measuring channels for a direct and continuous test control
  - multiple control by status signals of preparation and start of the blowdown test against unintentional release
  - the start and record of calibrating the amplifier chains
  - the optimization of the selection velocity for the evaluation of PCM data
  - the statistical evaluation of densitometer signals and calculation of "computed parameters"
  - the formating of measured data on digital tapes and the documentation of the data acquisition
- A program for interactive data evaluation allows the one- to three-dimensional graphical data representation, the overlay of several curves, digital filtering, the generation of histograms
- Hardware has been extended to improve the heating power control and the pump speed control as well as the process control
- Installation of a real time clock and a second magnetic tape unit
- Commissioning of the data acquisition and process control system during the first blowdown test SD-01.

7. Next Steps

- Specification and performance of the first LOBI Standard Problem

- Final definition of test matrix for programme A
- Performance of pre-test and some post-test prediction calculations
- Completion and evaluation of LOBI shake down tests
- Start and performance of experimental programme A 1
- A 1 tests validation and documentation
- Installation of "shock arrestors", containment pressure control system and HPIS
- Improvement of measurement instrumentation system
- Evaluation of LOBI pump test results
- Performance of LOBI discharge nozzle calibration tests and evaluation of results.

8. Relations with Other Projects

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

Two-Phase flow Measurement Techniques

RS 135 : Signal correlation techniques
RS 145 : Joint two-phase flow test loop
RS 146 : Radiotracer techniques
RS 147 : Dragbody

RS 188 : NMR techniques

RS 225 : Density measurement by ultrasonic probes

9. Reference Documents

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Technical Note Nr. I.05.01.78.47, Oct. 1978

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DYNES - A Computer Program to Model the Secondary Circuit of the LOBI Facility
Technical Note (in preparation)

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Technical Note (in print)

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Technical Note Nr. I.05.12.78.44, Sept. 1978

[10] Hung Nguyen:
System Design Description of Semiscale MOD3 System and Comparison with Semiscale MOD1 and LOBI Loop.
Technical Note (in print)

[11] E. Ohlmer, W. Schulze:
Druckverlustmessungen an der LOBI-Anlage
Technical Note (in print)

10. Degree of Availability

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

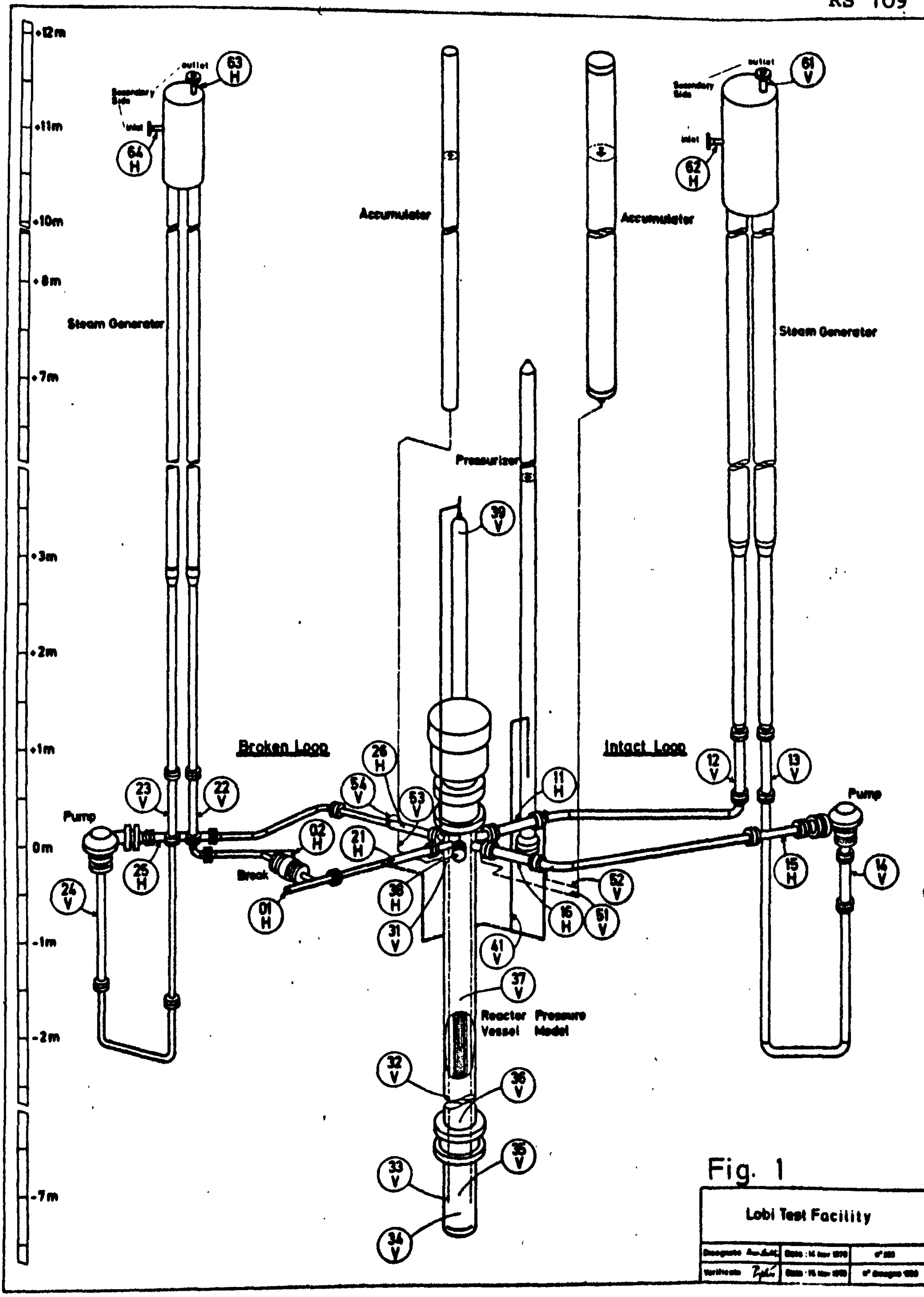


Fig. 1

Lobi Test Facility

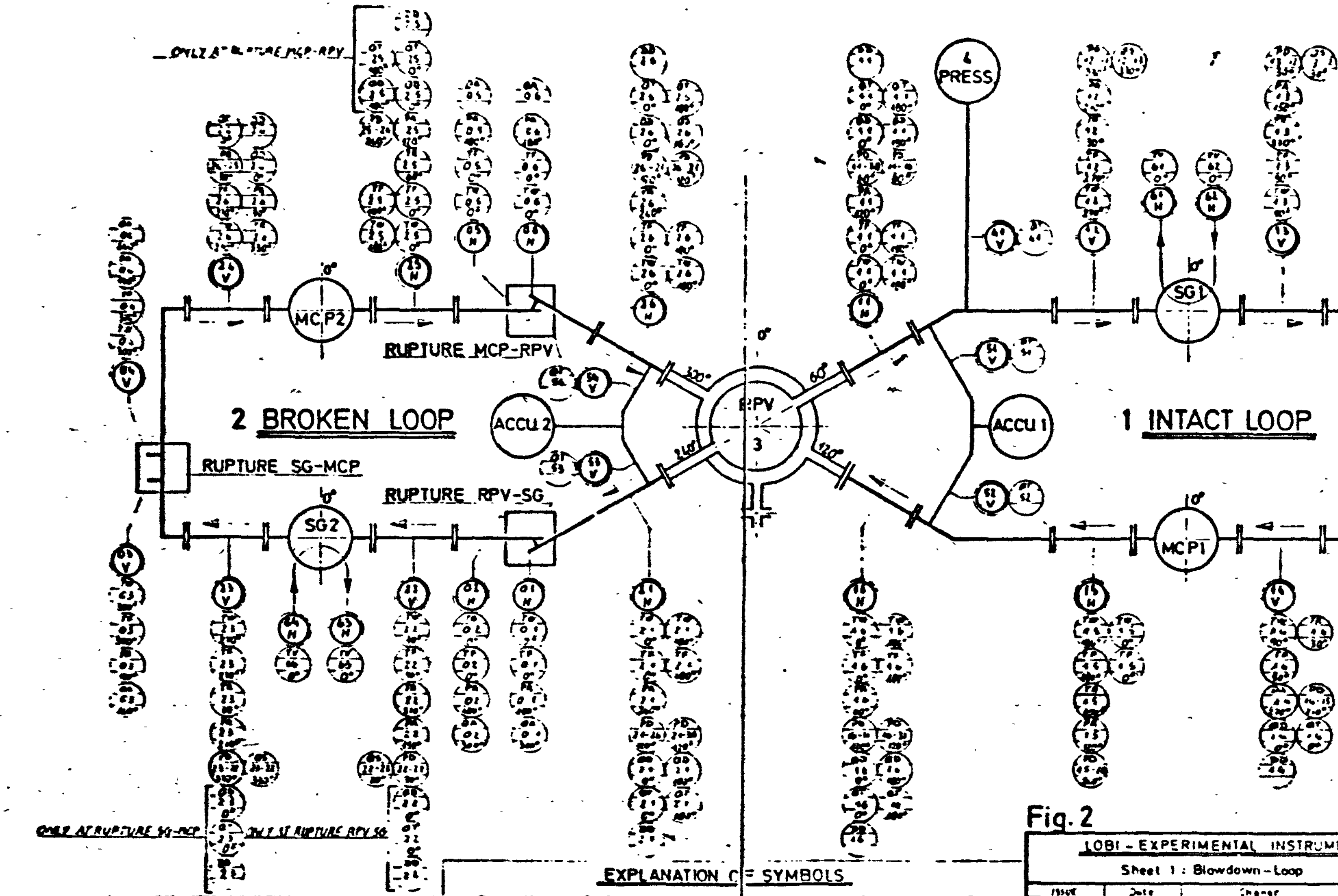


Fig. 2

EXPLANATION OF SYMBOLS

TW TEMPERATURE, WALL
TF " , FLUID
TR RESISTANCE THERMOMETER
TV TEMPERATURE VAPOUR
PA PRESSURE, ABSOLUTE
PD " , DIFFERENCE

( ) FLOW, DRAG BODY - IMPULSE
" , TURBINEMETER
" , APERTURE DIFFERENT PRESSURE
" , SG-DIFFERENT PRESSURE
(S or D) DENSITY (1 or 2 BEAM)

LOBI - EXPERIMENTAL INSTRUMENTATION

Sheet 1 : Blowdown - Loop

SERVIZIO DET : E4 72

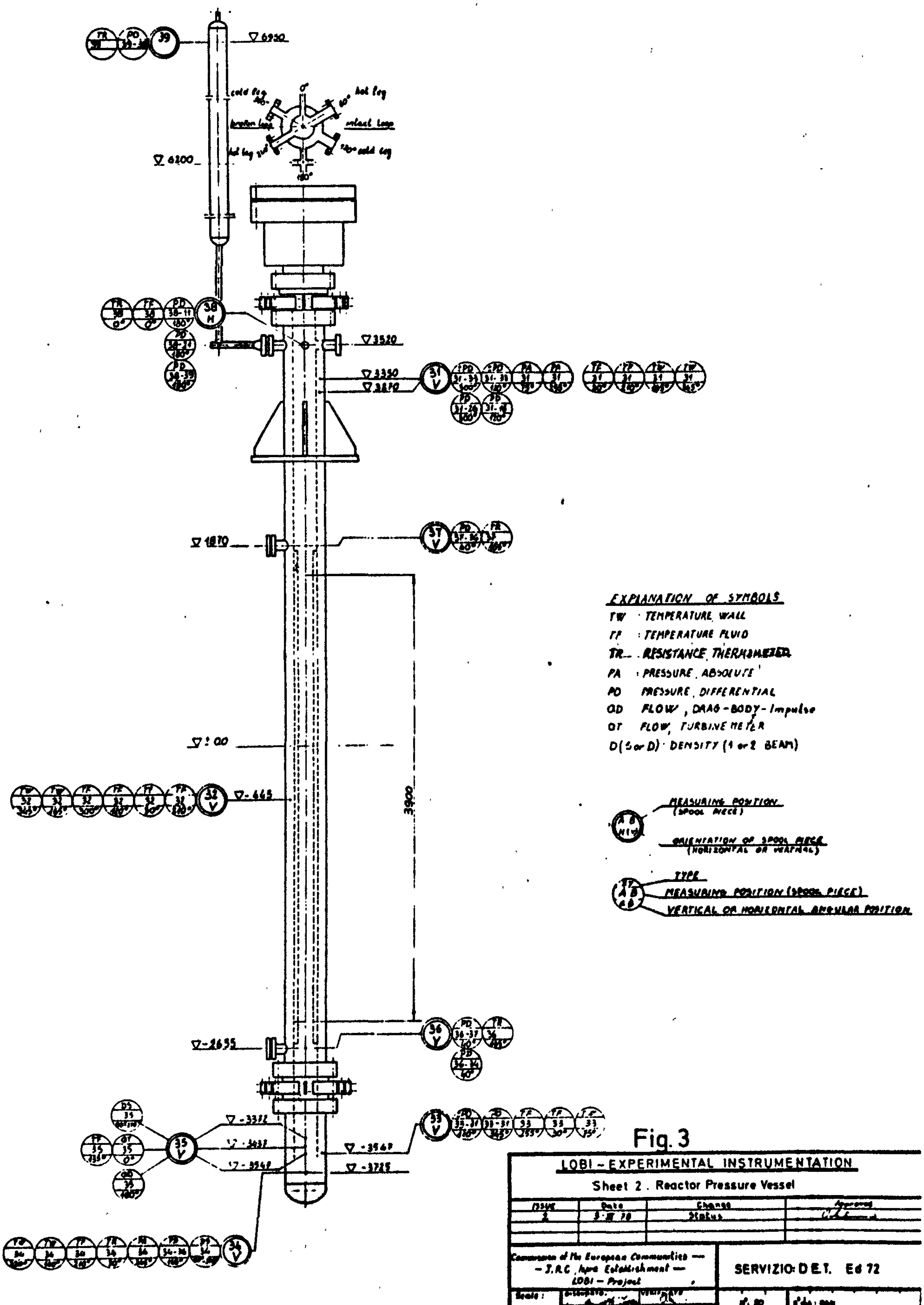

EXPLANATION OF SYMBOLS
TW : TEMPERATURE, WALL
TF : TEMPERATURE FLUID
TR : RESISTANCE THERMOMETER
PA : PRESSURE, ABSOLUTE
PD PRESSURE, DIFFERENTIAL
QD FLOW, DRAG-BODY-Impulse
QT FLOW, TURBINE METER
D(S or D) DENSITY (1 or 2 BEAM)
MEASURING POSITION (SPOOL PIECE)
ORIENTATION OF SPOOL PIECE (HORIZONTAL OR VERTICAL)
TYPE
MEASURING POSITION (SPOOL PIECE)
VERTICAL OR HORIZONTAL ANGULAR POSITION
cold leg
hot leg
intact loop
3900
LOBI - EXPERIMENTAL INSTRUMENTATION
Sheet 2. Reactor Pressure Vessel
SERVIZIO D.E.T. Ed 72

Fig. 3

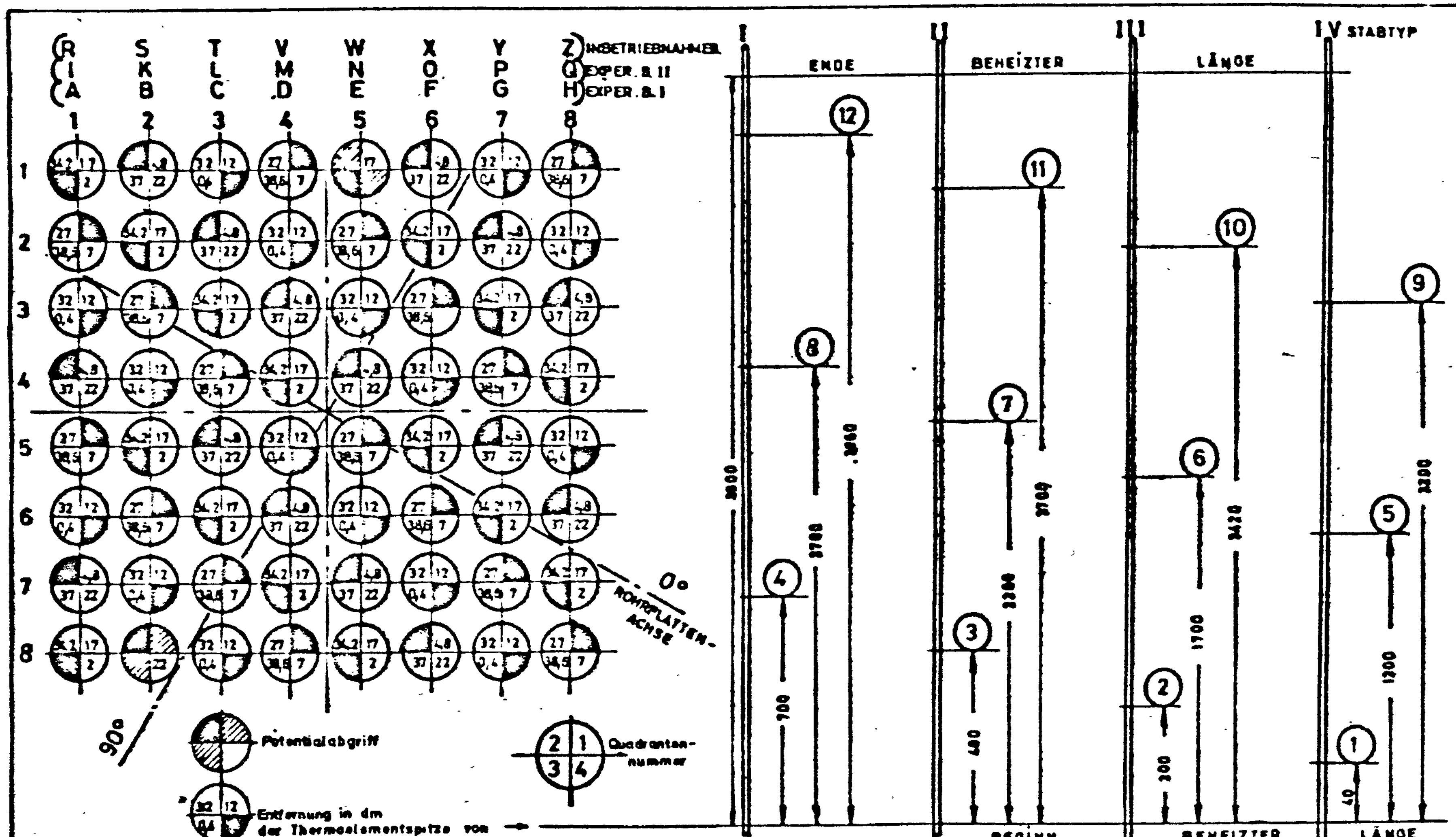


Fig. 4	Instrumentation of Experimental Heater Rod Bundle.

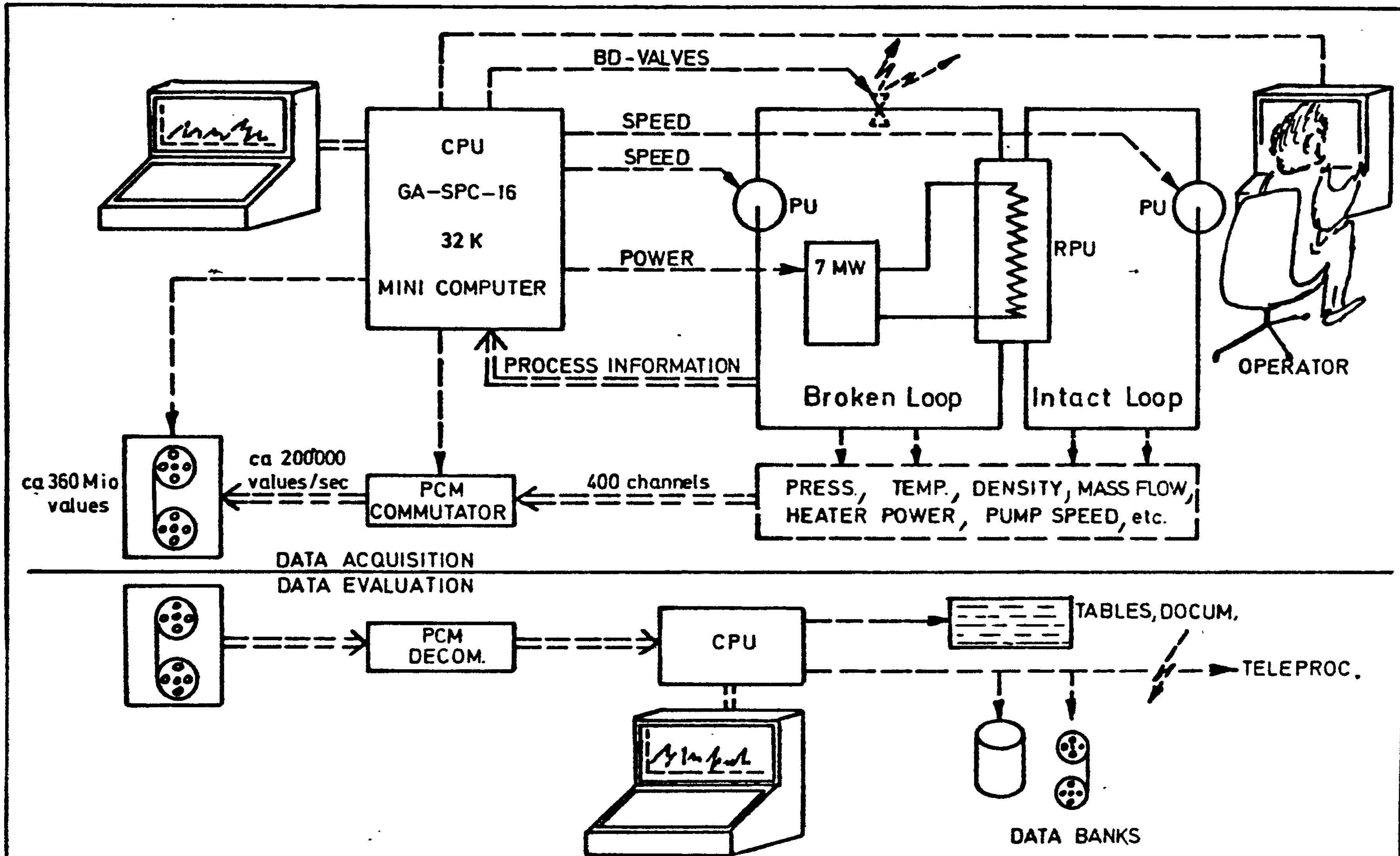


Fig. 5 : LOBI - DAPCS

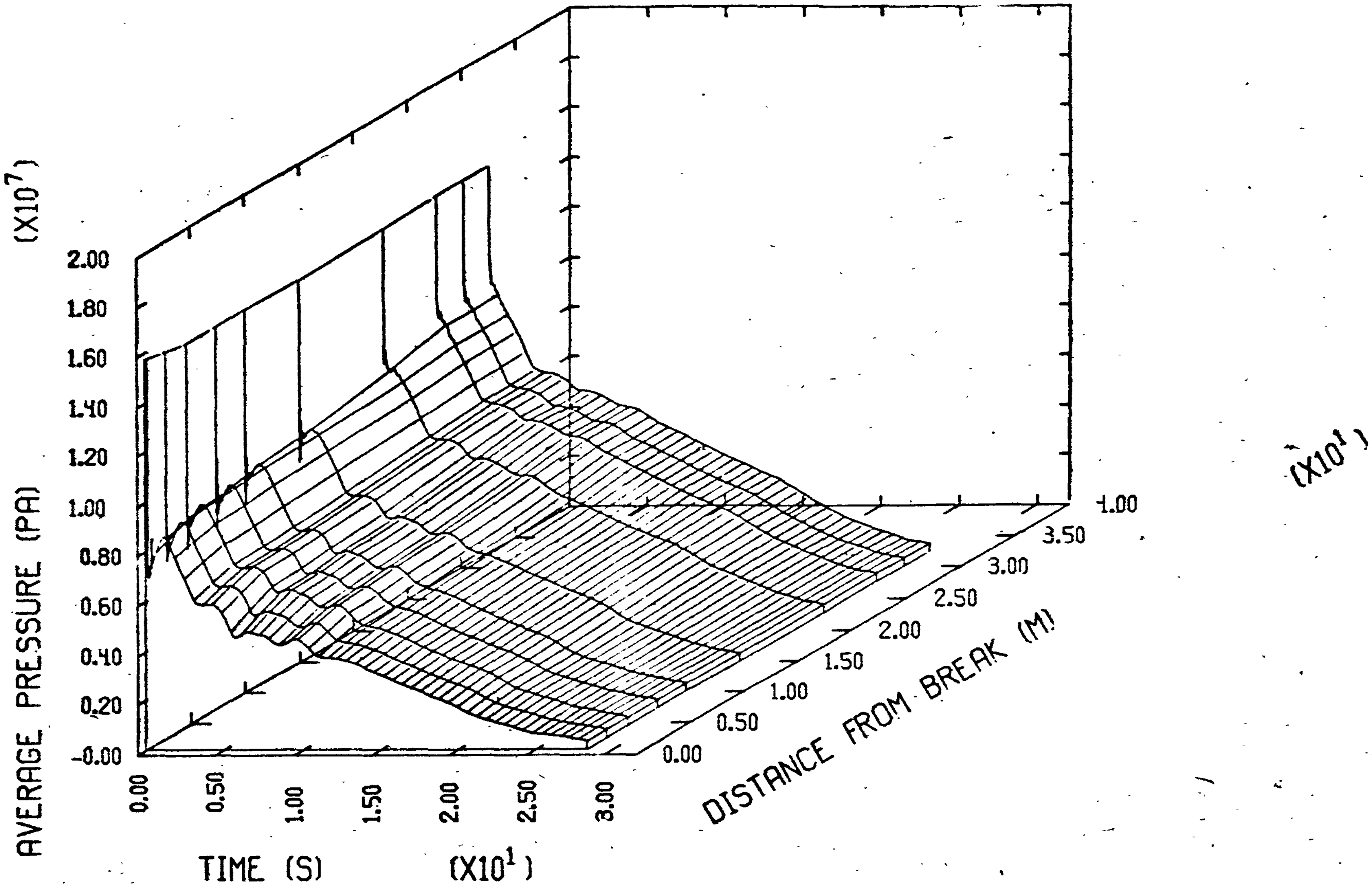
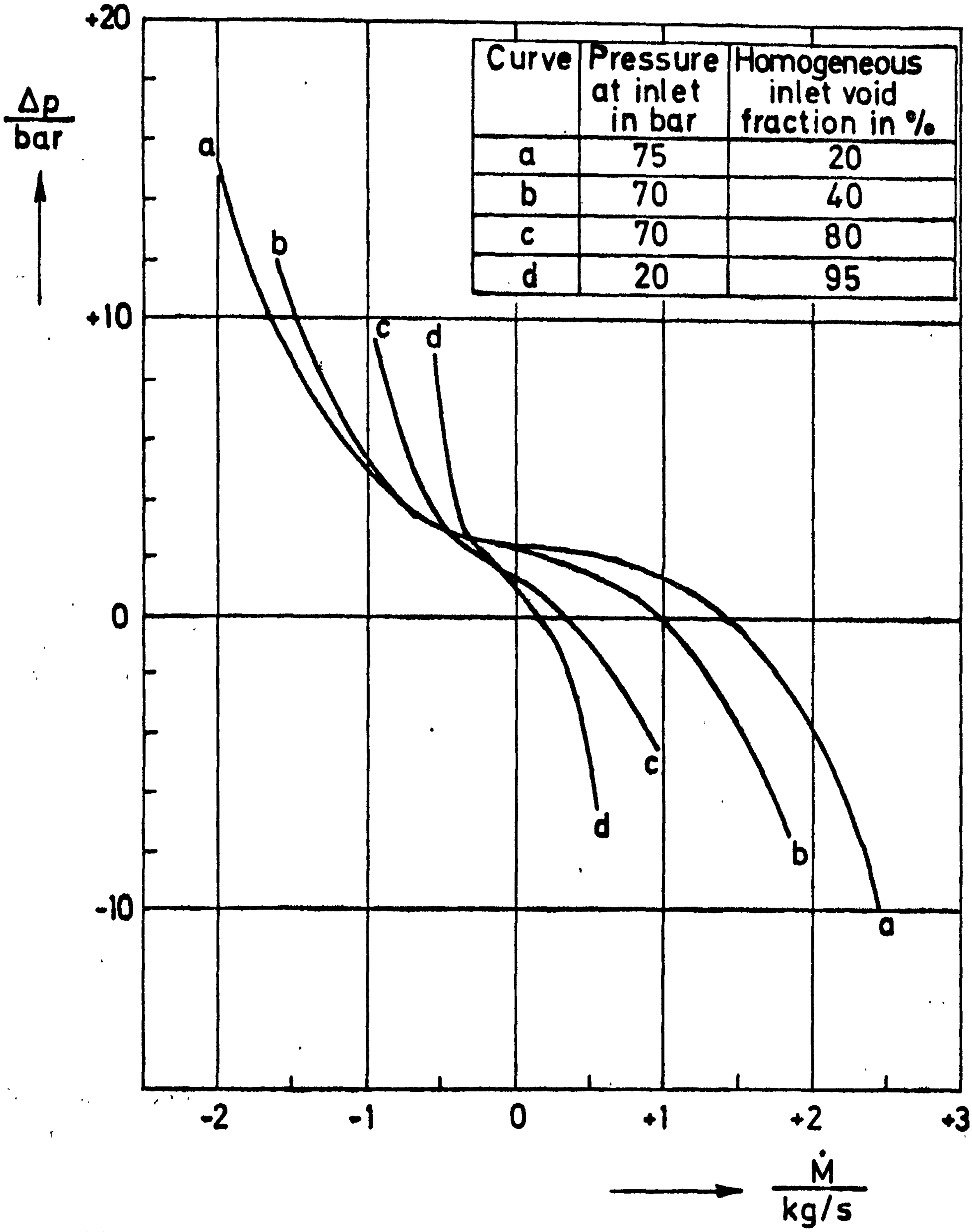


Fig. 5: Example of a three-dimensional plot with data from survey calculations [4]

COLD LEG BREAK   2A DOUBLE ENDED

PRESSURE HISTORY ALONG FLOW PATH IN BROKEN LOOP (BREAK TO VESSEL OUTLET)



Curve	Pressure at inlet in bar	Homogeneous inlet void fraction in %
a	75	20
b	70	40
c	70	80
d	20	95

Fig. 7 : Δp across the Lobi Pump at different inlet conditions for a constant speed of $n = +3250\ \text{min}^{-1}$

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	1.1.2	RS 111

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

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 9. 74	31. 12. 80	W. Kastner
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 78	

1. General Aim

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

2. Particular Objectives

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

3. Research Program and 4. Experimental Facilities

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

A test loop at C-E will be modified in order to measure two-phase pump characteristics for parameter variations of interest: pressure, flowrate and void. The test matrix for these two model pumps contains about 325 steady-state points in the two-phase region. Also 9 transient tests will be carried out, to investigate whether steady-state results are applicable to transient LOCA calculations.

5./6. Progress to Date/Results

C-E Pump Model

After the completion of the experiments, which were conducted in the frame of Phase II of the EPRI program, a supplementary calibration of the built-in instrumentation was undertaken. Subsequent to the examination of all measured data, the final evaluation and analysis of the results have started. It can already be established with high reliability that the quality of the measured data is very good. The final report is under preparation.

KWU model pump

The motor drive system of AEG was shipped to ASTRÖ in Graz/Austria. Together with the gearbox and the mechanical brake the motor was tested. The whole drive system has then been shipped to C-E in Windsor/USA.
The model pumps of ASTRÖ were manufactured. All components of the instrumentation were prepared. Cold water tests to get characteristic curves in three quadrants for the both model pumps were performed under single-phase conditions. The pumps were assembled to perform function tests. Due to the delay of delivering the model pumps, the hot water tests were shiffed and will be performed at C-E during the shakedown tests.
Planning, design and analysis work in anticipation of equipment installations was done at C-E. Also C-E has started procurement of the required instruments and other test loop equipment. C-E personnel witnessed operation of the test pump auxiliaries at Westinghouse Canada Limited in Hamilton/Ontario, where the LOBI-pump of EURATOM/Ispra using the same auxiliaries has been tested. C-E personnel also witnessed test pump motor tests at ASTRÖ and participated in training on the test pump motor and controller at AEG in Berlin.
Based on the experience of C-E with the C-E/EPRI pump performance program a test matrix was proposed by KWU and discussed with ASTRÖ and C-E. These and additional discussions with EURATOM/Ispra, Westinghouse Canada Limited and CREARE required some improvements.

7. Next Steps

For the motor drive system AEG has to provide some spare parts. ASTRÖ will perform a function test for both model pumps. The model pumps will be delivered in February 1979. The experiments with the LOBI-pump of EURATOM are finished at the end of January 1979. After disassembling the test equipment the auxiliaries will be shipped from Westinghouse Canada to C-E. Installation of all the components will start in February 1979. The pump tests will start in the middle of 1979. At KWU a reworking of the test matrix will be performed.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.78	1.1.2	RS 176

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

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.9.1975	31.12.1978	Fulfs
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

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 as regards DNB-measurements.

2. Particular objectives

The program is divided in the following four parts:

I. cold flow pressure drop measurements,

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

III. critical heat flux measurements.

IV. power and/or massflow-transients (as addition similar to research program RS 64 Part II).

In addition and supplement to the research program RS 164 these experiments could 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 conditions 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 enthalphy.
Critical heat flux tests will be carried out over a large range of inlet conditions e.g. massflow rate, inlet tempera-

ture, system pressure, mostly valid for PWRs. For some aspects CHF-points at a pressure of 70 bar will be investigated.

To get information about the DNB-time delay at transient contitions a new part - similar to Part II of research program RS 64 - was added. These tests enclose power and/or massflow-transients over a large range of inlet and starting conditions.

The influence of different grid configuration will be inquired with two test sections of different spacer type (test section 1 - with mixing vanes, test section 2 - without mixing vanes).

4. Experimental facilities

See RS 176 annual report A 76 and /1/.

5. Progress to Date

The modification of the electrical power supply was finished. After installation of test section 1 (spacers with mixing vanes) and calibration tests the cold flow pressure drop measurements and most of the steady state DNB-measurements (part I and III of the program) had been carried out. The mixing tests will follow after installation of a heating and isolation system of the shroud. The massflow-transients with constant power have just started.

Parallel to the tests a computer code had been developed to compare steady state DNB-data in water (research program RS 164) and freon (research program RS 176). At this moment the code works with the Stevens-method.

6. Results

The above mentioned tests look sufficiently.
The code is proved with few water/freon data of the literature.
(Publications during report period see 9.)

7. Next steps

The test program will be finished with test section 1. After installation of test section 2 and the necessary calibration tests the test program (part I to IV) will be repeated. Water and freon measurements will be compared by the method of Stevens and the similarity factors will be calculated.

8. Relation with other Projects

See RS 176 annual report A 76.

9. References

/1/ H. Fulfs, H.-O. Boie, K. Hofmann, A. Katsaounis, M. Kreubig, R. Orlowski, J. Ottersbach, C.-H. Peters, H. Rosomm, K. Stellwag:
Technischer Fachbericht Notkühlprogramm - Forschungsvorhaben RS 176, Beschreibung des Frigen-Versuchsstandes und des DWR-25-Stabbündels,
GKSS 78/I/11

/2/ A. Katsaounis, R. Orlowski, L. Ladeira, H. Fulfs, K. Hofmann:
Burnout experiments in fréon 12 using different types of orifices to simulate the core grids.
(Vortrag zu European Two Phase Flow Group Meeting, Stockholm 1978),
GKSS 78/E/17

10. Degree of Availability of the Reports

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

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	1.1.2	RS 64
Vorhaben/Project Title Untersuchung der stationären und instationären kritischen Heizflächenbelastung an Vielstabbündeln von Druck- und Siedewasserreaktoren mit Frigen als Modellflüssigkeit. Investigation of the Steady State and Transient Critical Heat Flux of Multi-Rod Bundles for PWR and BWR with Freon as Model Fluid.		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor GKSS - Ceesthacht Institut für Anlagentechnik
Arbeitsbeginn/Initiated 1.11.72	Arbeitsende/Completed (30.6.78)	Leiter des Vorhabens/Project Leader Fulfs
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General aim

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

2. Particular objectives

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

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

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

3. Research program

See RS 64 annual report A 77

4. Experimental Facilities, Computer Codes

See RS 64 annual report A 76, A 75

5. Progress to Date

After the failure in the power supply system (see RS 64 - A 77) the research program RS 64 was stopped at the beginning of 1978 in favour of research program RS 176.

6. Results

The results of the mixing experiments at the PWR-test section are reported /1/.

7. Next steps

The research program RS 64 is closed.

8. Relation to other projects

RS 176 and RS 164.

9. References

/1/ H. Fulfs. K. Hofmann. A. Katsaounis, M. Kreubig, C. von Minden, R. Orlowski:
Technischer Fachbericht Notkühlprogramm - Forschungsvorhaben RS 64 - Durchmischungsversuche am DWR-49-Stabbündel (Phase I)
GKSS 78/I/6

10. Degree of availability

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

Berichtszeitraum/Period 01.01. - 31.12.1978	Klassifikation/Classification 1.1.2	Kennzeichen/Project Number RS 163
Vorhaben/Project Title Theoretische und experimentelle Untersuchungen zum thermohydraulischen Verhalten des Reaktors Cores in der ersten Blowdown-Phase Theoretical and Experimental Investigations on Thermo- and Fluiddynamic Behaviour of the Reactor Core in the First blowdown Period		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Institut für Verfahrenstech. der Universität Hannover Callinstr. 36 3000 Hannover
Arbeitsbeginn/Initiated March 1975	Arbeitsende/Completed February 1979	Leiter des Vorhabens/Project Leader Prof. Dr.-Ing. F. Mayinger
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General aim

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

2. Particular objectives

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

3. Research program

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

3.1 Entraiment-investigations

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

3.2 Mixing-investigation

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

3.3 Loop behaviour

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

4. Experimental facilities, computer codes

to 3.1 In 1978 a theoretical model was developed and a systematically analytical interpretation of the entrainment-tests at an inside cooled tube for steady state and transients conditions was done. The agreement between theoretical prediction and experimental results shows good agreement. The maximum deviation is in the range of $\pm$ 20 %.
The measurements of the liquid fraction entrained in a 4-rod bundle at transient blowdown condition was performed. For these investigations the applicablitity of STORZ-lenses was tested.
A computer code was developed to calculate the entrainment mass flow rate in annular flow for transient and steady state conditions.

to 3.2 For the mixing messuremtents in heated fluid a two channel test-section consisting of two adjacent central-subchannel was used. The profile of the mixing test-section in shown in fig. 1. The walls are heated indirectly by 6 heater rods. Fig. 1 shows that the profile is spaced by 1 mm diameter stainless steal pins. By this assembly it was possible to simulate the boiling phenomena occuring in the gap-reon nuclear reactor.
In order to recalculate the experimental results the MIT-COBRA-IIIC code was enlarged by a new mixing modell.

to 3.3 The blowdown experiments were performed with a test rig (Fig. 5) containing a direct electrically heated four rod bundle with PWR-geometry, an intact loop representing three intact loop an standard-PWR and the break configuration representing the broken loop.

5. Progress to date

to 3.1 The comperison of the inside-cobled tube tests results with the theoretical preditiom shows satisfactory agreement. The optical measurements of the phase distribution in a 4-rod bundle show good quantitative results especially for the post-dryout phase, when a sprayflow occurs in the subchannel.

to 3.2 The mixing investigations led to better physical understanding of the two phase mechanism. By the evaluated new mixing model a better assesment of the results was achieved than by models taken from the literature.

to 3.3 The experimental work concernd with the influence of loop components on the tehrmohydraulic bahaviour of the coolant in the reactor core has been finished during this report period.
The results obtained from the R 12-tests show good quantitive agreement with Semiscale tests.

6. Results

to 3.1 Fig. 2 and fig. 3 show two typical examples for the low pattern in a 4-rod bundle occuring under transient blowdown conditions. Fig. 2 shows an annular flow - just before dryout - with a very thin liquid film flowing on the rod surface and only very little entrainment in in the center of the channel.
Fig. 3 is characteristic for the post-dryout period of the LOCA : a spray flow with filaments of liquid on the dry tube wall and some droplets carried away by the vapour phase. These pictures give a first qualitative idea about mechanism occuring under blowdown conditions. The rods are diretcly electrically heated.
The theoretical model evaluated to recalculate the transient entrainment data taken at the inside cooled tube, bases on the conservation law for mass subdivided in these for the liqiud film and the droplet loaden gas core -, the energy balance for the total flow rate and the momentum balanve valid for the droplet loaden gas core limitted by the liquid film surface.
In addition to this system of differential equations two constitutive equations are needed describing the interfacial, shwar stress and the total pressure drop of the two-phase flow occuring to Levy's theoretical model /2/.
This system of equations can be used to calculate the entrainment mass flow rate in a transient annular flow, if the time depending gradients of total mass flux, system pressure and vapour velocity are known from measurements or from a blowdown code like RELAP.
A comparison between caculated and measurd entrainment mass flow rates is shown in fig. 4. The maximum deviation between test-results and theoretical prediction is in the range of about 20 %.

to 3.2 The freon 12 test loop shown in fig. 5 was built for steady and transient mixing investigations. The loop consists essentially of the test-section with adjacent central channels and six separately heated rods, the lower and upper plenum in scale of a standard PWR, cold and hot leg break bursting discs and a new two-phase mass flow measuring technique, which was developed during the last report period.

The instruments - similar to these use in the LOFT-Tests - used for the mass flow measuring technique and their arrangement are shown in figures 6 to 8.

The undisturbed subchannel-flow enters coaxial impedance void meter. Only by the exact knowledge of the phase distribution, it is possible to analyse the two-phase momentum on the drag screen.

For the determintaion of the aberage velocity the two phase flow was homogenized by a mesh arrangement, which was placed in front of the turbine flow meter.

The preliminary experimental mixing investigations with air water two phase flows have shown that produced by velocity differences between the adjacent subchannels a plane shear flow exists at the smallest cross-section between the rod. By this shear flow periodical vortices are generated, which have an important influence on mass and energy exchange. With help of theoretivcal equations for viscous fluids the extension of the vortices and the velocity profile between the rods could be caculated. A mixing model for mass - and energy exchange based on viscous vortices shown in equation 6 was derivated and implied to the COBRA-III-C code. See fig. 9. Fig. 10 shows the calculated velocity distribution of the plane shear flow between two spces. The shear flow has been calcualted takin into account only the influence the channels does not only depend on the transverse velocity but also on the different densities in the channels. Recalculation of experimental results by aid of the new model showed better agreement than by the models existing in the literature.

to 3.3 The Blowdown-Testprogram was diveded in investigations of hot leg and cold breaks. Following races of test parameters were used :

<u>Hot leg break :</u> - heat flux density : $\dot{q}/\dot{q}_{DO;St.State} = 0,3 \ldots 0,7$
- break area : $A_q = 0,7\,F \ldots 2,0\,F$
- simulated loop behaviour :
steam generator ⇒ $\zeta_{SG} \cdot 0$; ζ_{CORE} ; ∞

cold leg break : - heat flux density : $\dot{q}/\dot{q}_{10;ss}$ = 0,3 ... 0,7
- break area A_q : 0,7 F ... 1,4 F
- simulated loop behaviour:

pump resistance
$$\left[\begin{array}{ll} \xi_p = 0 & ; \quad \xi_p = \xi_{CORE} \\ \xi_p = 1{,}5 \cdot \xi_{CORE} & ; \quad \xi_p = \infty \end{array}\right.$$

In Fig. 12, 13 the measured dryout delay times for the leg breaks and different loop resistances are plotted versus the heat flux ratio stat. One can see that the loop resistance has little effect on the dryout delay time for a heat-flux density $\dot{q}/\dot{q}_{ss} > 0.5$. With decreasing heat flux density $\dot{q}/\dot{q}_{ss}$ < 0,5 there is a strong dependence from break area and loop resistance. This effect is supported by the flashing in the "Cold part" of the intact loop. With a large resistance of the simulated steam generator fluid from the lower plenum and cold leg enters the core which results in a better cooling of the fuel rods. At higher heat flux densities the Dryout occurs before flashing is started. But there is a remarkable increasing of the part dryout heat transfer, when there is strong, because of the high mass flux acceleration of the vapour-liquid mixture in the rod bundle at high loop resistance.
From Fig. 14, 15 one can see that there is a strong dependence of the dryout delay time from loop resistance for breaks simulated in the cold leg. That is why a rapid flashing in the upper plenum and hot leg starts at the beginning of the blowdown. For high loop resistance most of the discharge two phase mixtuee will be transfered from the upper plenum into the rod bundle while at low loop resistance the dischargement through the intact loop to the break area becomes greater.
The smallest Dryout dealy time occurs in the middle of the rod bundle. This has two reasons :

1. The two-phase mixture which enters the upper core will be evaporated by reaching the lower part of the core.

2. The steam quality in the middle part of the core will be higher than in the lower rod bundle at beginning of the blowdown.

7. Next steps

to 3.1 The entrainment and phase distribution measurements in the four-rod bundle will be finished in the next two month.

to 3.2 Detailed mixing-test will be carried out with boiling mixtures at the two-channel test-section. The test-results obtained from these experiments will be recalculated with the new developed mixing-model.
The experimental work for the investigation of the loop-components influence was brought to an end. In 1979 a recalculation with one of the wellknwon computer-codes - for example Frelap - is planned to be done.

8. Relations with other projects

RS 37, 37-1, 37-2

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

9. Literature

/1/ BMFT-RS 163 - 03
Jahresbericht 1977 zum Forschungsvorhaben RS 163
Universität Hannover - Institut für Verfahrenstechnik

/2/ Levy, S.
Prediction of Two-Phase Flow with Liquid Entrainment
Int. Journal of Heat & Mass Transfer 9 (1966) 171

10. Degree of acailbility

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

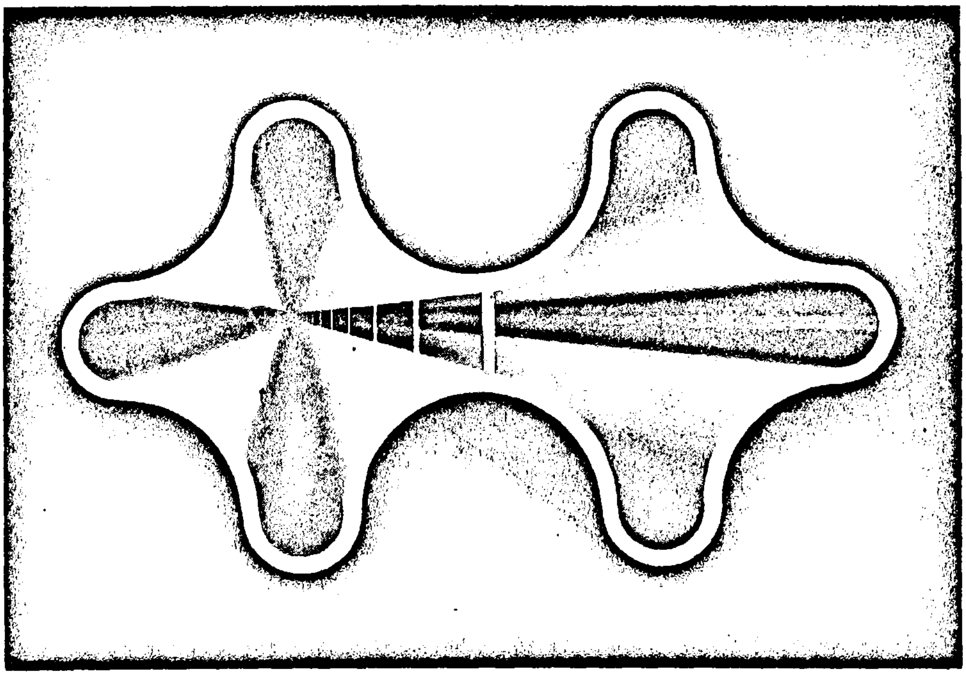


Fig. 1 : Profile of the two channel mixing test-section

Fig. 2 : Annular flow in a rod bundle just before dryout

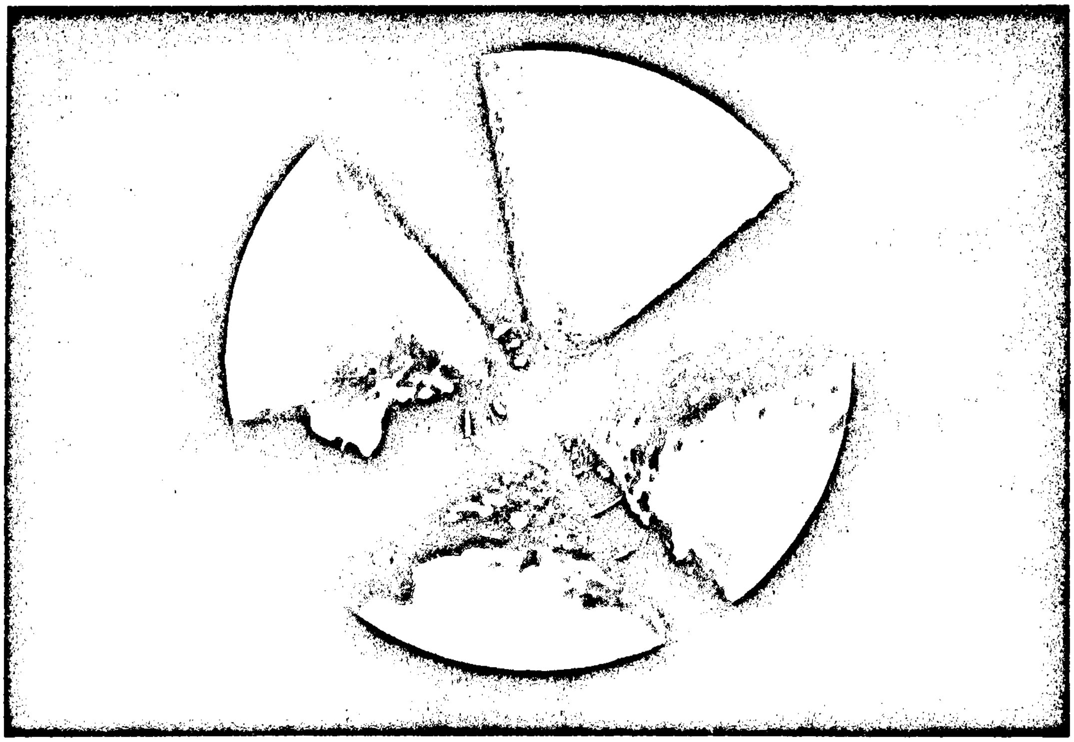


Fig. 3 : Spray flow in the post-dryout region

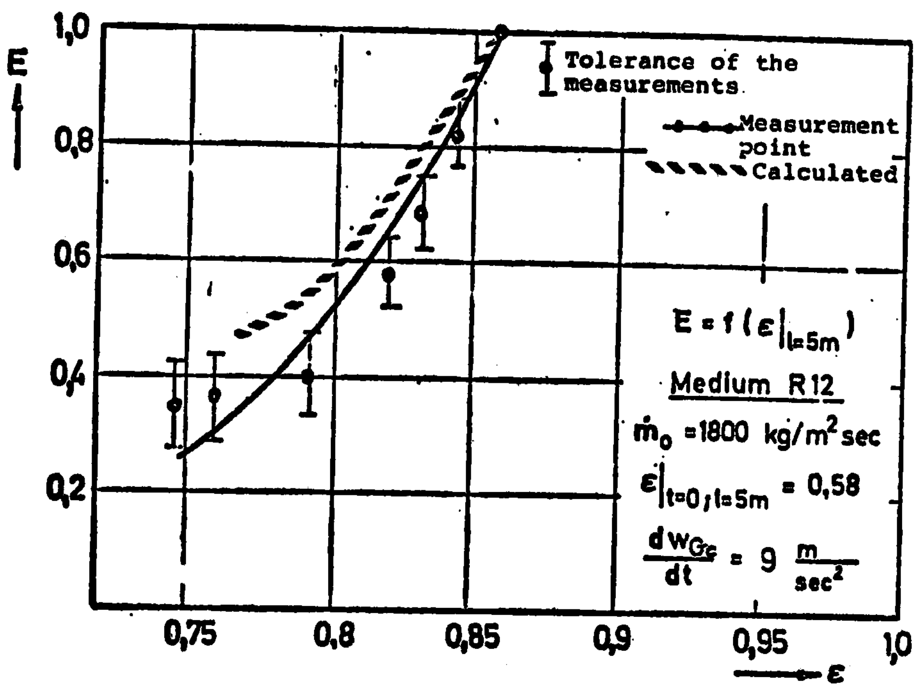


Fig. 4 : Comperision between measurd and calculated entrainment conditions

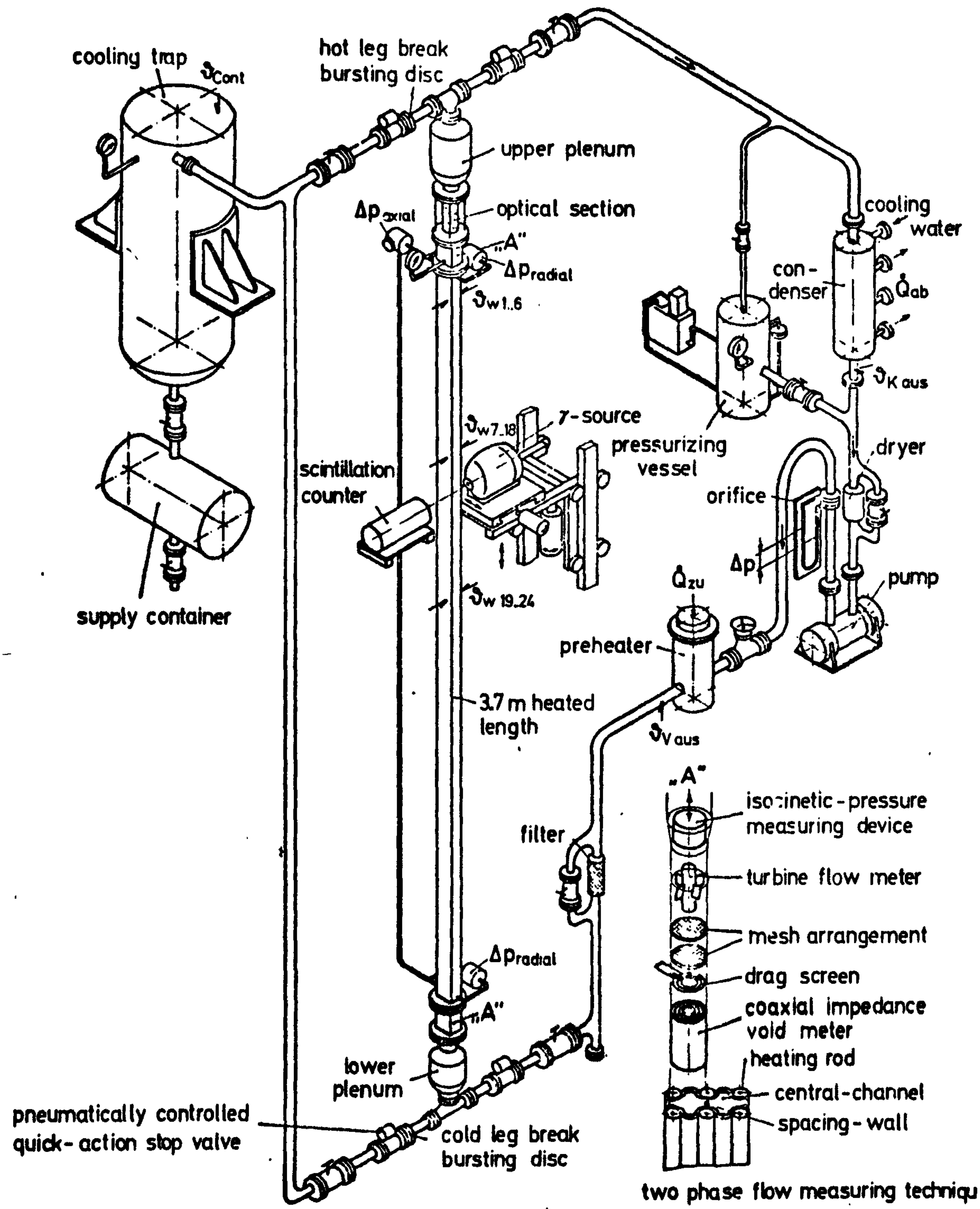


Fig. 5 : Refrigerant test loop for two channel steady state and transient two phase mixing measurements

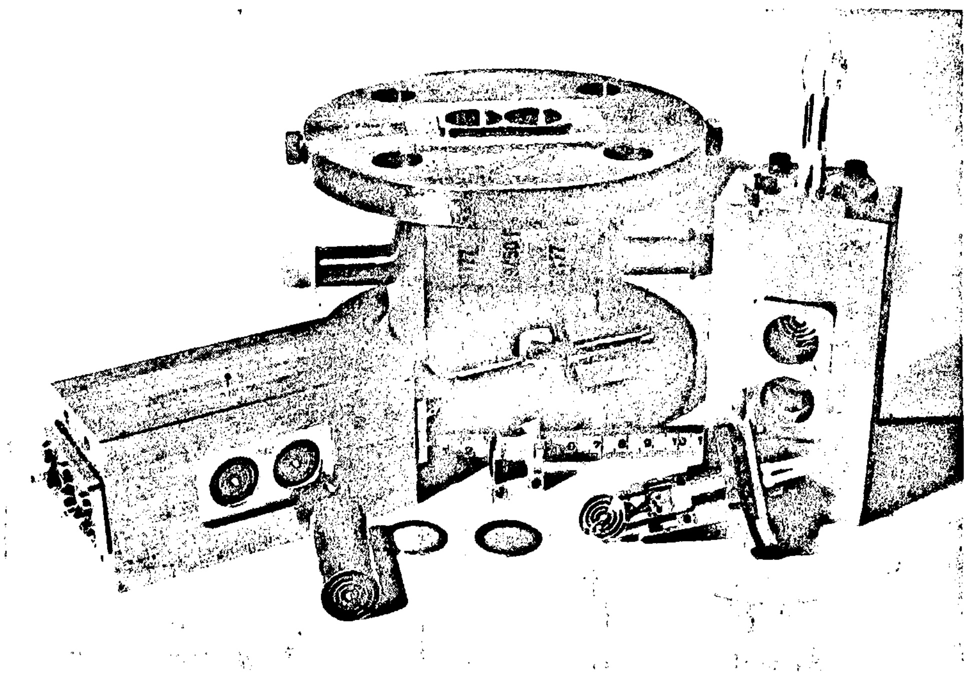


Fig. 6

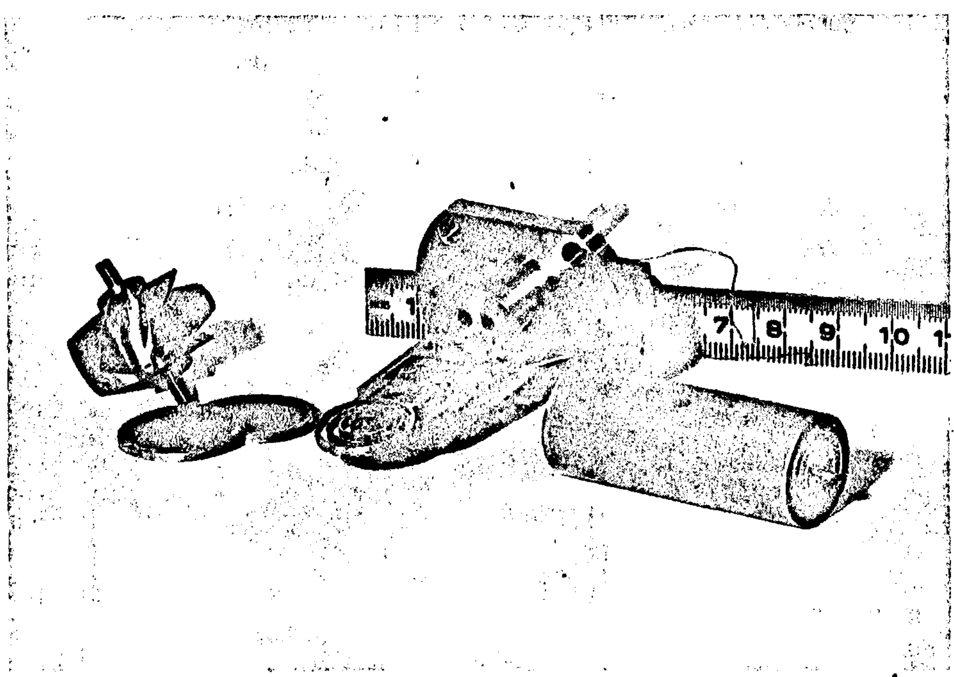
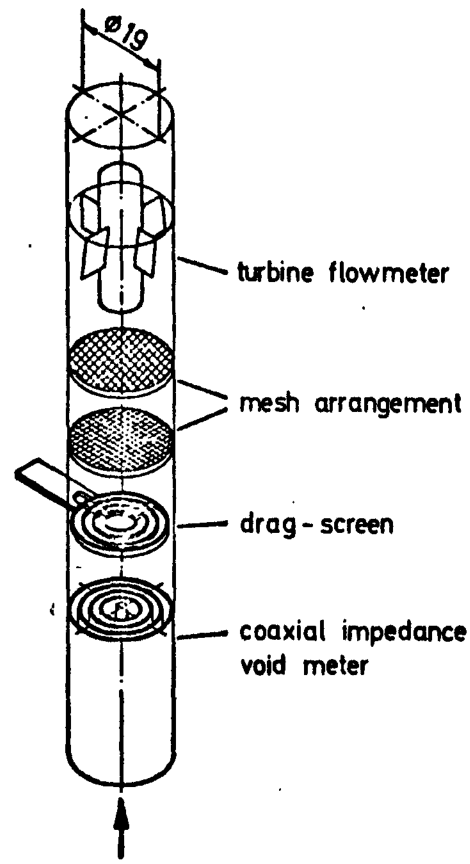
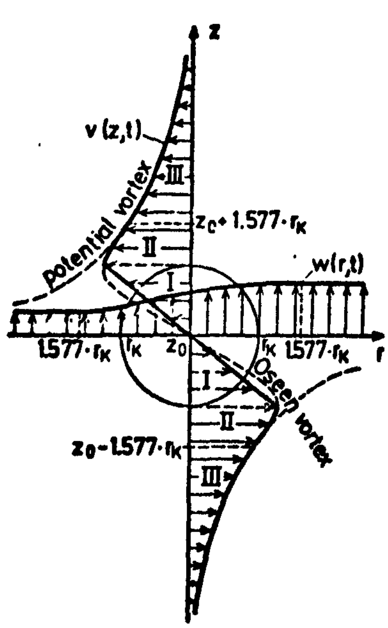


Fig. 7 : Two-phase mass flow maesuring technique with a coacial impedance void meter, drag screen, mesh arrangement and turbine flowmeter



<u>Fig. 8 :</u> Mass flow measuring technique for steady state and transient two phase flow



$$\frac{\partial w}{\partial t} = 2\cdot\nu\cdot\frac{\partial \omega}{\partial r} \quad (1)$$

$$v(r,t) = \frac{\Gamma_\infty}{2\cdot\pi\cdot r}\cdot\left[1-\exp\left(\frac{-r^2}{4\cdot\nu\cdot t}\right)\right] \quad (2)$$

$$0 \leqq r \leqq r_K \qquad v_n(r,t)_I = \frac{\Gamma_\infty}{2\cdot\pi\cdot r_K}\left[1-\exp\left(\frac{-r_K^2}{2\cdot\nu\cdot t}\right)\right]\cdot\frac{r}{r_K} \quad (3)$$

$$r_K \leqq r \leqq 1{,}577\, r_K \qquad v_n(r,t)_{II} = \frac{\Gamma_\infty}{2\cdot\pi\cdot r}\left[1-\exp\left(\frac{-r^2}{2\cdot\nu\cdot t}\right)\right] \quad (4)$$

$$1{,}577\cdot r_K \leqq r \qquad v_n(r,t)_{III} \approx \frac{\Gamma_\infty}{2\cdot\pi\cdot r} \quad (5)$$

$$\boxed{\dot{m}'_K = (s-2\cdot\Delta)\int_z^{z+\Delta z}\bar{\rho}''\cdot\sum_{n=1}^{m} v_n(z,t)\,dz} \quad (6)$$

$$\bar{\rho}'' = \begin{cases}\bar{\rho}_i \text{ if } v_n(z,t) \geqq 0\\ \bar{\rho}_j \text{ if } v_n(z,t) < 0\end{cases} \quad (7)$$

$$n = \begin{cases} i_j \text{ if } v_n(z,t) \geqq 0\\ j_i \text{ if } v_n(z,t) < 0\end{cases} \quad (8)$$

<u>Fig. 9 :</u> Equations for the new mixing modell

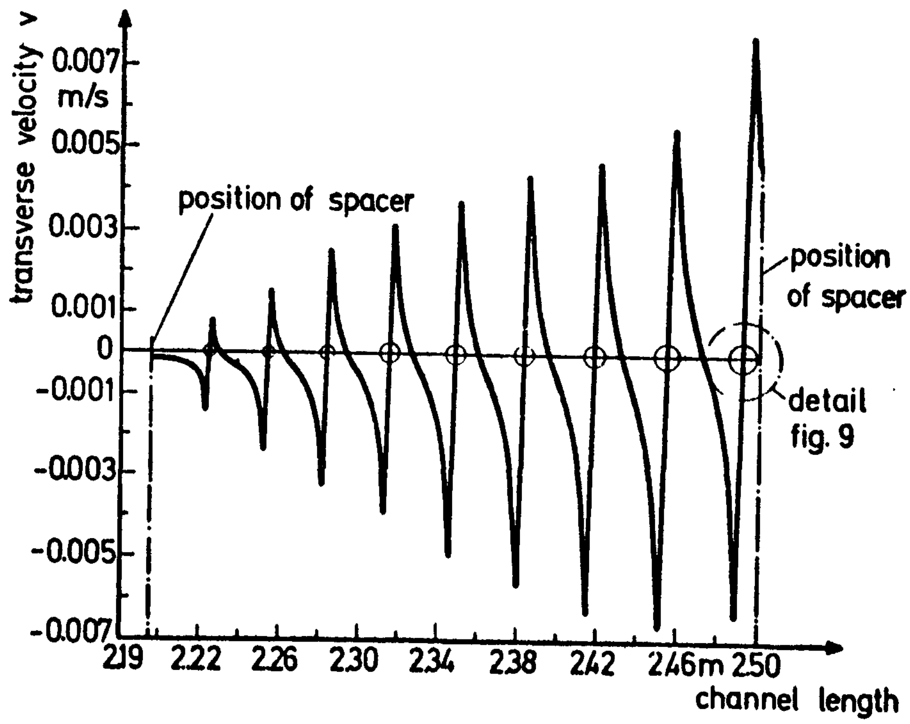


Fig. 10 : Velocity distribution in a plane shear flow (calculated)

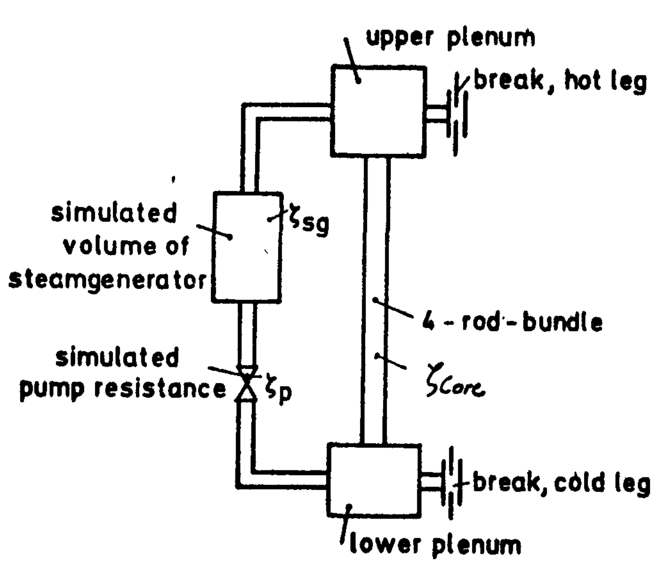


Fig. 11 : Schema of the Blowdown Test facility

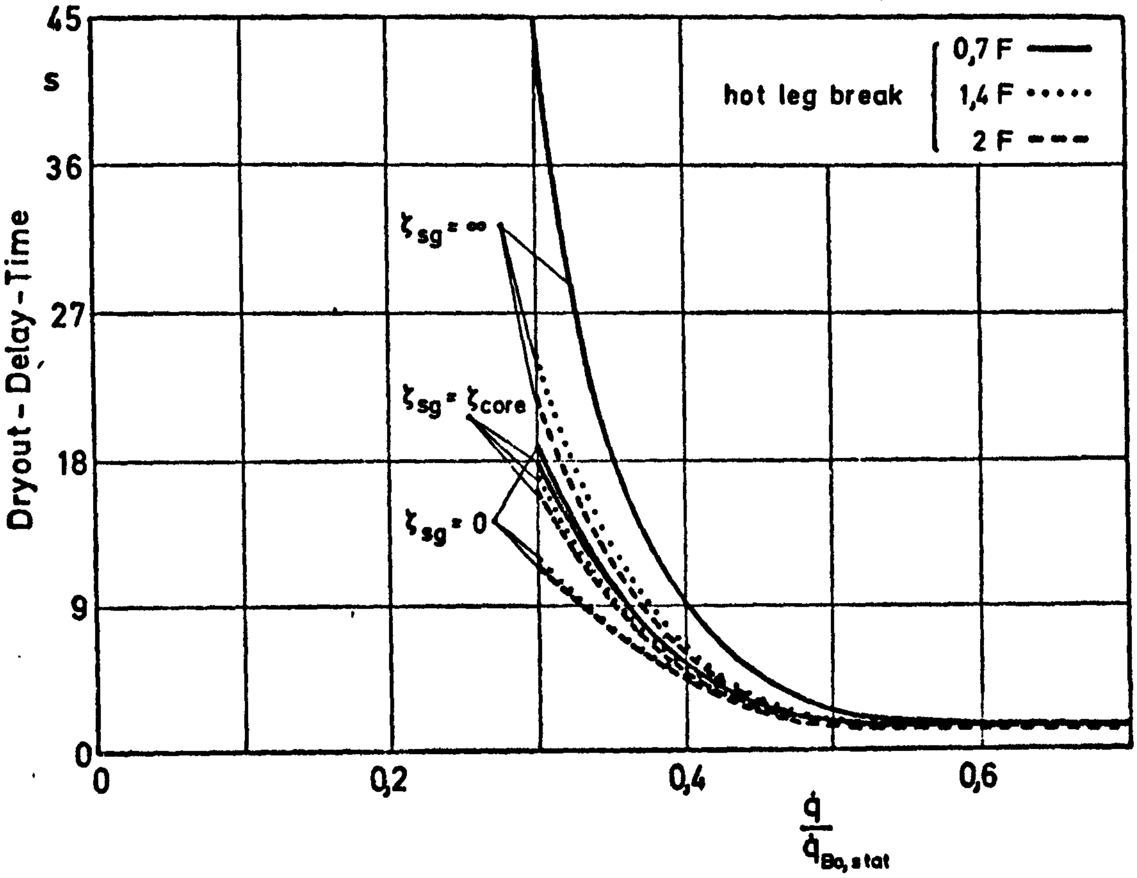


Fig. 12

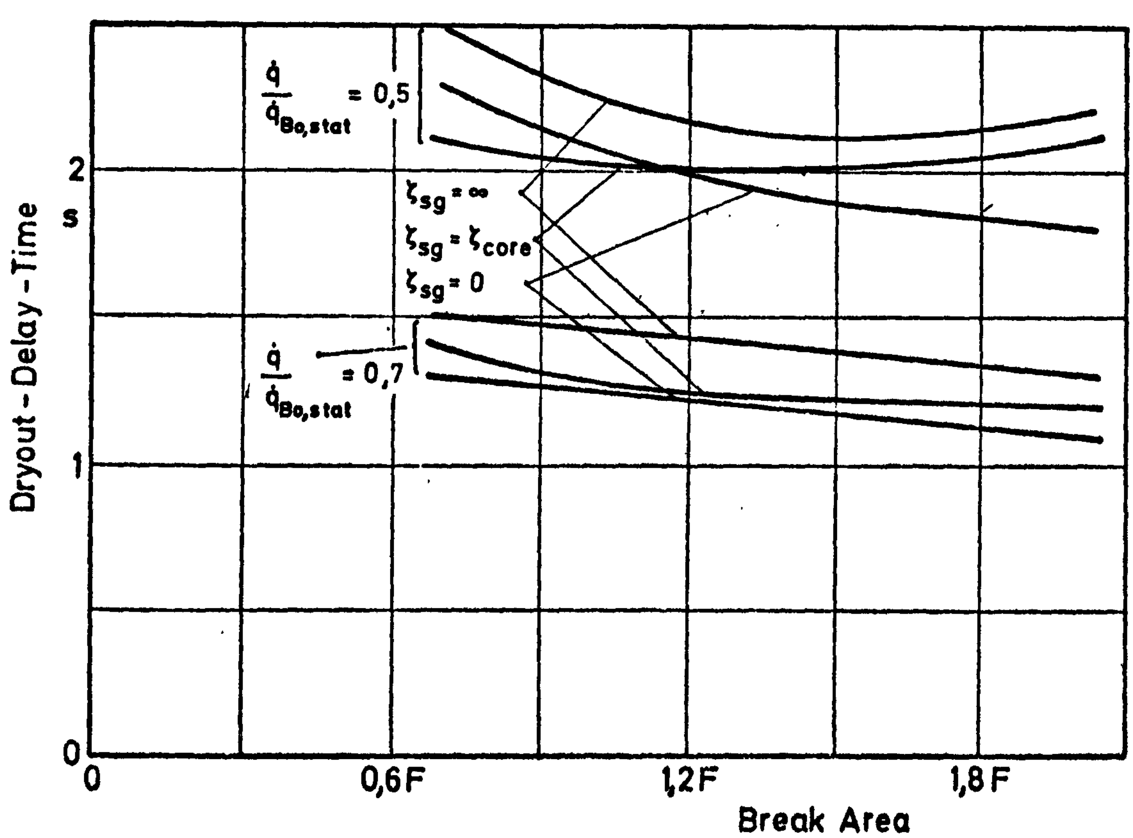


Fig. 13 : Dryout delay-times for different steamgenerator flow restistance; hot leg break

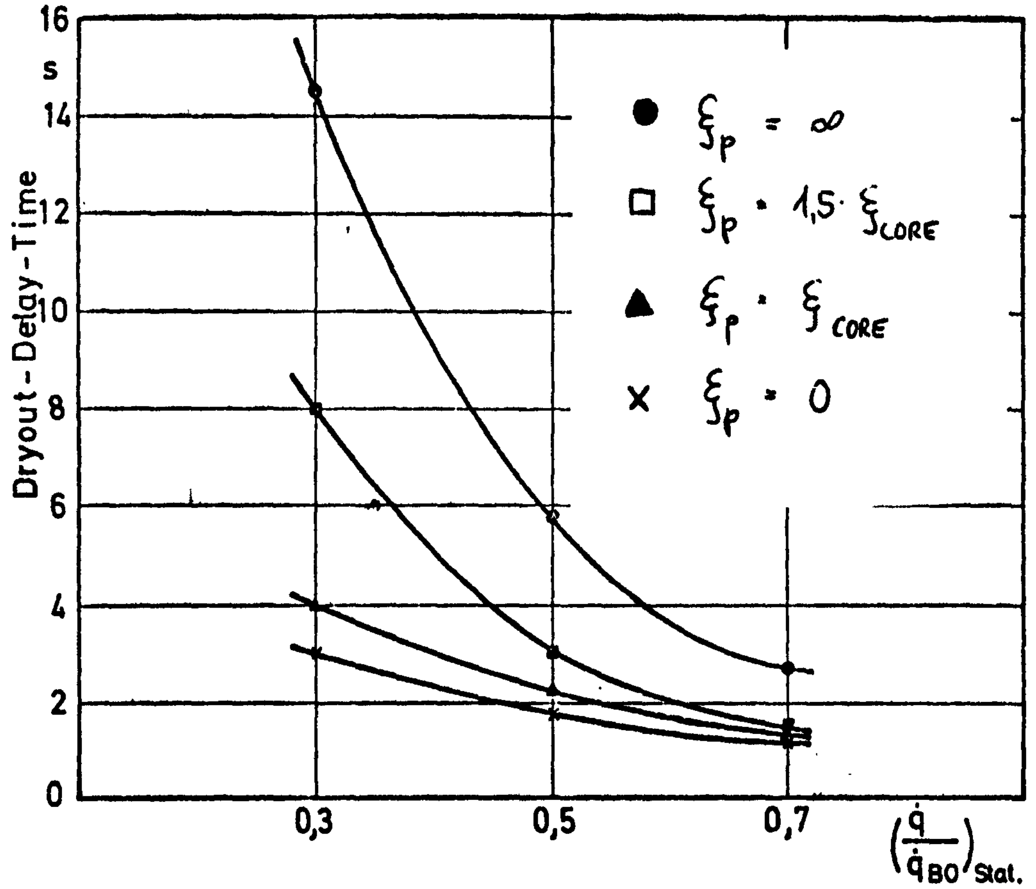


Fig. 14 : Dryout delay time with the middle for rod bundle for different pump resistance, 0,7 F - cold leg break

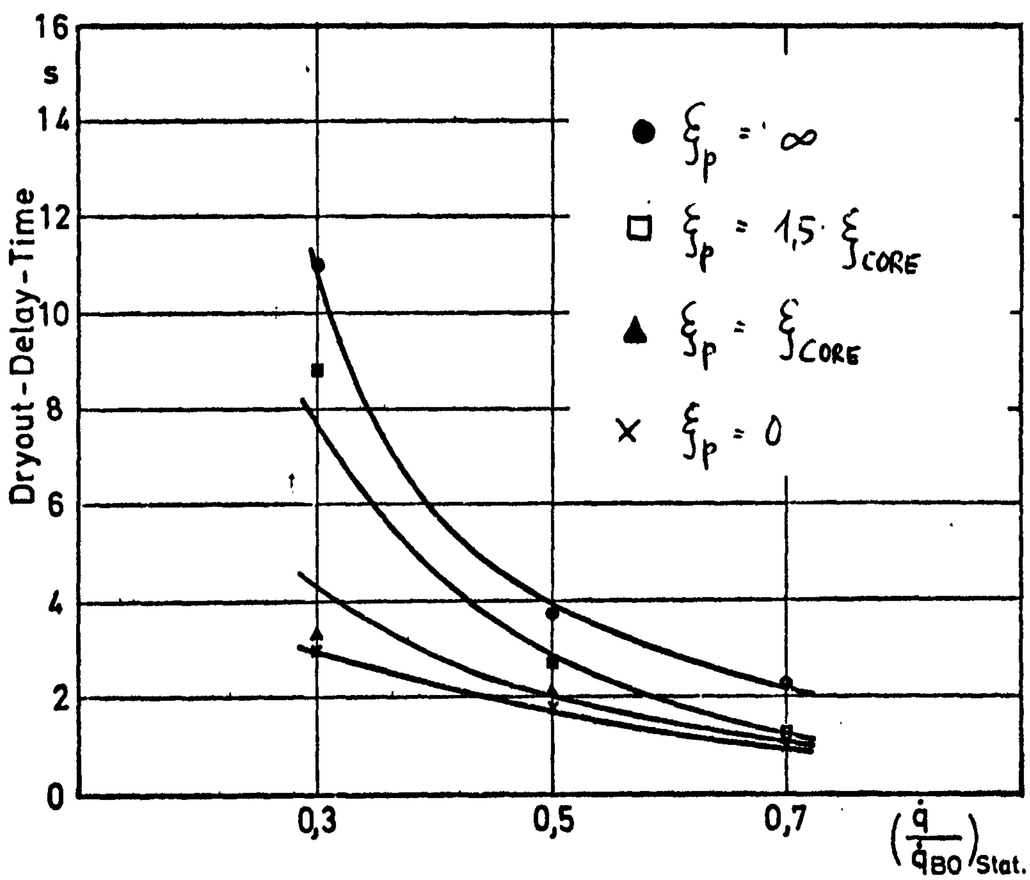


Fig. 15 : Dryout delay time on the lower rod bundle for different pump resistance, 0,7 F - cold leg break

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	1.1.2	RS 135 A

Vorhaben/Project Title	
Entwicklung von Meßverfahren zur Bestimmung transienter Massenströme (Wasser/Dampf) durch Signalkorrelation Development of methods for measuring transient two-phase flows (steam/water) by signal correlation	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Technische Universität Berlin, Institut für Kerntechnik

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.1978	31.12.1978	Prof.Dr.-Ing.U.Wesser
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	Dec. 31, 1978	

1. General Aim

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

2. Particular Objectives

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

3. Research Program

The investigation program consists of measurements on the KFK Test Loop (RS-145), which enables investigation of the correlation procedure over a large range of two-phase flow regiones to determine the range of applicability, and also measurements on the IKT Blow-Down facility, where the density measurement procedure from RS-135 may be employed.

4. Computer Codes

Program VELTRA evaluates the temperature signals by Time Series Analysis. The cross-correlation is calculated, its peak is located and finally the velocity characteristics are graphically displayed.
Program DENTRA determines the variation of fluid density with time.
Finally MATRA calculates the total flow mass by integration of results obtained from VELTRA and DENTRA.
The IKT rig was described in point 5.4 in the fourth quarterly report of 1976.

5. Progress to Date

In the period of time with which this report is concerned, extensive measurements were made on the Blow-Down Facility at the IKT (Institute for Nuclear Engineering, Berlin) and at KFK-Karlsruhe. The aim of these measurements was to test the previously developed measurement and analysis procedures. Most measurements at the IKT were made close to the system's limiting pressure of 50 bar. The time series analysis (the VELTRA program) was modified in its mathematical conception which led to some improvement in the analysis results.

6. Results

The series of experiments on the Blow-Down rig at the IKT at high pressures (50 bar) and higher fluid velocities in the test stretch showed clearly continuous temperature signal patterns. The improvement in the signals was made possible by better heat transfer between fluid flow and thermocouples. The transfer behaviour was similarly improved, and the thermocouple spectrum shifted to a higher frequency range.
This frequency shift made the selection of a better temporal resolution for analysis purpose possible [0.25 s (50 bar) and 1.25 s (12 bar)]. Three clearly distinguishable stages were to be seen in the variation of mass flow with time. (Diag. 1)
At the beginning of the flow is a short one phase stage (high density water). Later commences a two phase flow condition, characterized by a mass flowvalue which approaches constancy. At

the end of the run, the boiler is emptied of water and a pure steam flow begins, accompanied by a sharp pressure drop in the boiler and consequently a sharp increase in the specific volume. As well as being able to compare the thermal probes with the condenser (Reimann/Müller) probes, it was also for the first time possible to test the procedure at high flow rates. The signal from both kinds of probes were recorded on magnetic tape and evaluated using the program system VELTRA. The analysis results in similar coherence functions between the two detector signals. The sharp drop in the functions observed in both cases is found at ca. 40 Hz (Diag. 2). The frequency limit of the condenser probes is considerably higher, but in both cases the fall-off frequency is the same - the frequency content is thus clearly determined by the flow conditions. One may thus conclude that the signal's frequency limit is determined by the coherent frequency content of the flow, rather than by the probes themselves.

The excisting evaluation system was reworked and in part newly formulated. The changes concern the mathematical aspects of the VELTRA program system. This program performs a time series analysis of the probe signals, which yields a cross correlation function and further a velocity value. The digitalized probe signals are transformed into the frequency realm by use of the Fast-Fourier-Transform, yielding complex spectral density functions X and Y. The modification in the program is that these functions are normalised thus: $\bar{X} = X/|X|$; $\bar{Y} = Y/|Y|$ and then weighted with an idealised coherence function γ^*. (Diag. 3)

The normalised, weighted spectral density functions then have the form:

$$nx = (X/|X|) \cdot \gamma^*$$

$$ny = (Y/|Y|) \cdot \gamma^*$$

The determination of the CCF proceeds via the inverse FFT of the cross spectral density function, which in turn is obtained from the multiplication of the spectral density functions nx and ny.

The function's peak can be emphasised by exponentialising the CCF. By this modified procedure, the peak can be reliably located by appropriate software. The normalisation and weighting do not result in information loss as the location of the maximum value of

the CCF is determined by the phase characteristics of the cross spectral density function Gxx.
Further advantages are obtained through the digital filtering, wherebye the optimal cut off frequency is determined after the A-D convertion by the weighting function. Elimination of interference and equalisation of differing signal level amplitudes are obtained through the normalisation.
Initial comparisons of velocity value obtained from present and previous methods are shown in diags. 4 a-b. The example taken is that of a pump shut of own experiment.

7. Next Steps

The measurements at the Blow-Down rig will be continued. Further measurements at the KfK-rig are planned.

8. Relations to Other Projects

1) Measurements at the "Joint testing rig" of the KfK, Project RS 145
2) Using of the measurement-method at the Blow-Down rig at EURATOM-Ispra, Project RS 109

9. References

-

10. Degree of Availability of the Reports

-

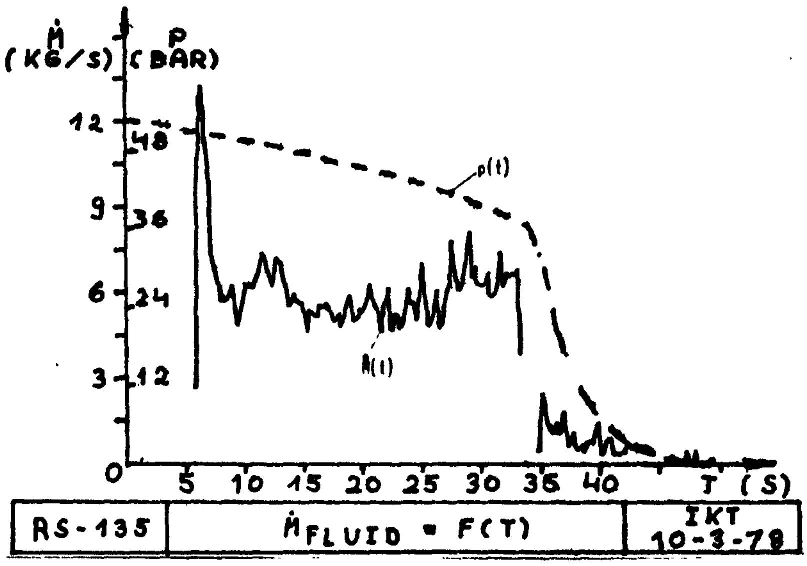


Bild 1: Massenstrom Ṁ und statischer Druck p als Funktion der Zeit;
Massflow Ṁ and static pressure p as function of time

Statischer Druck (static pressure) = 49 bar

Strömungsquerschnitt (flow cross section) = 4.9 cm^2

Masse (mass) = 181,9 kg

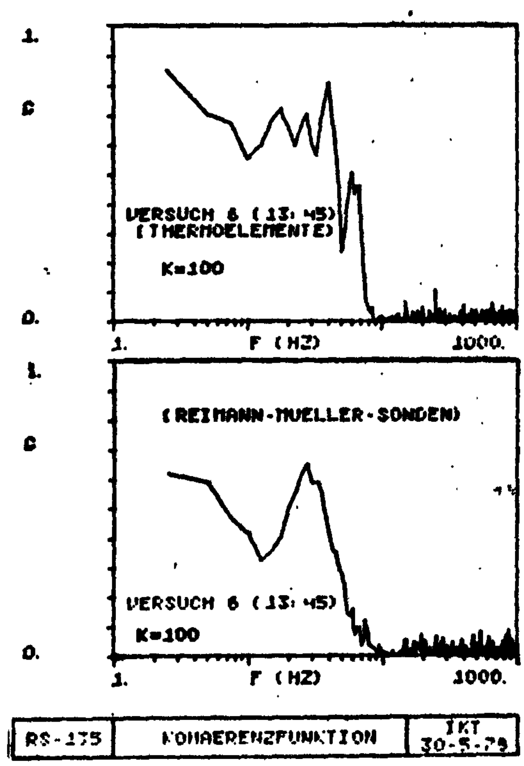


Bild 2: Vergleich der Kohärenzfunktion von Thermoelement- und Kapazitätssondensignalen

Comparison of the coherence-function from thermocouple- and condenser probe signals

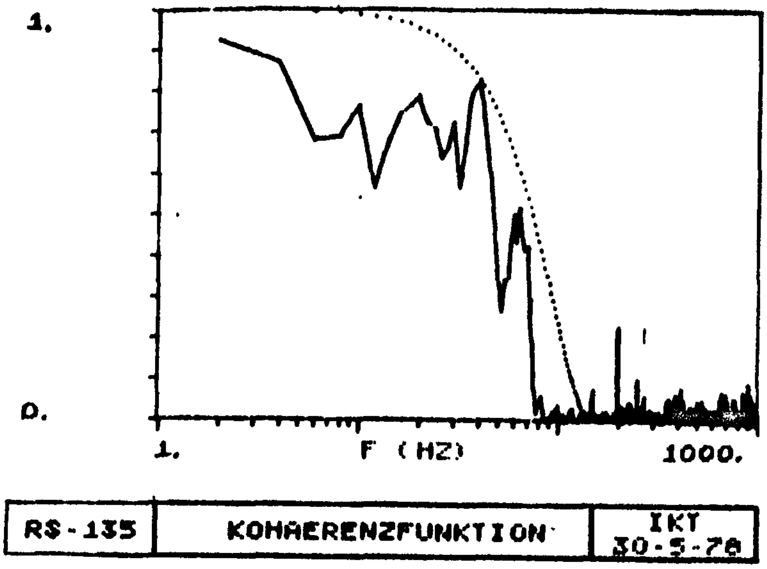


Bild 3: Kohärenzfunktion und idealisierte Kohärenzfunktion
Coherence function and idealised Coherence function

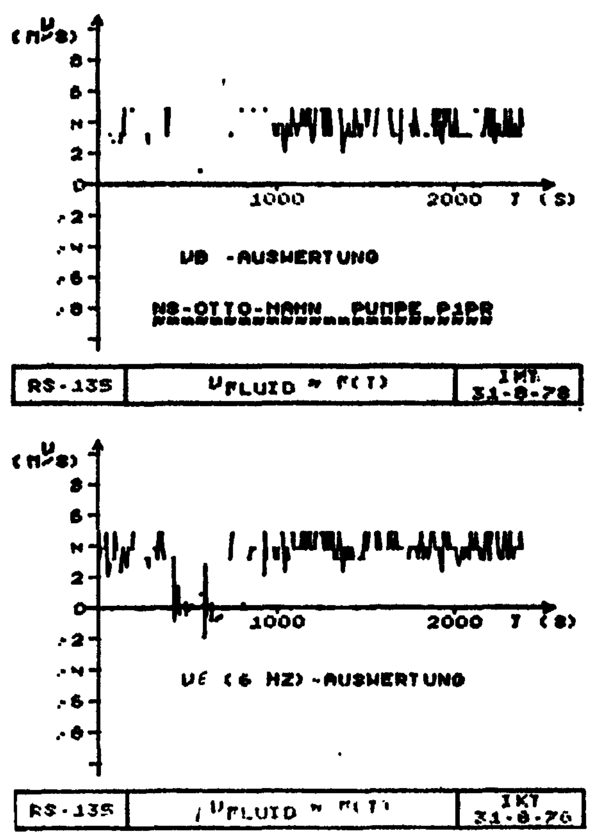


Bild 4a-b. Pumpenabschaltversuch NS-Otto-Hahn nach alter Analysemethode (a) und mit modifiziertem mathematischen Ansatz (b)

Pump shut down experiment NS-Otto-Hahn by old evaluation method (a) and after mathematical modification

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	1.1.2	RS 263/4

Vorhaben/Project Title	
Analytische Tätigkeiten der GRS im Rahmen des Reaktorsicherheitsforschungsprogramms des BMFT Hochdruck-Wärmeübergang Analytical Activities of the GRS in the Frame of the BMFT Program on Reactor Safety High Pressure Heat Transfer	Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.1977	31.12.1979	DI I. Vojtek
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	31.12.1978	

1. General Aim

Parallel and supplemantary analytical investigation to experimental research projects sponsored by the BMFT as well as further development of computer codes concerning reactor safety.

2. Particular Objectives

Post experimental analysis of 25-rod-bundle heat transfer tests, which were performed at the KWU test facility in Großwelzheim (RS 37C). Investigation of the correlations for the calculation of film-boiling heat transfer coefficients (HTC) and the critical heat flux (CHF).

3. Research Programm

3.1 Recalculation of DNB and post-DNB experiments performed, and comparison of calculated and measured results.

3.2 Comparison of heat transfer coefficients.

3.3 Assessment of HTC and CHF correlations used.

3.4 Comparison of results obtained with other similar experiments.

4. Test Facilities, Computer Codes

Test Facilities: KWU 25-rod-bundle heat transfer test facility.

Computer Codes: BRUDI-VA.

5. Progress to Date

The post-experimental calculations of DNB and post-DNB experiments performed with computer code BRUDI-VA, have been finished in April 1978. The important results, test facility, test program, and computer code are described in /1/.

6. Results

For the calculation of post-CHF heat transfer coefficients the following correlations have been used:

- Dougall-Rohsenow correlation,
- Modified Dougall-Rohsenow correlation,
- Miropolskij correlation,
- Groeneveld correlation,
- Condie-Bengston correlation.

The correlation, which led to the best agreement between measured and calculated wall temperatures for the entire range of test parameters, was the Condie-Bengston IV correlation.

The following correlations for the calculation of CHF have been employed in computer code BRUDI-VA:

- Westinghouse-W3 correlation,
- Babcock-Wilcox B & W-2 correlation,
- General-Electric correlation,
- MacBeth correlation,
- Healzer-Hench-Hansen-Levy correlation,
- Barnett correlation,
- Huges correlation,
- Hench-Levy transient correlation,
- Israel-Casterline-Matzner correlation,
- Smolin correlation.

The results obtained have shown that non of these correlations predicted DNB time for transients without flow reversal accurately.

7. Next Steps

Comparison of the results of RS 37C with other similar experimental investigations, such as the ORNL heat transfer tests, ROSA program at JAERI etc.

Recalculation of RS 37 tests with non-equilibrium codes.

9. References

/1/ Auswertung der 25-Stabbündel Versuch (RS 37C) mit dem Rechenprogramm BRUDI-VA. GRS-A-208, September 1978

10. Degree of Availability of the Reports

GRS-A-... reports are available through the Gesellschaft für Reaktorsicherheit (GRS) mbH, Forschungsbetreuung, Glockengasse 2, D-5000 Köln 1.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	1.1.2	RS 263/5
Vorhaben/Project Title Analytische Tätigkeiten der GRS im Rahmen des Reaktorsicherheitsforschungsprogramms des BMFT Hochdruckthermohydraulik Analytical Activities of the GRS in the Frame of the BMFT Research Program on Reactor Safety High Pressure Thermohydraulics		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH
Arbeitsbeginn/Initiated 1.1.1977	Arbeitsende/Completed 31.12.1980	Leiter des Vorhabens/Project Leader Dr. W. Winkler
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 31.12.1978	Bewilligte Mittel/Funds

1. General Aim

Parallel and supplementary analytical investigation to experimental research projects sponsored by the BMFT as well as further development of computer codes concerning reactor safety.

2. Particular Objectives

Analytical investigations of the influence of PWR primary loops on blowdown parallel to tests, e.g. description of the influence of

- break size,
- three different break positions,
- stored heat in structure,
- pumps,
- injection.

3. Research Program

3.1 Meetings of working group LOBI-GRS.

3.2 Proposal of test matrix for LOBI program A.

3.3 Coordination between concerned parties BMFT, GFS, GRS, KWU.

3.4 Calculations w.r.t. influence of downcomer gap-width on blowdown.

3.5 Calculation of electrical heating power.

3.6 Calculation of flow-loss-coefficients.

3.7 Analytical investigation of the influence of heat stored in structure on blowdown.

3.8 Adaption of computer programs to test facility.

4. Computer Codes

Ad 3.4,
3.5,
3.7 Code BRUCH-D06.

To 3.8 Codes BRUCH-D06, DRUFAN, RELAP4-GRS.

5. Progress to Date

Ad 3.1 6 meetings of working group LOBI-GRS. (preperation, execution and minutes /1/)

Ad 3.2 Establishment of matrix proposal for test program A.

Ad 3.3 Establishment of work plan.

Ad 3.4 Blowdown calculations for LOBI facility with 7, 12 and 50 mm downcomer gap-width, and for reference reactor BIBLIS-B. Comparison of calculated results.

Ad 3.5 Calculation of heating power for break sizes 2x1A, 2x0.5A and 0.02A /2/.

Ad 3.6 Calculation of flow-loss-coefficients according to VDI-Atlas.

Ad 3.7 Blowdown calculations with and without consideration of stored heat in structure.

Ad 3.8 Compilation of input data.

6. Results

Ad 3.1 Discussion of
- experimental applicabilities of test facility,
- test parameters,
- necessary improvements of test facility for small leak tests, high and low pressure injection tests.

Definition of
- matrix for pre-tests,
- reference test for test program A.

Ad 3.2 GRS-proposal for test matrix A as basis for discussions in working group.

Ad 3.3 Comprehensive presentation of necessary theoretical work and distribution of it between BMFT, GFS, GRS and KWU.

Ad 3.4 Corresponding to the LOBI scaling factor of 1:712 w.r.t. volume a reactor typical blowdown can be achieved with a 7 mm downcomer. The 50 mm downcomer results in a delayed blowdown compared to a reactor blowdown. With a 12 mm downcomer the reactor pressure loss of 1.1 bar in downcomer is represented. Due to expected adjustment problems a 12 mm downcomer is agreed upon instead of a 7 mm downcomer.

Ad 3.5 Calculations with break sizes 2A, 1A and 0.02A show that simulation of fuel rods with direct heated rods is possible by controlling heating power. Good agreement with the behavior of fuel rods can be achieved by controlling power according to a DNB dependent power versus time curve.

Ad 3.6 Pressure distribution in the LOBI facility based on theoretical flow-loss-coefficients.

Ad 3.7 Stored heat in the structure of LOBI facility is of minor influence on blowdown.

Ad 3.8 Up dated code input data.

7. Next steps

Meeting of working-group LOBI-GRS.
Completion of test matrix.
Calculation of heating power with code BRUCH-D06.
Analysis of pre-tests with code BRUCH-D06 w.r.t. flow-loss-coefficients.
Comparison of calculated flow-loss-coefficients and coefficients resulting from measured pressure losses.
Adaption of codes BRUCH-D06, DRUFAN and RELAP4/Mod6 to test facility characteristics.
Specification of standard problem LOBI.
Analysis of small leaks with code RELAP4-Mod6.
Pre-test and post-test blowdown calculations.

8. Relation with other Projects

Semiscale, LOFT, RS 37, LOBI (RS 109).

9. References

/1/ Ergebnisprotokolle 1. (bis 6.) Sitzung der AG LOBI-GRS.

/2/ W. Pointner, F.J. Ringer: Blowdown-Experiment RS 109 (LOBI), Berechnung der elektrischen Heizleistung. GRS-A-205, September 1978

10. Degree of Availability of the Reports

GRS-A-... reports are available through the Gesellschaft für Reaktorsicherheit (GRS) mbH, Forschungsbetreuung, Glockengasse 2, D-5000 Köln 1.

Ergebnisprotokolle: Restricted distribution.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	1.1.2	6.01.03/03A (PNS 4137
Vorhaben/Project Title Gemeinsamer Versuchsstand zum Testen und Kalibrieren verschiedener Zweiphasen-Massenstrom-Meßverfahren Joint Test Rig for Tests and Calibration of Different Methods of Two-Phase Mass Flow Measurement		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe mbH (KfK) Projekt Nukleare Sicherheit (IRB)
Arbeitsbeginn/Initiated 1.10.1974	Arbeitsende/Completed 31.12.1978	Leiter des Vorhabens/Project Leader J. Reimann
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating Dec. 1978	Bewilligte Mittel/Funds

1. <u>General Aim</u>

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

2. <u>Particular Objectives</u>

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

3. <u>Research Program</u>

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

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

4. <u>Experimental Facilities</u>

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

5. <u>Progress to Date</u>

The following experiments have been performed:

- Test of the LOFT DTT (modular and plenum DTT) combined with a 3 beam gamma densitometer in steam-water flow at low qualities.
- Comparison of the Semiscale Scanning Densitometer and the KfK-IRB 5 beam gamma densitometer including both air-water and steam-water flow.
- Test of the True Mass Flow Meter (TMFM).(KfK-PNS 4236) in steam-water flow at pressures up to 10 MPa.

- Investigations at highly accelerated (nozzle) two-phase flow (KfK-PNS 4126).
- Optimization of flow homogenizers (Battelle Frankfurt) for horizontal two-phase flow.

6. Essential Results

- The tests of the LOFT instrumentation performed in the 4th quarter of 1977 have been analysed, the final report will be completed in the near future. Some of the results differed significantly from the correct flow values, depending on the phase distribution in the pipe cross section. The radio-tracer technique combined with the LOFT densitometer showed a much better accuracy, independent of flow regime.
- The results obtained with the two reference densitometers in general agreed very well, in 80 % of all experiments the deviations were below 3 %.
- The TMFM tests, performed at mass flow rates up to 3.5 kg/s and qualities up to 55 % had a mean measuring error of 3.4% of the maximum measuring range with a standard deviation of 2.2%.
- To produce a highly accelerated flow, a nozzle geometry was used. Experiments in air-water flow ($p \lesssim 1$ MPa) and steam-water flow ($p \lesssim 13$ MPa) were made and data evaluation is underway.
- Experience on the qualitative influence of various flow homogenizers was gained by using a lucite test-section. To quantitatively measure the phase distribution downstream of the homogenizers in the subsequent experiments, the 5 beam gamma densitometer was successfully used.

7. Next Steps

- Layout of a test facility for transient experiments.
- Optimization of flow homogenizers; test of a drag body in homogenized flow.
- Documentation of the tests of the EG&G mass flow rate instrumentation.

8. Relation with Other Projects

RS109, 135, 136, 146, 147, PNS 4236, LOFT(INEL), Semiscale(INEL), PBF(INEL).

9. References

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

Berichtszeitraum/Period 1.1. - 30.9.1978	Klassifikation/Classification 1.1.2	Kennzeichen/Project Number RS 225
Vorhaben/Project Title Dichtemessung in Zweiphasenströmung (Wasser/Dampf) mittels Ultraschallsonden. Density-measurement in a two-phase-flow (water/steam) by means of ultrasonic detectors.		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Technische Universität Berlin, Institut für Kerntechnik
Arbeitsbeginn/Initiated Oct. 1, 1976	Arbeitsende/Completed Sep. 30, 1978	Leiter des Vorhabens/Project Leader Prof.Dr.-Ing. U. Wesse
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating Dec. 31, 1978	Bewilligte Mittel/Funds

1. General Aim

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

2. Particular Objectives

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

3. Research Program

The research program is devided into four (4) interconnected activities:

3.1 Design and testing of alternative probe configurations.

3.2 Design and optimization of transducers and electronic circuitry.

3.3 Density measurements (stationary/transient)

3.4 Fluid-level measurements (stationary/transient)

4. Experimental Facilities, Computer Codes

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

5. Progress to Date

5.1 Probe Design

The conditions on getting continous or RF-Pulse receiver signals with least sensitivity to extraneous disturbences and optimum interpretation features in dependency on the surrounding medium were investigated experimentally for testing probes of different helical geometry.
By means of theoretical models the ultrasonic propagation characteristics of helical rods were estimated concerning the dispersion-/frequency behaviour.
Computer programs to evaluate the characteristic matrix and the dynamical displacements(along the helix axis) were developed.
The possibility of mechanically extracting the probe signals from the testing environment via straight coupling-rods were examined. With segmented (notched) rods of straight geometry experimental investigations were performed.

5.2 Transducers-/Electronics Design

By means of a theoretical transfer-model the mechanical and electrical parameters to achieve acoustical-electrically matched transceiver systems were investigated.
A computer program was developed and theoretical and experimental results compared.

5.3 Density Measurements

The attenuation characteristics for specific operating frequencies with regard to the density information was investigated concerning linearity and possible disturbences.
The influence of instationary and constant (different) temperature was examined experimentally.

5.4 Fluid-level Measurements

Fluid-level curves (RMS-amplitude vs. submersed surface) within stationary and instationary surrounding were recorded for design frequencies of probes of different active lengths (200 - 800 mm).

5.5 Final Report

A final report was completed.

6. Results

6.1 Probe Design

The experimental investigations showed that on specific conditions as well optimal disturbence-insensitive als reliably interpretable receiver signals (RMS-values) can be achieved. These conditions are essentially system-dependent (material/geometry/operating conditions).
In single-phase as well as two-phase flows (air/water) reproducable results could be achieved experimentally.
The quantification of the underlying conditions and therewith quantitative criteria for designing probes which are matched to specific measurement problems, could nevertheless only be estimated. In this context ("geometry-mode" concept):

- the total stress/displacement conditions with regard to ultrasonic propagation is to be stable against extraneous 'disturbences' (flow/temperature/two-phase condition). (rod diameters $\leq$ 4 mm generally exhibited now reproducable results).
- 'constructive' interferences of the simultaneous existing propagation modes concerning the propagation along the helix axis.
- energy concentration within those modes, which are radiative to the environment (lateral displacement of the rod cross-sections)
- operation outside the fundamental rod-resonances (SWR).

By means of theoretical models of the dispersion characteristics, predictions concerning the rspctve. frequency-wave length behaviour can be assessed. The evaluation of the characteristic matrix permits estimations of the dominant components of displacement along the helix-axis, which are important with regard to the radiation properties.
Based on the experimental verified geometry- and operating parameters, the computional comparisons demonstrated the

anticipated criterion of essentially radiative mode coupling.

The extraction of probe signals from the environment under test via straight coupling rods (i.n. because of temperature-limitation of piezoceramics) seems to be principally feasible within the noted restrictions. The experimental investigation showed, that with adequate geometrical design (notches/ segmentation) a mechanical filter effect can be achieved. Preliminary tests of notched straight rods yielded linear RMS-value vs. fluid-level curves, provided that adequate segmentation and operating conditions be used. In that case, distinct segment indication by signal-dips occurred.

6.2 Transducer- and Electronic-design

Subject to the purpose of the tests, two different concepts were required. First, to experimentally check the probe behaviour broadband transfer characteristics matching the desired frequency region has to be provided, on the other hand operational conditions concerning density-/fluid level monitoring required relatively narrowband characteristic with (limited) mechanical/electrical tuning possibility.

With regard to easy interchangeability and 'standardizing' of transducer systems the experiments yielded mechanically coupled transducers with front and rear coupling parts, the rear coupling being shaped as conical labyrinth. The capsulated system enables enclosure of electronic circuitry (filter/ preamplifier).

Starting with probe-specific excitation conditions (frequency e.g.) the mechanical-electrical design requirements of the transducer systems are given. To include and optimize the respective mechanical-electrical parameters, an equivalent network was used. Theoretical comparison yielded satisfying results. Based on probe-specific frequency option, the design and optimization of the transducer system therewith is feasible.

6.3 Density- and Fluid-level probing

The experimental results concerning density-/fluid level measurements yielded the following conclusions (cf. 6.1)

1. In principle an optimal excitation frequency, dependent on specific geometrical and operational conditions could be established for helical probes with regard to insensitivity against extraneous disturbences as well as linear relationship between signalattenuation and submersed length/density in/of surrounding medium.
   The noted linearity of the signal implies, that the mechanism of propagation of sonic vibration does not represent simple SWR, but complex travelling pulsation characteristics. Determination of the signal RMS-value following mechanical and/or electrical filtering thus permits a substantial reduction in analytic/electronic requirements and yields a more reliable interpretation compared with previous tracking methods /1, 2/.

2. To enable probe design matched to arbitrary measurement problems and operational conditions (cf. 1.) quantification of the so called 'geometry mode' conditions is substantial. Because of the complex propagation and deformation mechanisms (comprising simultaneously 6 coupled and dispersive propagation modes) predictions concerning this could only be estimated.

3. The experimental and theoretical investigations confirmed, that in using arbitrary design and system combinations (probe-transducer) essentially non systematic and incidental results are effectuated.

6.4 Conclusions

Because of the complex mode coupling of essentially dispersive propagation modes, the transfer and radiation characteristics of helical ultrasonic probes can only be estimated.
Quantitative predictions with regard to disturbence -insensitivity and least intermodulation effects (signal distortion) concerning density/fluid level information as well as parameters of influence

(flow condition temperature) can be assessed only for specific operational and design parameters.
Despite the confirmation of the existence of an 'optimal measuring characteristic' of helix-probes, the investigation of simpler geometries (e.g. notched straight rods) with regard to practical implementation seems to be more appropriate.

7. Next Steps

The work has been finished.

8. Relation to other Projects

-

9. References

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10 Degree of Availability of the Reports

-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	1.1.2	RS 277
Vorhaben/Project Title Vergleich erprobter und in Entwicklung befindlicher Massenstrommeßverfahren für Zweiphasenströmung Comparison of Proved and Currently Developed or Planned Massflow Measuring Methods for Two-Phase Flows		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor BATTELLE-INSTITUT E.V. Frankfurt am Main Abt. Thermische Verfahrenstechnik
Arbeitsbeginn/Initiated 1.11.77	Arbeitsende/Completed 29.2.80	Leiter des Vorhabens/Project Leader G. Hampel
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

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

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

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

2. Particular Objectives

3. Research Program

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

- The first phase consists in the compilation of a manual (Catalogue) for tried two-phase mass-flow measuring methods for which first experiences and results have already been gathered.
- The second phase concerns new planned measuring methods that can be used in future thermohydraulic experiments in LWR safe-

ty research.

The following main steps are scheduled:

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

3.2 Collection of data on the planned measuring methods.

3.3 Establishment of criteria for the cataloging of the methods.

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

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

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

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

3.8 Discussion of the results and final considerations.

4. Experimental facilities, Computer Codes

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5. Progress to Date

- Collection of a major part of the required information and literature.
- Elaboration of criteria lists for the systematic comparison of the measuring systems according to the general and application-oriented performance.
- Classifying of the nearly 40 individual measuring systems and establishing 15 different groups.
- Description of the general performance of 10 individual measuring systems (in several versions) and short description of their measuring principle, evaluation and calibration requirements.

6. Results

-

7. Next Steps

Continuation of the work according to the research program.

starting with 3.1.

8. Relation with Other Projects
RS 16 B, RS 33 A, RS 37 A, RS 37 C, RS 48, RS 50, RS 81, RS 93, RS 109, RS 123, RS 135, RS 136, RS 145, RS 146, RS 147, RS 161, RS 163, RS 179, RS 188, RS 195 plus other European and US-American projects.

9. References
-

10. Degree of Availability of the Reports
-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	1.1.2	RS 182
Vorhaben/Project Title Beteiligung der GRS am LOFT-Programm der USNRC Participation of the GRS at the LOFT Program of the USNRC		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH
Arbeitsbeginn/Initiated 1.4.1976	Arbeitsende/Completed 31.12.1980	Leiter des Vorhabens/Project Leader Dr. A.B. Wahba
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31.12.1978	Bewilligte Mittel/Funds

1. General Aim

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

2. Particular Objectives

2.1 Verification of computer codes by pre- and post-test calculations of the LOFT and Semiscale series. Performing modifications of the codes were necessary.

2.2 Comparison of the results of the LOFT and Semiscale series tests with the German RS projects and analysis of this comparison.

3. Research Program

The work of the GRS includes two main points:

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

4. Experimental Facilities, Codes

The codes BRUCH-D-06, RELAP4-GRS, and DRUFAN were applied to simulate the complete decompression process including the ECC injection process until the beginning of the flood phase. For the calculation of the refill and flood phase the codes RELAP4/MOD6 /1/, and REFLOS /2/ will be used.

5. Progress to Date

Post-test analysis of the LOFT experiment L1-4 with BRUCH-D-06 was completed /3/. Modifications concerning the simulation of the accumulator injection and flow resistance in the broken loop were carried out.
RELAP4/GRS was used to analyse the L1-4 /4/ and L1-5 /5/ experiments. During the post-test calculations of L1-4 it was found that the model for heat transfer under free convection (secondary side of the heat exchanger) had to be modified. For the calculation of L1-5 the core simulator in the code (RELAP4/GRS) was replaced by appropriate heat slabs which simulated the nuclear heated core.
After the implementation of RELAP4/MOD6 on the GRS computer (AMDAHL 470/V5), test calculations were done using input data model simulating the LOFT counterpart test S06-4. An input data model of the first nuclear test L2-2 for RELAP4/MOD6 was generated. The LOFT facility was simulated by the system code ALMOD /6/ in order to analyse some of the operational experiments for the nuclear test series. Steady state calculations for 100 % power were done. Transient calculations for a selection of the operational experiments are underway.
Using the basic version of DRUFAN, the L1-4 test was simulated /7/. It was the first time that DRUFAN code has been applied to an integral test with a complex system, such as the LOFT facility. Particular effects or components, like heat addition from steam generator and structural materials, pumps, pressurizer and ECC injections, were simulated by time-dependent input functions. The results of this post-test calculation as presented at the Workshop on CSNI LOCA standard problems /8/, have shown good results, especially after the start of the ECC injection. This is due to the inclusion of thermodynamik nonequilibrium effects in DRUFAN.
In order to study the fluiddynamic processes in the brocken loop in more detail, numerous variations in the flow area had to be considered. This was achieved by increasing the number of nodes and junctions, simulating the different parts of the facility. A second post-test analysis of L1-4

was also important for testing and verifying new models (heat slab model, heater rod model, heat transfer model), developed and implemented in DRUFAN.
This extended version of DRUFAN was also used to simulate the LOFT counterpart test SO6-3. The results of this analysis will be submitted to the standard problem No. 8 of the OECD.

6. Results

A comparison between results from the LOFT and Semiscale LOCA tests and their application to PWR/LOCA performance was done with EG & G and was presented at the Reaktortagung 1978, Hannover, FRG /9/. Also pre- and post-test analyses of the L1-4, using RELAP4/GRS, were presented at the Reaktortagung 1978 /10/.
The results of the post-test calculations of the L1-4, computed with BRUCH-D-O6 /11/, RELAP4/GRS /12/, and DRUFAN /8/, were presented at the Workshop on CSNI LOCA Standard Problems 1978, Garching.

7. Next Steps

Detailed analysis of the nuclear tests (L2-series), using RELAP4/GRS, DRUFAN and REFLOS. Application of fuel rod behaviour models. Computation of selected experiments from the Semiscale tests.

9. References

/1/ S.R. Fischer, et. al.: RELAP4/MOD6 - A Computer Program for Transient Thermal Hydraulic Analysis of Nuclear Reactors and related Systems. CDAP TROO3, January 1978

/2/ E.J. Kersting: Rechenprogramm REFLOS - Programm zur Berechnung des Wiederauffüll- und Flutvorgangs. GRS-A-163, September 1978

/3/ G. Lerchl, K.J. Liesch: Ergebnisse zum LOFT-Versuch L1-4 (Voraus- und Nachrechnung mit BRUCH-D-O6). GRS-A-122, März 1978

/4/ R. Ullrich: RELAP4/GRS - Analysen zum nichtnuklearen

LOFT-Versuch L1-4 (Voraus- und Nachrechnungen). GRS-A-212, September 1978

/5/ M. Firnhaber: Nachrechnungen zum nichtnuklearen LOFT-Versuch L1-5. GRS-A-252, Dezember 1978

/6/ W. Frisch, A. Höld, R. Meißner, K.D. Schmidt: ALMOD-2 - Nichtlineares Anlagenmodell zur Simulation von Störfällen in Druckwasserreaktoren, Modellbeschreibung. GRS-A-158, August 1978

/7/ W. Buhl, K.J. Liesch: Ergebnisse zum LOFT-Versuch L1-4 (Nachrechnung mit DRUFAN). GRS-A-243, Dezember 1978

/8/ W. Buhl, K.J. Liesch: Results of the Post-test Calculation of the L1-4 Test Standard Problem Calculated with the Code DRUFAN. Presented at the Workshop on CSNI LOCA Standard Problems, Garching, FRG, April 1978

/9/ L.P. Leach, D.J. Olson, E.F. Hicken: Application of Results from the LOFT and Semiscale LOCA Tests to Reactor Safety Questions. Reaktortagung, Hannover, 1978

/10/ R. Ullrich: Analyse isothermer LOFT-Blowdownexperimente mit Notkühleinspeisung. Tagungsbericht des DATF zur Reaktortagung 1978, S. 185-188

/11/ G. Lerchl, K.J. Liesch: Results of the Pre-test and Post test Calculations of the L1-4 Test Standard Problem Calculated with the Code BRUCH-D-06. Presented at the Workshop on CSNI LOCA Standard Problems, Garching, FRG, April 1978

/12/ R. Ullrich: RELAP4/GRS Calculations for LOFT L1-4. Presented at the Workshop on CSNI LOCA Standard Problems, Garching, FRG, April 1978

10. Degree of Availability of the Reports

GRS-A-... reports are available through the Gesellschaft für Reaktorsicherheit (GRS) mbH, Forschungsbetreuung, Glockengasse 2, D-5000 Köln 1.

Berichtszeitraum/Period 01.01. - 31.12.1978	Klassifikation/Classification 1.1.2	Kennzeichen/Project Number RS 179
Vorhaben/Project Title Erstellung eines theoretischen Phasenseparationsmodells zur Erfassung des Wassermitrisses und Vergleich der theoretischen Vorhersage mit experimentellen Ergebnissen - RS 179 Jahresbericht A 78 Development of a theoretical phase separation model with respect to liquid entrainment. Comparison of theoretical prediction with experimental data.		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Universität Hannover Institut f. Verfahrenstechn. Callinstr. 36 3000 Hannover 1
Arbeitsbeginn/Initiated Aug. 1975	Arbeitsende/Completed July 1978	Leiter des Vorhabens/Project Leader Prof. Dr.-Ing. F. Mayinger
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General aim

Experimental and theoretical investigations in the thermo- and hydrodynamic behaviour of a two phase pool with regard to nuclear safety studies.

2. Particular objectives

Experimental analysis and physical description of the fluidhydraulic conditions determining the phase distribution in a two phase pool, the phase separation at the horizontal surface and the droplet carry-over and entrainment in the vapour space.

3. Research programm

Steady state vapour injection into a liquid pool via different injector geometries, to analyse the vapour distribution in the mixture, the separation at the horizontal surface and the mechanism of droplet generation and carry over leading to entrainment in the vapour flow.

Quantitative tests to determine the void fraction and its vertical distribution within the mixture, depending on the superficial vapour velocity, on the initail height of the mixture (meaning) in absence of bubbling), on the system pressure and on different vesseldiameters.

Furthermore quantitav experiments, to gain data upon the diameter spectra of entrained droplets, information about the phase velocities and the mass of liquid entrained in the vapour flow are needed.

These values depend on the hydrodynamic conditions occuring in the phase pool and at its surface including the superficialvelocity of the vapour, the thermodynamic state and the location under investigation

i. e. the distance to the mixture surface.

In a second step the phase separation behaviour during pressure transients is regarded, to investigate the above mentioned subjects for flash evaporation at moving surface.

4. Experimental facilities, Computer codes

All experiments involved in this project are carried out by the use of the refrigerant R 12 (CF_2Cl_2) as a modelling fluid instead of water substance. To assure reactor relevant hydraulic conditions the fields of assumed main influencing parameters on this subject were scaled based on equal density ratios for water and freon.

The different types of test sections for the steady state as well as for the blowdown experiments were constructed as cylindrical glass vessels, allowing an optical and cinematographic investigation of the phase interaction and fluid behaviour.

For the vapour injection tests different internals could be built in to allow modified vapour distributions. To study the influence of locally high velocity differences on the phase separation and resulting droplet carry over, up to seven individually controllable nozzles were used.

A nearly homogenous vapour distribution which is more similar to evaporating pools could be simulated by the use of a porous sinter plate injection.

For the blowdown tests the liquid pool was heated up to saturation-temperature by use of electrical heaters.
Each vessel was instrumented with a number of thermocouples, pressure gauges, flowmeters or a weight tranducer (to measure the mass discharge during transients). The line void fraction in the mixture was picked up at different heights by a traversable γ-ray attenuation arragement, while the volumetric liquid fraction in the vapour space, the number, diameters and velocities of droplets were taken photograhycally.
For a mathematical treatment of the experimental data some computer codes were developed especially to investigate the transient fluid behaviour. Basing on a comparison of heat conduction and heat convection in the boundary layer around a growing bubble the onset of flashing in a depressurized pool could be calculated applying an implicit method.
To get some information about the vapour escape from the moving surface of

a flashing pool, the evapuation rate and the increase of the mixture surface were computed from a measured pressure decrease.
The used model consists of a theory for critical out-flow a theory for the vapour distribution in the saturated mixture and a theory discribing the motion of the evaporation front within the superheated liquid.

5. Progress to date

Within the steady state investigations, two semiempirical models discribing on the one hand the void fraction in the two phase mixture and on the other hand the stable (meaning the continously upwards moving) entrainment could be Verified by use of own R 12 and literature given H_2O data.
For the transient investigations, a flashing criterion and the time dependent onset of flashing could be calculated. Theoretical results are in fair agreement with available experimental data.
The separated rate of vapour from the moving surface of a flashing pool was computed by comparing experimental and theoretical results for the time depending mixture level swell.

6. Results

Optical investigations of former flashing tests have shown, that a liquid pool starts to evaporate in a nearly homogeneous manner throughout the whole vessel cross section if a depressurization is initiated. After a certain delay time the evaporation originated at the surface moves downwards within the liquid while a two phase mixture level rises upwards like a piston. To investigate the interaction of both phase and resulting hydraulic phenomena a comparable situation was simulated for steady state conditions, injecting vapour into a liquid pool via a porous sinter plate.
From the physical understanding the phase separation
and the liquid carry over as well as the entrainment behaviour in the vapour space are mainly dependant on the void fraction in the mixture and on the phase velocities.
Fig. 1 illustrates the relationship between the superficial vapour velocity and the resulting mixture averaged void fraction, which can be expressed by a power law function.
A change in the system pressure results in a parallel shifting of the above relationship.
Applying the theory of similarity to discribe a bubbling process, several dimensionless groups which appear to be significant have been regarded. As proposed by Wilson /1/ the volumetric void fraction was approximated by

a power law relation and expressed as a function of three groups,
- the Froude number, the bubble to vessel diameter and a density ratio -

$$\varepsilon_{theor.} = 0{,}573 \cdot \left[\frac{u_0^{*2}}{g\cdot\sqrt{\frac{\sigma}{g(\rho'-\rho'')}}}\right]^{0{,}309} \cdot \left[\frac{\sqrt{\frac{\sigma}{g(\rho'-\rho'')}}}{d_{vessel}}\right]^{0{,}085} \cdot \left[\frac{\rho''}{\rho'-\rho''}\right]^{0{,}15} \qquad (1)$$

In evaluating the coefficient C_1 and the exponents, own freon data and literature given H_2O data of Behringer /2/ and Wilson /1/ were used. The values thus determined were $C_1 = 0{,}572$, $a = 0{,}309$, $b = 0{,}055$ and $c = 0{,}150$. Equation (1) represents the data in Fig. 2 with an accuracy of $\pm$ 20 %.
To investigate the entrained liquid fraction ultra short duration photos with a magnification factor of 6 : 1 were taken at different distances to the mixture interface. From counting and measuring the recorded droplets size and frequency spectra could be found out. These histograms always gave two favorised diameters, one in the range of 25 - 35 microns and the other at about 100 to 200 microns. Sporadicly drops up to a diameter of 1 milimeter could be detected but these large ones only appear at close distances to the mixture surface. As already reported in RS 179 (1977) the small droplets are generated from the disintegration of bursting bubbles, while the large ones resulted from the liquid jet mechanism. It is remarkable that the bubble generated droplets became predominant with increasing vapour flowrates.
This is obviously, if one imagines an entranced turbulence of the mixture surface, so that the jet mechanism, resulting form bubble craters in the surface - is no longer able to be established.
With regard to the limited region wherein refalling drops could be detected, it appeared to be convenient to go foreward in investigating the behaviour of the small droplets being continously entrained in the vapour stream.

With the help of the flooding equation

$$(u_g - u_{drop}) = \left[\frac{4}{3}\,\frac{d_{drop}}{C_D}\cdot g\left(\frac{\rho'-\rho''}{\rho''}\right)\right]^{1/2} \qquad (2)$$

and a relation for the drag coefficient

$$C_D = (0{,}63 + 4{,}8/\sqrt{Re})^2 \tag{3}$$

the maximum diameter of a drop able to be entrained in a flow of a given velocity could be calculated iteratively from equation (4)

$$d_{max} = \left[\frac{\eta'' \cdot (u_g - u_{drop}) \cdot 3}{(\rho' - \rho'')\quad 4} \left\langle \left(\frac{(u_g - u_{drop}) \cdot d_{max} \cdot \rho''}{\eta''} \right)^{1/2} \cdot 0{,}63 + 4{,}8 \right\rangle^2 \right] \tag{4}$$

introducing a drop velocity to be zero.
The total droplet spectrum obtained at an observation altitude could now be diveded into to parts, applying the maximum diameter criterion. From the range of diameters smaller than the flooding limit an representative mean diameter was calculated. For that value the terminal slip ratio of a continously accellerated drop in the vapour stream was computed.
From the known slip ratio and volumetric liquid fraction the entrainment defined as

$$E = \frac{\dot{M}_{droplet}}{\dot{M}_{vapour}} = \frac{1 - \dot{x}}{\dot{x}} \tag{5}$$

could be determined.
Fig. 3 gives this so defined stable entrainment as a function of a modified Froude number, where the mixture averaged void fraction was taken into account.

$$Fr = \frac{u_0''^2}{\bar{\varepsilon} \cdot g \cdot H_v} \tag{6}$$

= superficial vapour velocity
= vertical distance of the optical axis to the mixture surface
= gravitational acceleration

With increasing system pressure the upstreaming vapour is able to carry over an increased liquid fraction.
To approximate the entrainment rate with respect to different system pressures, this influence was taken into account by some dimensionless groups, thus

$$E = C_2 \cdot \left[\frac{u_0''^2}{g \cdot H_v \cdot \bar{\varepsilon}}\right]^n \cdot \left[\frac{g\left(\frac{\sigma}{g(\rho'-\rho'')}\right)^{3/2} \cdot \rho'^2}{\eta'^2} \cdot \frac{(\rho'-\rho'')}{\rho''}\right]^m \cdot \left[\frac{\eta' \, u_0''}{\sigma}\right]^o \cdot \left[\frac{\sqrt{\frac{\sigma}{g(\rho'-\rho'')}}}{H_v}\right]^p \cdot \left[\frac{u_0''}{u_{drop}}\right]^q \qquad (7)$$

The constante C_2 and the exponents were correlated to

$C_2 = 2{,}98 \cdot 10^9$, $n = 0{,}515$, $m = {}^-0{,}631$, $o = 1{,}642$, $p = 0{,}504$
and $q = 0{,}328$
Equation (7) represents the experimental data with an accuracy of 10 %.

Transient investigations

Trying to make use of the steady state results on phase separation and liquid carry over, blowdown tests of an initually saturated liquid pool have been performed. As already reported in RS 179 (977) the vapour generation always started at the mixture surface after a certain delay time, so that the liquid became strongly superheated in the first period. For the calculation of the separating vapour rate from the flashing mixture an energy balance is used. For this procedure it is necessary to know the maximum superheating of the liquid and the time when flashing occurs.

Basing on the differential equation of the temperature field around a spherical bubble

$$\frac{\partial \vartheta}{\partial t} = \frac{\lambda}{\rho' \cdot c} \cdot \left(r^2 \frac{\partial \vartheta}{\partial r}\right) \qquad (8)$$

and the equation of bubble growth due to heat conduction

$$\frac{\partial r}{\partial t} = \frac{\lambda}{\rho'' \cdot h_{lg}} \cdot \frac{\partial \vartheta}{\partial x} \qquad (9)$$

the temperature distribution in the boundary layer was computed with respect to a motion of the bubble wall.

With the help of the Galilei transformation

$$r = x + R_0 + \dot{R} \cdot t \qquad (10)$$

equation (8) and (9) lead to

$$\frac{\partial \vartheta}{\partial t'} - \dot{R} \cdot \frac{\partial \vartheta}{\partial x} = a \cdot \frac{1}{r^2} \cdot \frac{\partial}{\partial x}\left(r^2 \frac{\partial \vartheta}{\partial x}\right) \qquad (11)$$

consisting of the convective term $\dot{R} \cdot \frac{\partial \vartheta}{\partial x}$ and the conduction term $a \cdot \frac{1}{r^2} \cdot \frac{\partial}{\partial x}\left(r^2 \frac{\partial \vartheta}{\partial x}\right)$

Fig. 4.1 shows the time depending temperature profiles within the boundary layer, whereby the ordinate always represents the bubble wall. Fig. 4.2 shows these profiles in comparison to the growth rate of the bubble, indicated by the vertical lines.
Besinde the measured pressure (index 3), the according saturation temperature (5), the bubble radius (1) the growth rate (2) and the temperature gradient at the bubble wall (4) the integrated area between the temperature profile in the boundary layer and the line of zero degree centigrade is inicated (dotted line) in fig. 4.3. The minimum indicates the onset of flashing.
The point is described by the first moment when the heat conduction velocity becomes less than the bubble growth velocity. This configuration can bee seen in fig. 4.1 by the first crossing of a later profile with an earlier one.
The values thus determined for the beginning of flashing are in fair agreement with the measurd pressure decrease curves, which shows at this time a beginning reincrease.
Knowing the time delay until flashing occurs, the miximum superheating of the liquid can be calcualted from the pressure decrease and the initial values. The following evaporation rate can be determined from an energy balance, if the liquid volume is known, which is cooled down to saturation and from which the destored superheating is used to evaporate a certain part of liquid.
Optical investigations have shown, that the evaporation always started at the mixture interface and that the flashing front moved downwards within the liquid. Basing on this knowledge the actual evaporated volume

was calculated interatively by comparing the generated volume with out streaming volume as long as the calculated system pressure becomes equivalent to the measurd one.
Fig. 5 shows an example for a suchwise computed escape rate of vapour from the flashing mixture.
Fig. 6 gives a comparison of the total evaporation rate to the remaining vapour in the mixture. The good agreement of the calculated value and the measured value for the remaining vapour fraction seems to prove the validity of the computational procedure.
A more detailed and complete description of the results can be found in the final report of RS 179 to be available in spring of 1979.

7. Next steps

No further investigations are planned.

8. Relation to other projects

RS 163

9. References

/1/ Wilson, J. F. Steam Volume Fraction in a bubbling two phase mixture
Trans. Am. Nuc. Soc. 4 No. 2 (Noc. 1961)

/2/ Behringer, P. Steiggeschwindigkeit von Dampfblasen in Kesselrohren
V.D.I. Forschungsheft 365, 1934

/3/ RS 179 (1977) Annual report

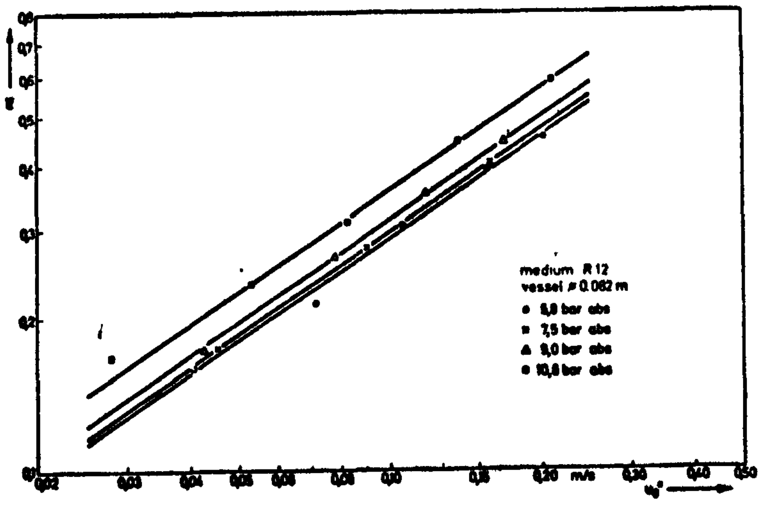

Fig. 1 : Averaged void fraction vs. superficial vapour velocity

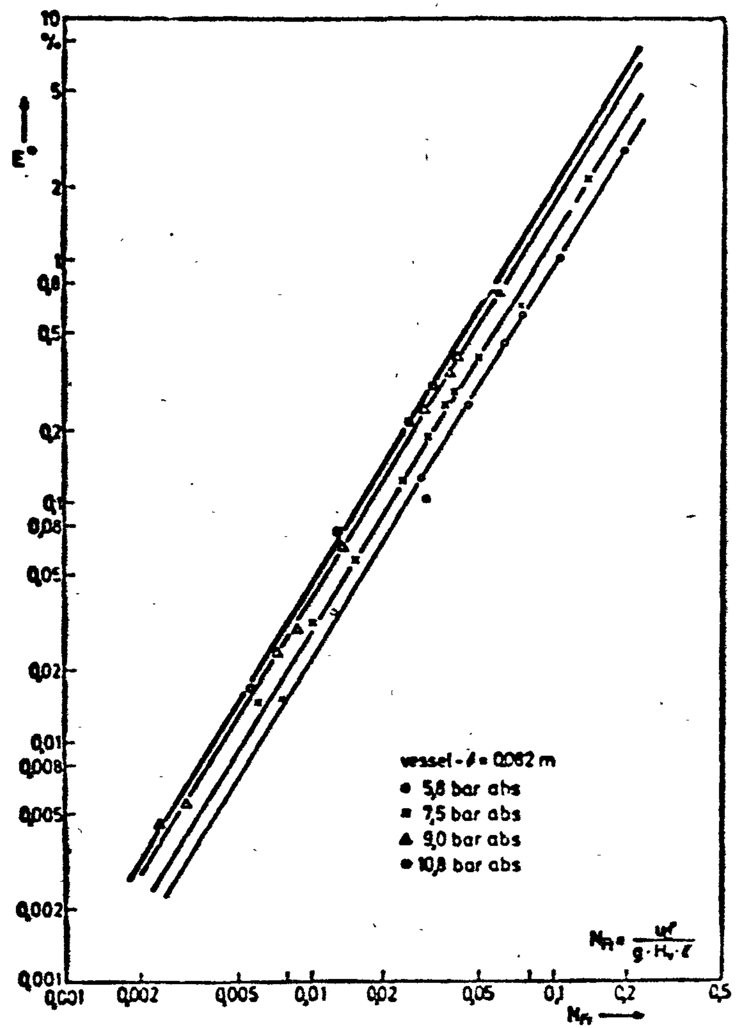

Fig 3 : Stable entrainment E* versus a modified Froude number

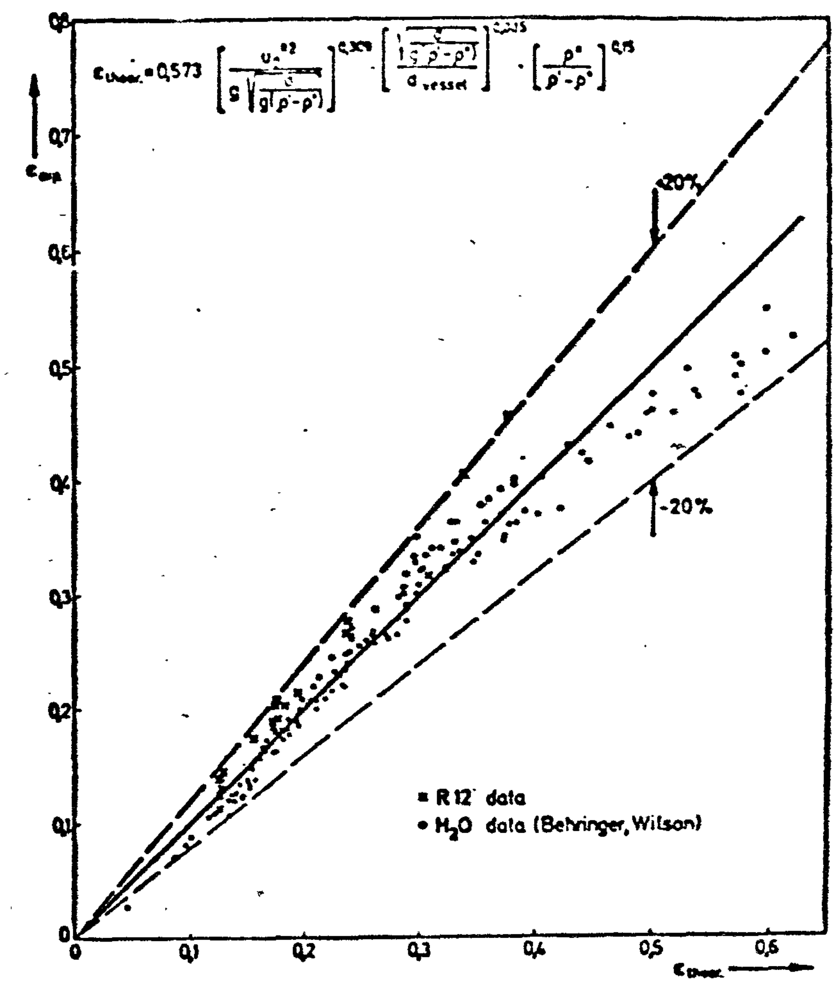

Fig. 2 : Deviation of experimental data from the theoretical approximation

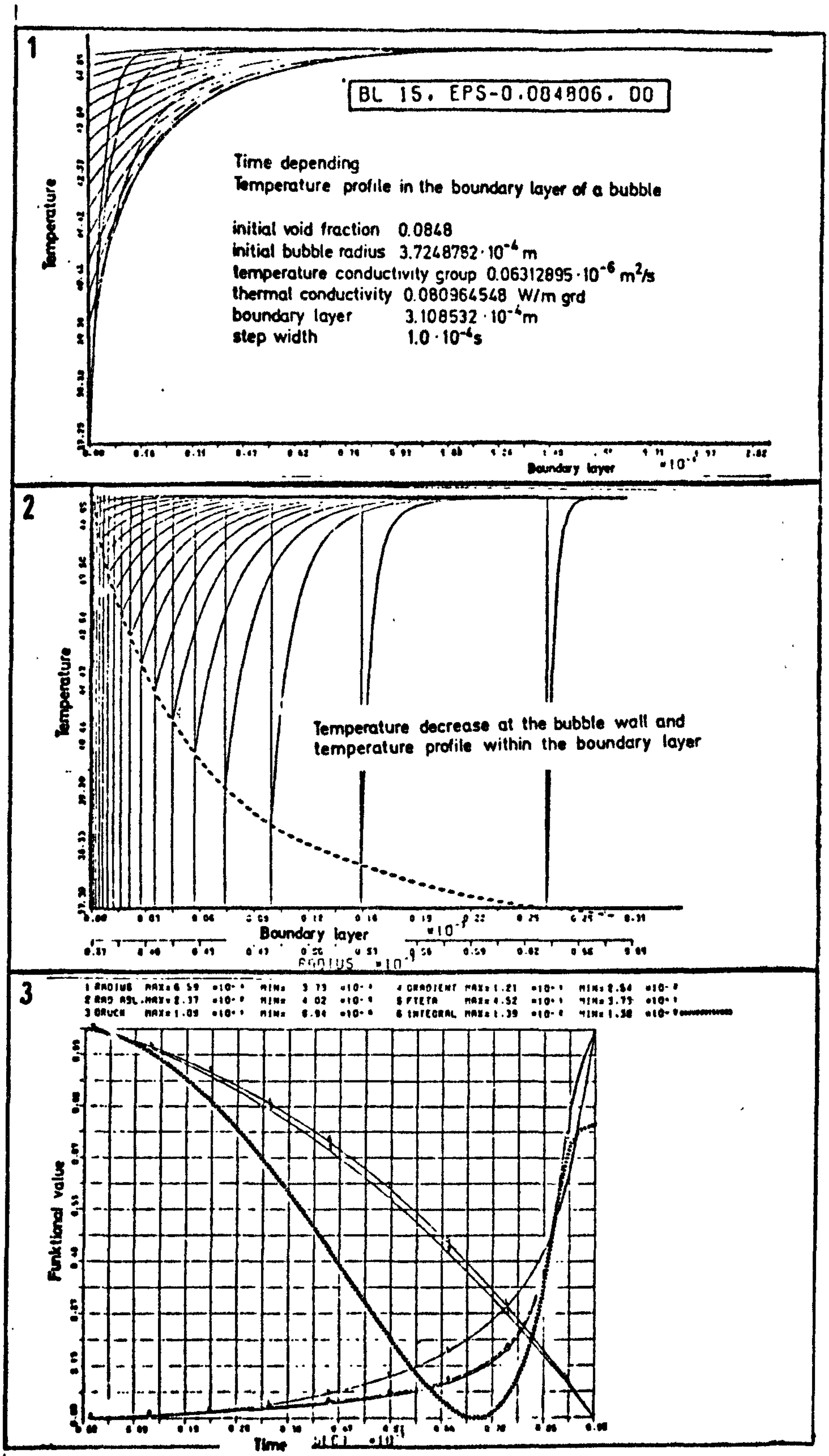

Fig. 4.1,4.2,4.3 Calculated onset of flashing in a superheated pool

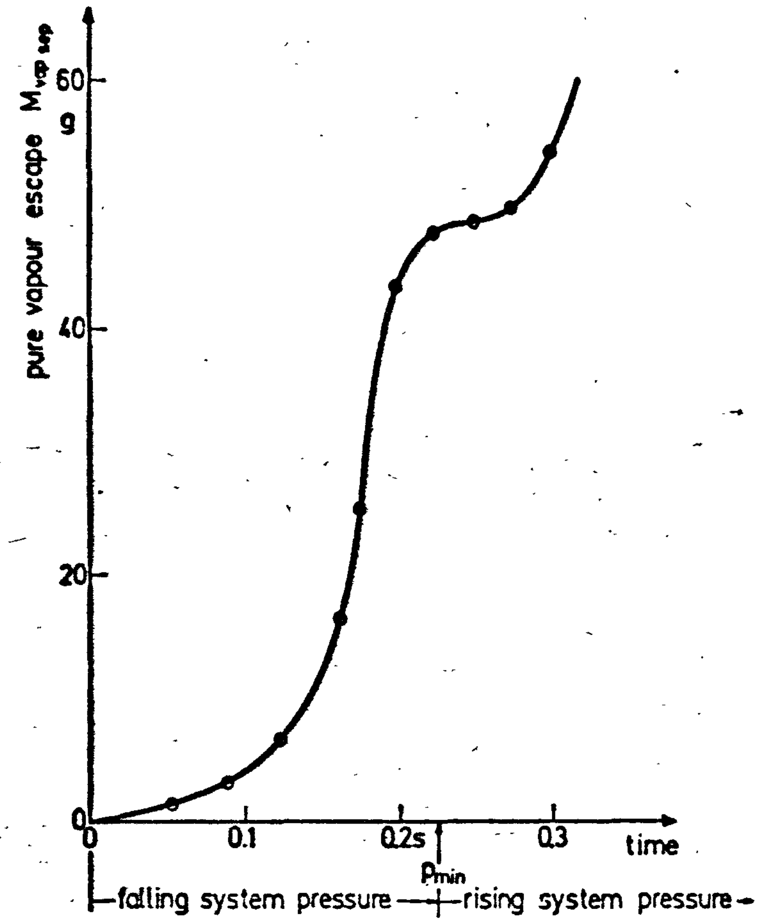

Fig. 6 : Pure vapour escape from a rising pool during depressurization (z_0 = 320 mm, ϕ-break = 20 mm, p_0 = 7.45 bar_{abs}, p_K = 0.2 bar_{abs})

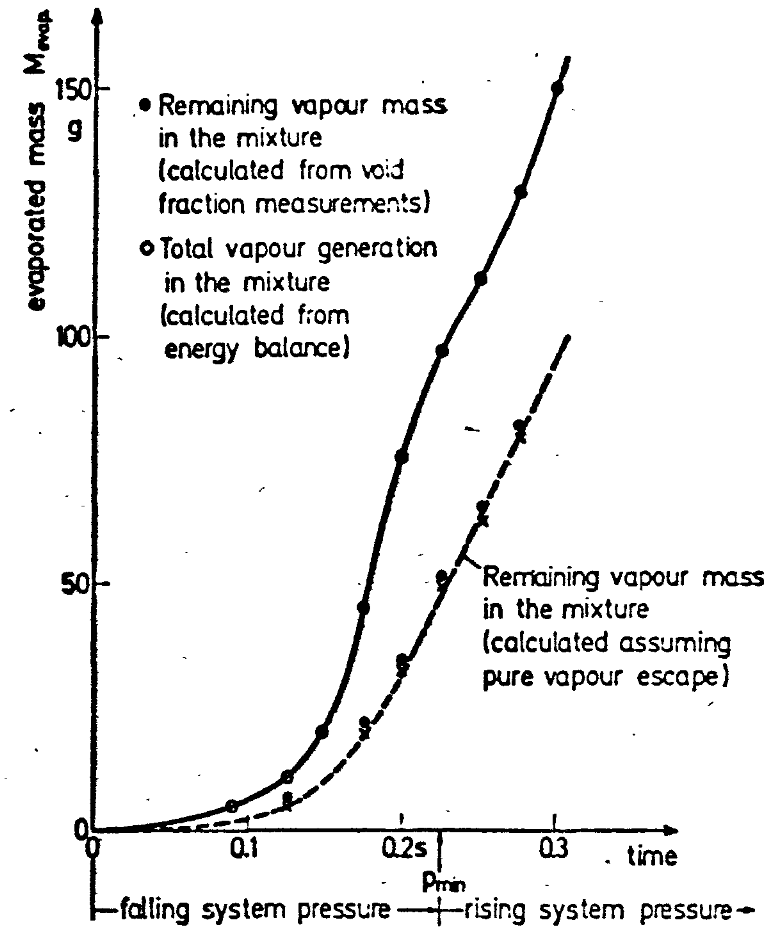

Fig. 5 : Comparison of total evaporated mass and remaining vapour in the mixture during level swell of a pool blowdown

145-1-06		1-1-2
TITRE Méthodes de mesure en double phase		Pays FRANCE Organisme Directeur CEA/DgCS EDF/SEPTEN
TITLE (Anglais) Development of two phase flow instrumentation		Organisme exécuteur CEA/DTCE-STT (GRENOBLE) Responsable
Date de démarrage 1974	Etat actuel en cours	Scientifiques
Date d'achèvement 31/12/79	Dernière mise à jour 01/79	

1. OBJECTIF GENERAL

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

2. OBJECTIFS PARTICULIERS

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

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

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

CANON : Test en régime transitoire.

4. ETAT DE L'ETUDE

Avancement à ce jour

Appareils de mesure par neutronographie étalonnés.
Moulinet et venturi étalonnés en régime permanent.
Développement technologique en cours des paliers de moulinet.
Utilisation d'un moulinet (Ø=20mm) avec roulement à billes effectuée avec succès.
Moulinet Ø 80mm et Ø 70mm à roulement à billes, qui seront utilisés respectivement sur OMEGA (grappe) et PHEBUS, en cours d'essais. ./.

Moulinet et venturi étalonnés en régime transitoire sur DEDIF.

5 - PROCHAINES ETAPES

Venturi à l'étude sur CANON.

6 - RELATION AVEC D'AUTRES ETUDES

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

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

7 - DOCUMENTS DE REFERENCE

- "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"
  JC. ROUSSEAU, J. CZERNY, B. RIEGEL
  Communication au Transient Two Phase Flow Meeting
  TORONTO (3-4 août 1976)

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

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

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

- "Expérience CANON : Dépressurisation d'une capacité en double phase : eau-vapeur"
  B. RIEGEL — Note TT 547, Avril 77

- "Contribution à l'étude de la décompression d'une capacité.
  B. RIEGEL
  Thèse de Docteur Ingénieur de l'Université de GRENOBLE, Juin 78

145-1-11		1-1-2
TITRE MODELISATION DE POMPES PRIMAIRES P.W.R. EN DOUBLE PHASE		**Pays** FRANCE **Organisme Directeur** CEA/DgCS
TITLE (Anglais) Modeling of P.W.R. primary pumps (two phase flow)		**Organisme exécuteur** C.E.A. DRE/STRE **Responsable**
Date de démarrage 1976	**Etat actuel** En Cours	**Scientifiques**
Date d'achèvement 12/1980	**Dernière mise à jour** 01/1979	

1. OBJECTIF GENERAL

Etude du comportement des pompes primaires pendant l'accident de perte de caloporteur dans les réacteurs à eau pressurisée. Développement d'un module de pompe en double phase dans le cadre du système POSEIDON.

2. OBJECTIFS PARTICULIERS

2.1 - Contribution au développement du modèle de l'EDF - sous-programme de CLYSTERE - par l'interprétation des essais EVA.

2.2 - Développement d'un modèle simplifié.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

Néant.

4. ETAT D'AVANCEMENT

Objectif 2.1 - Après l'établissement d'un jeu de données géométriques caractérisant un trajet fluide "moyen" dans la pompe de type 93A des essais EVA, le code a été recalé en simple phase sur les caractéristiques expérimentales - hauteur manométrique et couple hydraulique -

Dans le premier quadrant le recalage pour la pompe 93A n'est pas très satisfaisant. De plus, le recalage du 2ème quadrant (débit inverse, rotation directe) se heurte encore à des difficultés fondamentales.

Des comparaisons calcul-expérience ont été faites, en écoulement diphasique :

- calculs au débit volumique nominal et pour des taux de vide variant de 0 à 100% avec la dernière version du code transmise par l'EDF au STRE,
- études de sensibilité.

Il apparaît nettement, aussi bien pour la hauteur que pour le couple, que le code ne rend pas compte de :

- l'effet de compressibilité dans les faibles taux de vide,
- la dégradation importante dans les forts taux de vide.

Des études de l'EPRI ainsi que de récentes études de sensibilité réalisées par Westinghouse pour l'établissement de son modèle ont mis en évidence l'importance fondamentale de la séparation des phases et du glissement dans la pompe pour l'explication de la dégradation des caractéristiques. Le code de l'EDF utilise, rappelons-le, un modèle d'écoulement homogène sans glissement.

Objectif 2.2 - Compte-tenu des difficultés décrites ci-dessus, la décision a été prise d'engager une étude sur une modélisation simplifiée.

5. PROCHAINES ETAPES

1 - Formulation du modèle simplifié

Ce modèle sera ponctuel ou multiponctuel. Par contre l'intérêt sera focalisé sur les aspects essentiels : séparation des phases, glissement ...

Début de cette tâche : janvier 1979.

2 - Qualification de ce modèle sur des essais : EVA, EPOPEE ...

6. RELATIONS AVEC D'AUTRES ETUDES

Cette étude rentre dans le cadre général de l'étude n° 145-1-02 : "développement de modèles et codes de calcul de 2ème génération pour l'étude de l'accident de perte de caloporteur dans les réacteurs à eau préssurisée : programme général coordonné".

145-1-12		1-1-2
TITRE DEVELOPPEMENT DU CODE OMNIBUS POUR L'ETUDE DES PETITES BRECHES DANS LE CIRCUIT PRIMAIRE D'UN REACTEUR A EAU PRESSURISEE ET LES ATWS		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) OMNIBUS code development for small breaks and ATWS studies		Organisme exécuteur C.E.A. DRE/STRE Responsable
Date de démarrage 1978	Etat actuel En Cours	Scientifiques
Date d'achèvement 12/1980	Dernière mise à jour 01/1979	

1. OBJECTIF GENERAL

Développement d'un code pour calculer l'écoulement monodimensionnel vertical d'un fluide diphasique avec apparition de niveau libre.

2. OBJECTIFS PARTICULIERS

2.1 - Mise au point d'un module pour petites brèches dans le cadre du système POSEIDON.

2.2 - Utilisation du code, en liaison avec le code SIRENE, pour l'analyse des ATWS.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

Néant.

4. ETAT D'AVANCEMENT

Un module de calcul pour un canal vertical avec conditions aux limites imposées a été réalisé. Il comprend :

- un modèle d'écoulement à quatre équations, avec déséquilibre thermodynamique de la phase liquide et écart de vitesse entre les phases liquide et vapeur,
- une méthode de résolution numérique implicite-explicite,
- deux corrélations à ajuster sur des expériences : l'une pour l'écart de vitesse, l'autre pour l'entraînement des gouttelettes d'eau en aval de la surface libre.

Une série de calculs en régime permanent, sans entraînement de gouttelettes, a permis de faire une première vérification du modèle (référence 7.1).
Le régime transitoire sera testé prochainement sur l'installation expérimentale TAPIOCA.

5. PROCHAINES ETAPES

- Adaptation du module au système POSEIDON.
- Etude de certains ATWS, en liaison avec le code SIRENE.

6. RELATIONS AVEC D'AUTRES ETUDES

. L'objectif 2.1 rentre dans le cadre général de l'étude n° 145-1-02 : "développement de modèles et codes de calcul de 2ème génération pour l'étude de l'accident de perte de caloporteur dans les réacteurs à eau pressurisée : programme général coordonné".

. L'objectif 2.2 rentre dans le cadre de l'étude n° 146-1-01 : "développement de moyens de calcul nécessaires à l'étude des transitoires anormaux sur les réacteurs PWR".

7. DOCUMENTS DE REFERENCE

7.1 - "Etudes de dépressurisation par petites brèches - Code de calcul OMNIBUS"
I. SZABO, D. MENESSIER, D. ABRAMSON, M. FAJEAU
Rapport DRE/STRE/LET 78/005 - Présentation au séminaire franco-soviétique (Moscou : du 28.01 au 07.02.1978).

7.2 - "MINIBUS - Module de calcul d'un écoulement eau vapeur dans un canal vertical avec apparition éventuelle d'une surface libre"
I. SZABO, M. MALAMAS
Rapport commun CEA - DRE/STRE/LET 78/126
CISI- LOG/MED/n° 135

145-1-03		1-1-2
TITRE Thermohydraulique du LOCA. Etude des débits critiques en double phasé : Programmes MOBY-DICK et SUPER MOBY-DICK.		Pays FRANCE Organisme Directeur CEA/DgCS - EDF/SEPTEN
TITLE (Anglais) LOCA thermohydraulics. Critical two phase flow studies : MOBY-DICK and SUPER MOBY-DICK projects.		Organisme exécuteur CEA/DTCE/STT (GRENOBLE) Responsable
Date de démarrage 01/01/72	Etat actuel en cours	Scientifiques
Date d'achèvement 31/12/80	Dernière mise à jour 01/79	

1. OBJECTIF GENERAL

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

2 - OBJECTIFS PARTICULIERS

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

3 - INSTALLATIONS EXPERIMENTALES ET PROGRAMME

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

../..

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

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

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

4 - ETAT DE L'ETUDE

Avancement à ce jour

MOBY-DICK : La campagne d'essais eau-air a été terminée en janvier 78. Rapport final en cours d'édition.
SUPER MOBY-DICK :
Un retard dans la livraison de la pompe ne permettra les premiers essais qu'en mars 79.

5 - PROCHAINES ETAPES

SUPER MOBY-DICK : Les essais eau-vapeur pour 50 <P <100 bars sur un canal de 20 mm avec divergent se dérouleront jusqu'en décembre 79. Ensuite des essais sur un canal de 20mm avec élargissement brusque sont prévus.

6 - RELATION AVEC D'AUTRES ETUDES

MARVIKEN C.F.T. : Expérience suédoise de débit critique pour des sections de brèches importantes (200 à 500 mm). (voir fiche 145-1-10)

R E B E C A : Etude de débits critiques d'un mélange à 3 composants : eau, air, vapeur. (voir fiche 145-1-09)

7 - DOCUMENTS DE REFERENCE

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

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

145-1-04		1-1-2
TITRE Thermohydraulique du LOCA. Etude de la phase dépressurisation d'un réacteur à eau pressurisée : Programme OMEGA.		Pays FRANCE Organisme Directeur CEA/DgCS EDF/SEPTEN
TITLE (Anglais) LOCA thermohydraulics. P.W.R. Blowdown studies : OMEGA Project .		Organisme exécuteur CEA/DTCE-STT (GRENOBLE) Responsable
Date de démarrage 01/01/72	Etat actuel en cours	Scientifiques
Date d'achèvement 31/12/82	Dernière mise à jour 1/79	

1. OBJECTIF GENERAL

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

2. OBJECTIFS PARTICULIERS

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

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

- Boucle OMEGA : Pression 170 bars, débit max. 20 kg/s, puissance 4,5 MW.

- Dépressurisation d'une section d'essais tubulaire puis d'une grappe 36 barreaux :
  1) Section tubulaire de 12 mm de diamètre et 3m65 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 combustible type 17 x 17 de 3m65 de long avec un flux cosinus ; taille de brèche : 4 cm2 (amont, et aux deux extrémités).

- C A N O N : 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.
- D E D I F : Développement d'instrumentation pour écoulement diphasique.

4 - ETAT DE L'ETUDE

1) Avancement à ce jour

Campagne d'essais de dépressurisation OMEGA sur section tubulaire terminée. Dépressurisation sur CANON 30 bars terminée.
Campagne préliminaire d'essais 36 barreaux de 3,65 m de longueur, flux cosinus, terminée.
Campagne super Canon 155 bars terminée.
Deuxième campagne d'essai de dépressurisation sur section tubulaire avec mesures améliorées et changement de volume des capacités.

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.

Détermination des coefficients d'échange à partir du calcul inverse de conduction.

5 - PROCHAINES ETAPES

2ème campagne d'essais 36 barreaux (2ème semestre 79)

Dépouillement des essais avec le modèle à 6 équations

6 - DOCUMENTS DE REFERENCE

- " 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 DTC-STT 491.
- "Dépressurisation d'un sous-ensemble du circuit OMEGA comprenant une section d'essais chauffante tubulaire "
  R. RICQUE et al - Note DTCE-STT 146 (Sept. 77).
- "OMEGA : Dépressurisations adiabatiques. Analyse par le code RELAP 4"
  M. MEZZA - Note SETSSR 77/193.
- Dépouillement de la première campagne d'essais de "décompression tube" sur la boucle OMEGA.
  par M. BONNETON - Note DTCE-STT n° 580 (août 78)
- Dépressurisation d'un sous-ensemble du circuit OMEGA comprenant une grappe chauffante à 36 barreaux
  . Description du dispositif expérimental

./.

. Présentation des résultats expérimentaux
par R. FRANK, R. RICQUE, R. BOURGINE, G. COULON, J. LAMBERT
rapport DTCE-STT n° 152 (Sept. 78)

- Contribution à l'étude de la décompression d'une capacité.
B. RIEGEL - Thèse de Docteur Ing. de l'Université de Grenoble - Juin 1978

- Super-CANON Experiments
2nd Specialist on transient two phase flow OCDE Meeting Paris June 1978
J.C. ROUSSEAU

145-1 - 10		1.1.2
Titre MARVIKEN CFT (Etude de débits critiques)		Pays SUEDE
		Organisme directeur PAYS NORDIQUES USA (N.R.C. + EPRI) FRANCE (CEA+ EDF) ~~PAYS BAS (KEMA)~~
Titre (anglais) MARVIKEN CRITICAL FLOW TESTS		Organisme exécuteur MARVIKEN - SUEDE
		Responsables français J. PELCE (CEA/DSN) J. AZAM (EDF/SEPTEN)
Date de démarrage 1/01/77	Etat actuel en cours	Scientifiques français M. REOCREUX (CEA/DSN) G. HOUDAYER (EDF/SEPTEN)
Date prévue d'achèvement 1/07/79	Dernière mise à jour 1/78	

1 - Objectif général :

1) Etude des débits critiques dans le cas des réacteurs à eau ordinaire.

2 - Objectifs particuliers :

2) La particularité des essais MARVIKEN réside dans l'étude des ruptures de grosses sections, ce qui n'est jamais le cas dans les installations expérimentales. Les sections de fuite étudiées seront : 200, 300 et 500 mm de diamètre.

3 - Installations expérimentales et programme :

On utilise pour l'expérience, le réacteur désaffecté de MARVIKEN (Suède). La cuve du réacteur est remplie sous pression. La brèche est située au bas de la cuve. Les paramètres étudiés sont :

.../

- pression à la brèche (30,40 et 50 bars)
- diamètre de la brèche (20,30 et 50 cm)
- sous-saturation de l'eau (5,15 et 30°C)
- rapport L/D (L longueur de la section d'essai, D diamètre de la brèche (0, 1, 2,5 et 3)

Le programme a été défini lors d'une réunion d'experts qui'd'est tenue à MARVIKEN en avril 1976. La durée des essais est de 18 mois.

L'accord concernant les essais a été signé par :

Les Pays Nordiques (Suède, Danemark, Norvège, Finlande)

Les USA (NRC + EPRI)

La France (CEA + EDF)

Les Pays Bas (KEMA)

La participation française est de 20% du coût total répartis également entre CEA et EDF.

4 - Etat de l'étude :

La phase de modification de l'installation et de mise au point est terminée et la période des essais proprement dits a débuté (durée prévue : jusqu'au 1/7/79)

6 - Relation avec d'autres études :

Essais de débits critiques réalisés en FRANCE : MOBY DICK, SUPER MOBY DICK.
Méthodes analytiques d'interprétation.
N.B. : l'interprétation des essais ne fait pas partie de l'accord.

	CLASSIFICATION: 1.1.2
TITLE (ORIGINAL LANGUAGE): Influenza di ostruzioni e di piegamento di barre in un fascio a flusso termico disuniforme di tipo BWR	COUNTRY: ITALY SPONSOR: C.N.E.N.
TITLE (ENGLISH LANGUAGE): Influence on DNB of Subchannel Obstructions and Rod Bowings in BWR Bundles with Non-Uniform Heat Flux Profiles	ORGANISATION: C.N.E.N. PROJECT LEADER: G. PALAZZI

INITIATED: April 1976	COMPLETED:	SCIENTISTS:
STATUS: In progress	LAST UPDATING: July 1979	

1) General aim — Experimental study for determination of effect of fuel bundle distortions in faulty conditions.

2) Particular objectives — The influence of single subchannel obstructions and rod bowings in a 4 rods (freon cooled) bundle will be tested.

3) Experimental facility — The experiment has been carried out on Casaccia CF2 loop (power 260 kW; mass flow-rate 10 t/h)

4) Project status — After 106 runs without obstructions, a complete series of burn-out tests have been performed with different kinds of blockages.The blocked area varies from 40% to 80%. Besides rod bowing of the bundle has been tested. More than 200 runs are avaible.

5) Next steps — Analysis of the DNB phenomena through the main parameters.

6) Degree of availability — Free
Contact person:
G. Palazzi, CNEN, CSN Casaccia, CP 2400, I-00100 ROMA

	CLASSIFICATION: 1.1.2
TITLE (ORIGINAL LANGUAGE): STUDIO DELL'EFFLUSSO CRITICO BIFASE IN CONNESSIONE CON IL LOCA NEI REATTORI AD ACQUA LEGGERA	COUNTRY: ITALY SPONSOR: UNIVERSITA' PALERMO[x]
TITLE (ENGLISH LANGUAGE): STUDIES ON TWO-PHASE CRITICAL FLOW IN CONNECTION WITH LOCA IN LIGHT WATER REACTORS	ORGANISATION: UNIVERSITA' PALERMO PROJECT LEADER: ELIO OLIVERI
INITIATED: 1976 / COMPLETED:	SCIENTISTS: F. CASTIGLIA, G. VELLA
STATUS: IN PROGRESS / LAST UPDATING: July 1979	

x 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 a particular test conditions - a theoretical model for the prediction of steam / water critical pressure and critical flow rate in terms of upstream stagnation properties.

A theoretical model, which allows to evaluate both critical flow rate and critical conditions in ducts with friction and adiabatic walls, has been proposed (Ref. 3). This model was based on the assumption of a purely annular motion, without accounting for the entrainment phenomenon . Futher studies will be therefore run in order to extend the theoretical model to the above-mentioned phenomenon.

REFERENCE DOCUMENTS :

1- F. CASTIGLIA - E. OLIVERI - G. VELLA
Sulla determinazione della portata nell'efflusso critico bifase. ACCADEMIA DI SCIENZE LETTERE E ARTI DI PALERMO- serie IV- Vol. XXXV- 1975-76- parte I .

2- F. CASTIGLIA - E. OLIVERI - G. VELLA
Un programma in BASIC per il calcolo delle proprieta' termodinamiche dell'acqua. ACCADEMIA DI SCIENZE LETTERE E ARTI DI PALERMO Febbraio 1979.

3- F. CASTIGLIA - E. OLIVERI - G. VELLA
Sull' efflusso critico di miscele bifasi monocomponenti. ENERGIA NUCLEARE - Vol. 26 - n. 4 - Aprile 1979.

(Istituto Applicazioni ed Impianti Nucleari, Università, Parco d'Orleans, I-90128 Palermo)

45.897

		CLASSIFICATION: 1.1.2
TITLE (ORIGINAL LANGUAGE): Esperienze di scambio termico in condizioni stazionarie di post-crisi in geometria BWR 4x4		COUNTRY: Italy SPONSOR: NUCLITAL
TITLE (ENGLISH LANGUAGE): Steady-state post-dryout experiments in 4x4 BWR geometry		ORGANISATION: NUCLITAL/CISE PROJECT LEADER: G.P. Gaspari (NUC)
INITIATED: January 1978	COMPLETED:	SCIENTISTS: A. Azzalin (CISE) E. Giammari (NUC) C. Medich (CISE)
STATUS: In progress	LAST UPDATING: June 1979	

General Aim

Verification of the applicability of the available post-dryout correlations to complex geometries.

Particular Objectives

In addition to the general aim at the previous point, these experiments are in order to overhaul the experimental techniques and procedures to be used in future programs in very complex test sections (8x8).

Experimental Facilities and Programme

The experiments have been carried out on the CISE IETI-4 loop.
The following conditions have been investigated:
- pressure: 50-70 bar
- specific mass flowrate: 250-1000 kg/m^2s
- inlet quality: + 5 ÷ 60%
- wall temperature: up to 700 °C.

Project Status

The experiments have been completed in May 1978. Preliminary analysis of data has been performed.

Next Steps

The analysis completion has been delayed up to 1980 spring (see below).

Relation with Other Projects

When the post-dryout data planned in tubular geometry will be available (end of 1979) present data will be reviewed, in order to determine the relation between simple and complex situations.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION :
Steady-state post-dryout experiments in 4x4 BWR geometry	1.1.2

Reference Documents

A. Azzalin, C. Medich, O. Vescovi, "Trasmissione del calore in condizione di parete asciutta nel fascio BWR 4x4 con distribuzione assiale di flusso disuniforme", CISE NT-78-064.

Degree of Availability

Proprietary.

G.P. Gaspari, Nuclital, V. G. D'Annunzio 113, I-16121 Genova.

	CLASSIFICATION: 1.1.2
TITLE (ORIGINAL LANGUAGE): Esperienze di scambio termico in condizioni di post-crisi in geometria tubolare.	COUNTRY: Italy
	SPONSOR: NUCLITAL
TITLE (ENGLISH LANGUAGE): Post-dryout experiments in tubular geometry	ORGANISATION: NUCLITAL/CISE
	PROJECT LEADER: G.P. Gaspari (NUC)

INITIATED:	COMPLETED:	SCIENTISTS:
January 1979		E. Giammari (NUC) R. Granzini (CISE) R. Martini (CISE) G. Barzoni (CISE)
STATUS: In progress	LAST UPDATING: June 1979	

General Aim

Verification and development of post-dryout heat transfer correlations applicable to transient analysis.

Particular Objectives

Obtaining experimental data mainly in steady-state conditions (but also transient experiments are planned) using a particular tubular test section axially subdivided in two parts indipendently heated. This in order to obtain conditions (Ø, T, X) not obtainable in usual steady-state experiments, and representative of the behaviour during LOCA transients.

Experimental Facilities and Programme

The experiments will be carried out on the CISE IETI-1 loop.
The following test conditions will be investigated:
- pressure: 30-70 bar
- specific mass flowrate: 125-1000 kg/m^2s
- outlet quality: up to superheated steam
- wall temperature: up to 800 °C.

Project Status

The test section, which is particularly instrumented, is completed and tests are under commissioning.

Next Steps

The experiments are planned in the 1979 second half. The analysis completion is scheduled in December 1979.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Post-dryout experiments in tubular geometry	1.1.2

Relation with Other Projects

The data will be compared with those obtained in 4x4 geometry, in order to determine the relation between simple and complex situations.

Reference Documents

G. Barzoni, R. Martini, "Procedure sperimentali e di elaborazione dati prove ultracrisi NUCLITAL", Promemoria DTN/79/SAT/007.

Degree of Availability

When available the data will be probably published.

G.P. Gaspari, Nuclital, V. G. D'Annunzio 113, I-16121 Genova.

	CLASSIFICATION: 1.1.1,1.1.2,1.1.4,1.2,7.
TITLE (ORIGINAL LANGUAGE): Analisi dei transitori termici ed idraulici a seguito di LOCA nei reattori ad acqua leggera e nei sistemi di contenimento associati.	COUNTRY: ITALY SPONSOR: CNEN and CNR
TITLE (ENGLISH LANGUAGE): Analysis of thermal and hydraulic transients following a LOCA in L.W. reactors and in associated containment systems.	ORGANISATION: University of Pisa PROJECT LEADER: N. CERULLO
INITIATED: 1974 / COMPLETED: 1980 STATUS: in progress / LAST UPDATING: May 1979	SCIENTISTS: F. ORIOLO, G. MEI, G. CECCARELLI, G. BITETTI

<table>
<tr><td colspan="2"></td><td>CLASSIFICATION:
1.1.1, 1.1.2</td></tr>
<tr><td colspan="2">TITLE (ORIGINAL LANGUAGE):
Programma P.I.Pe.R. - Esperienze di blow-down da un recipiente in pressione dotato di strutture interne.</td><td>COUNTRY:
ITALY

SPONSOR:
CNEN and CNR</td></tr>
<tr><td colspan="2">TITLE (ENGLISH LANGUAGE):
Blow-down experiments by PIPER apparatus with internal structures.</td><td>ORGANISATION:
University of Pisa

PROJECT LEADER:
Piero VIGNI</td></tr>
<tr><td>INITIATED: (second phase)
1975</td><td>COMPLETED: (second phase)
1980</td><td rowspan="2">SCIENTISTS:
Umberto ROSA
Francesco D'AURIA</td></tr>
<tr><td>STATUS:
in progress</td><td>LAST UPDATING:
May 1979</td></tr>
</table>

ENERGIEONDERZOEK CENTRUM NEDERLAND		CLASSIFICATION : 1.1.2
TITLE : CHARME: Een computerprogramma ter bestudering van uitstroming		**COUNTRY:** THE NETHERLANDS
		SPONSOR : ECN
		ORGANIZATION : ECN
TITLE (ENGLISH LANGUAGE): CHARME: A computer program to study blowdown		**PROJECTLEADER :** Speelman, J.E.
INITIATED : April 1976	**LAST UPDATING** : July 1979	**SCIENTISTS :** Putten, A.P.W.M. van der; Bogaard, J.P.A. van den; Koning, H.
STATUS : progressing	**COMPLETED** : 1980	

General aim

Development of a computer code to study the blowdown-process.

Particular objectives

CHARME solves a set of partial differential equations describing the conservation of mass, energy and momentum in a tube as a finction of axial coordinate and time, using the method of characteristics. Notably the pressure, temperature, velocity and void fraction are calculated as function of time and of axial coordinate including the transition to supersonic velocities. The model also includes pressure losses due to friction. A special subroutine has been developed to calculate the fluid parameters in the jet arising after the system opening. This jet model is necessary for the cases in which the blowdown process proceeds with subcritical velocities.

Experimental facilities and program: Not foreseen

Project status

A first model has been developed for the jet; CHARME calculations have been verified on basis of experimental results, performed by others, with encouraging agreement.
The code has been updated, especially the subroutine in which the fluid properties are calculated.

Next steps

Improvement of the jet subroutine. A sensitivety study will be performed with regard to slip and non-equilibrium effects between two phases.

Relation with other projects

Usable to check gross models in other thermo-hydraulic blowdown computer codes.

Reference documents

J.P.A. v.d. Bogaard, H. Koning, A.P.W.M. v.d. Putten: CHARME, A time and space dependent model to predict the discharge rate of single and two-phase fluids through pipes.
NEA-CSNI: Specialist's meeting on transient two-phase flow, Toronto, 3rd-4th August, 1976.

Degree of availability: Upon mutual agreement at ECN-Petten

Budget: -

Personnel:

ENERGIEONDERZOEK CENTRUM NEDERLAND (ECN)		CLASSIFICATION: 1.1.2
TITLE: Reflood experimenten		COUNTRY: THE NETHERLANDS
		SPONSOR: ECN ORGANIZATION: ECN
TITLE (ENGLISH LANGUAGE): Reflood experiments		PROJECTLEADER: S.B. van der Molen
INITIATED : 1977	LAST UPDATING : May 1978	SCIENTISTS: S.B. van der Molen H. Hoogland F.W.B.M. Galjee
STATUS : started	COMPLETED : 1980	

General aim

Study of the reflood and rewetting phenomena in bundles.

Particular objectives

a) Investigation of the heat transfer from high temperature fuel pins in bundles by radiation to waterdroplets and vapour convection before rewetting.
b) Study of the influence of a temperature profile over a bundle cross-section on the velocity of the quench front and the cross flow between adjacent channels.
c) Study of the influence of instabilities in parallel bundles on the velocity of the rewetting-front.

Experimental facilities and program

a) Testloops for low and high pressure experiments
b) High speed film camera

Project status

Experiments to study the rewetting phenomena by high speed cinematography have been started in order to study the two phase flow downstream of the quench front. First calculations have been made to obtain an order of magnitude of the convective and radiative heattransfer to the dispersed two phase flow.

Next steps

Development of measuring techniques to determine void fraction and droplet concentration in the two phase flow, downstream of the quench front.

Relation to other projects: no

Reference documents: ECN-78-064, The entrained droplet and vapour velocity and the heat transfer in the dispersed flow region in case of bottom flooding.

Degree of availability: Through E.C.N. library channel

Budget: Hfl. 150.000,--/yr

Personnel: 3 manyears/yr.

Netherlands Energy Research Foundation (ECN)		CLASSIFICATION : $\boxed{1.1.1}$/1.1.2
TITLE : Mechanisch gedrag van het reactorbinnenwerk tijdens grote ongelukssituaties		COUNTRY: THE NETHERLANDS SPONSOR : ECN ORGANIZATION : ECN
TITLE (ENGLISH LANGUAGE): Mechanical behaviour of reactor internals during major accident situations.		PROJECTLEADER : L.H. Vons
INITIATED : 1977	LAST UPDATING :May 1977	SCIENTISTS : H. van Rij L.G.J. Janssen
STATUS : in progress	COMPLETED : 1980	

	CLASSIFICATION: 1.1.2, 1.1.3, 1.1.4, 1.1.2
TITLE (ORIGINAL LANGUAGE): LOCA UNCERTAINTY MARGINS (GSD4.2)	COUNTRY: UK
	SPONSOR: UKAEA
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/UKAEA
	PROJECT LEADER: RNH McMILLAN
INITIATED: DECEMBER 1978 / COMPLETED:	SCIENTISTS/ENGINEER: AA DEBENHAM
STATUS: / LAST UPDATING: MAY 1979	

BACKGROUND

Currently the US NRC specifies that LOCA analyses must be carried out using 'evaluation models' with supposedly pessimistic data and assumptions (Ref 1). However there is considerable national and international activity in the development of 'best estimate' analytical methods. Introduction of such methods requires the estimation of uncertainty margins so that adequate conservatism can be demonstrated in the application of the results. Additionally with the increasing tendency towards frequency/consequence forms of safety justification, allowances for calculational and data uncertainties must be quantified for accident sequences other than the design basis accident.

A programme of work is proposed to estimate uncertainty margins to be applied to LOCA analyses based on the following:-

(a) differences between integral experimental data and predictions,

(b) sensitivity studies using RELAP or other calculational methods,

(c) consideration of residual uncertainties.

OBJECTIVES

1. Review all available experimental data on LOCA integral experiments and comparisons with predictions. Identify modelling improvements which have lead to significant improvements in predictive capability, and modelling aspects, which require further improvement.

2. Set up a RELAP 4 Mod 6 model of the LOFT nuclear blowdown facility. Carry out sensitivity studies to try to identify sources of incertainty in predictions.

3. Make initial review of uncertainty margins applicable to commercial PWR LOCA analysis.

FACILITIES

Analytical codes
RELAP TRAC FRAP

INTERNATIONAL COLLABORATION

USNRC Water Reactor Safety Research
ISPRA

REFERENCE DOCUMENT

1. 10 CFR 50

	CLASSIFICATION: 1.1.2, 1.2
TITLE (ORIGINAL LANGUAGE): WATER COOLED REACTOR DEPRESSURIZATION STUDIES:- CSNI STANDARD PROBLEM CALCULATIONS FOR ECCS.	COUNTRY: UK
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/UKAEA
	PROJECT LEADER: G M JORDAN
INITIATED: 1974 / COMPLETED:	SCIENTISTS: A A DEBENHAM
STATUS: Continuing / LAST UPDATING: MAY 1979	

1. GENERAL AIM

To improve the understanding of the sequence of events following a loss of coolant accident on a water cooled reactor, and to study the effectiveness of emergency core cooling systems.

2. PARTICULAR OBJECTIVES

To perform computer calculations of agreed test problems and to discuss computer code shortcomings at special meetings organized under the auspices of the CSNI.

3. PROJECT STATUS

To date six problem calculations have been carried out. Two more are currently being examined.

4. FUTURE WORK

CSNI meetings identify outstanding problem areas in the LOCA/ECCS field and try to select suitable experimental work for comparisons. Pre-predictions of a LOFT nuclear test has been suggested as a future problem.

TITLE Water Cooled Reactor Loss of Coolant Accident Studies				CLASSIFICATION 1.1.2
COUNTRY	SPONSOR UKAEA		ORGANISATION SRD	PROJECT LEADER A. R. EDWARDS
INITIATED		COMPLETED		SCIENTISTS
STATUS Continuing		LAST UPDATING May 1978		

1. General Aim

To study the physical limitations in thermal-hydraulic phenomena associated with the loss of coolant accident in a water reactor.

2. Particular Objective

To study the basis for heat transfer and fluid flow correlations used in computer codes which purport to calculate the sequence of events following a loss of circuit integrity of a water cooled reactor.

3. Experimental Facilities and Programme

The study makes use of information generated by facilities all over the world.

4. Programme Status

A continuing commitment, to enable SRD to judge the validity of computer code calculations presented for LOCA studies.

Classification

1.1.2

Title 1	COUNTRY UNITED KINGDOM
DEPRESSURISATION DISCHARGE RATE	SPONSOR UKAEA
	ORGANIZATION AWRE FOULNESS
Title 2	Project Leader
	A R EDWARDS
Initiated 1968 Completed :	Scientists:
Status : Last updating 1976	

Description:

1. General Aim

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

2. Particular Objectives

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

3. Experimental Facilities

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

4. Project Status

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

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

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

Data reduction of a representative sample of tests is in progress and reports should be available in 1977.

Reference Documents

Aldermaston Report AWRE/44/86/97 (SRD R29)

	CLASSIFICATION: 1.1.2
TITLE (ORIGINAL LANGUAGE): Post dryout heat transfer at low quality and low pressure (HTH 1.2.1)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Harwell)
	PROJECT LEADER: J Fell (Winfrith) J B Sayers (Harwell)
INITIATED: March 1978 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: March 1979	

Background

During the past two years, an experimental rig and special test sections have been built at Harwell to measure post-dryout heat transfer to steam-water mixtures at low quality and low pressure. (1)

Objectives/Programme

1. During the year 1978/79, it is proposed to use the rig to obtain post-dryout data over the following range of conditions:

Mass velocity:	20-2000 Kg/m^2s
Pressure:	2-4 bars
Thermodynamic Quality :	-0.1 to +0.3
Surface Temperature:	300-600°C

Upflow and downflow.

2. With these data, an attempt will be made to produce a new correlation for post-dryout heat transfer in tubes at low qualities. This correlation should be compatible with existing correlations which describe the heat transfer at high qualities.

3. Experiments will also be performed to examine rewetting phenomena during transient cooling tests and an attempt will be made to obtain transition boiling data.

Facilities

4. An existing rig is in use at Harwell; additional test sections will be added for programme item 3.

Reference Documents

1. Newbold, Ralph and Ward AERE R 8390 (1977).

	CLASSIFICATION: 1.1.2
TITLE (ORIGINAL LANGUAGE): Theoretical modelling of two-phase flow in complex geometries (HTH 2.1.1)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Harwell)
	PROJECT LEADER: J Fell (Winfrith) J B Sayers (Harwell)
INITIATED: March 1978 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: March 1979	

Background

A mathematical model of annular two-phase flow has been developed in which the processes of entrainment of liquid droplets from the liquid film into the turbulent gas flow of the core and their subsequent redeposition at the central features. This model has been used to calculate dryout in a heated flow in a number of situations.

1. Steady state flow in a round tube[1]

2. Transient flow in a round tube[2]

3. Steady state flow in a rod bundle[3,4]

Objectives

1. To investigate the relevance of previous work on steady state dryout in rod bundles to the PWR core geometries.

2. To verify the extension of the analysis and numerical methods for transient flow to cover pressure transients in a round tube.

3. To extend the work on flow, power and pressure transients in a round tube to PWR core geometries.

4. To extend work from PWR core geometries to other complex cases such as the PWR lower plenum.

Programme

Objectives 1 and 2 should be completed in 1979 and items 3 and 4 by mid 1981.

HTH 2.1.1 contd

Facilities

No experimental facilities will be used.

Reference Documents

1. Whalley P B, Hutchinson P and Hewitt G F (1974). AERE-R7520

2. James P W and Whalley P B (1978). AERE-R8980

3. Whalley P B (1876). AERE-R8319

4. Whalley P B (1978). AERE-R8977

Heat transfer during blowdown	1.1.2	Thermal Hydraulics

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

1,2. General Aims and Particular Objectives

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

3. Experimental facilities

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

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

4. Project Status

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

It is anticipated that detailed experimental work will commence shortly.

contd.

5. 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.

1.1.2.

	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 1.1.2
Title 1 Blowdown of a steam drum		Country U.K. Sponsor C.E.G.B. Organisation C.E.R.L.
Title 2		
Initiated 1976 Status Continuing	Completed Last updating Aug 1978	Project Leaders Dr P.R. Farmer Dr D.J. Woodford

1. General Aim

To examine the behaviour of the fluid in a steam drum separator during LOCA conditions using Freon 12 as a modelling fluid.

2. Particular Objectives

To determine the period during which liquid continues to flow down the downcomers. To determine the flashing behaviour in the drum. To compare data with existing models and formulate new ones.

3. Experimental Facilities and Programme

A drum slice 1.6 m dia. by 1.6 m long will be blown down from either top or bottom or both. Voidage, temperature and pressure measurements will be made in the vessel and pipework. Mass flow will be measured in the blowdown lines. The modelling fluid is Freon 12 at pressures up to 15 bar. Initial studies will examine a drum cleared of internals. Separation equipment will be installed for later work.

4. Project Status

The drum is in course of commissioning.

		Classification 1.1.2
Title 1 Four-Quadrant, Two-phase flow in pumps		Country U.K.
Title 2		Sponsor CEGB Organisation Marchwood Eng. Labs.
Initiated 1976	Completed	Project Leader
Status Continuing	Last updating 1978	Mr. E.M. Curtis

1. General Aim

To obtain the knowledge of pump flows required for the prediction of post LOCA conditions in water reactors.

2. Particular Objectives

To determine the performance characteristics of a typical reactor circulating pump in two-phase, four-quadrant operation as a function of voidage, flow regime, pressure ratio etc.

3. Experimental Facilities and Programme

Measurements will be made using scale model pumps (50 mm inlet and 125 mm inlet) in an air-water loop. Transparent pipes and pumps will permit flow visualisation.

4. Project status

Single phase experiments in progress.

CLASSIFICATION 1.1.3

	CLASSIFICATION: 1.1.2, 1.1.3, 1.1.4, 1.1.2
TITLE (ORIGINAL LANGUAGE): LOCA UNCERTAINTY MARGINS (GSD4.2)	COUNTRY: UK SPONSOR: UKAEA
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/UKAEA PROJECT LEADER: RNH McMILLAN
INITIATED: DECEMBER 1978 / COMPLETED:	SCIENTISTS/ENGINEER: AA DEBENHAM
STATUS: / LAST UPDATING: MAY 1979	

CLASSIFICATION 1.1.4

Berichtszeitraum/Period 1.1. - 31.12.1978	Klassifikation/Classification 1.1.4	Kennzeichen/Project Number 06.01.04 (PNS 4234)
Vorhaben/Project Title Bestimmung der Nachzerfallswärme der Spaltprodukte von ^{235}U im Zeitbereich 10 - 1000 sec Decay Heat Measurement of ^{235}U Fission Products in the Time Period 10 to 1000 Seconds		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe Projekt Nukleare Sicherheit /INR
Arbeitsbeginn/Initiated 1975	Arbeitsende/Completed 1979	Leiter des Vorhabens/Project Leader K. Baumung
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

LOCA Analysis

2. Particular Objectives

Providing decay heat data of ^{235}U for short cooling times up to 1000 seconds.

3. Research Program

Irradiation of pellet-like fuel samples and measurement of their adiabatic temperature rise and γ-energy output due to fission product decay.

4. Experimental Facilities, Computer Codes

A pneumatic transfer system providing a good cooling of the fuel samples during irradiation is connected to a computer-controlled adiabatic calorimeter. The energy loss due to γ-escape is recorded by a Moxon-Rae-type γ-energy-flux detector. Monte-Carlo-codes for the calculation of the burnup profiles, the γ-escape characteristics and heat source distributions of the fuel samples are provided for the transfer of the experimental results to the test-pin conditions of the blowdown-experiments.

5. Progress to Date

The experimental facility was installed at the FR2 reactor and test runs were performed. A contamination monitor was installed in order to reduce the release of radioactivity in the case of a clad-failure of the samples.

6. Results

The proper operation of the transport system was demonstrated and all security requirements of the reactor operator are met. The operating permission was granted. The regulation performance of the calorimeter-jacket was found to be $\pm 2 \cdot 10^{-3}$ K, the sensitivity of the calorimeter is 10^{-4} W, its response time constant about 2 seconds.

7. Next Steps

The next step will be the irradiation experiments, and in parallel, the calibration measurements required for the total fission number evaluation.

	CLASSIFICATION: 1.1.1,1.1.2,1.1.4,1.2,7.2
TITLE (ORIGINAL LANGUAGE): Analisi dei transitori termici ed idraulici a seguito di LOCA nei reattori ad acqua leggera e nei sistemi di contenimento associati.	COUNTRY: ITALY SPONSOR: CNEN and CNR
TITLE (ENGLISH LANGUAGE): Analysis of thermal and hydraulic transients following a LOCA in L.W. reactors and in associated containment systems.	ORGANISATION: University of Pisa PROJECT LEADER: N. CERULLO
INITIATED: 1974 | COMPLETED: 1980 STATUS: in progress | LAST UPDATING: May 1979	SCIENTISTS: F. ORIOLO, G. MEI, G. CECCARELLI, G. BITETTI

<table>
<tr><td colspan="2"></td><td>CLASSIFICATION:
$\boxed{1.1.2,}$ 1.1.3, 1.1.4, 1.1.2</td></tr>
<tr><td colspan="2" rowspan="2">TITLE (ORIGINAL LANGUAGE):
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<tr><td>SPONSOR:
UKAEA</td></tr>
<tr><td colspan="2" rowspan="2">TITLE (ENGLISH LANGUAGE):</td><td>ORGANISATION:
SRD/UKAEA</td></tr>
<tr><td>PROJECT LEADER:
RNH McMILLAN</td></tr>
<tr><td>INITIATED:
DECEMBER 1978</td><td>COMPLETED:</td><td rowspan="2">SCIENTISTS/ENGINEER
AA DEBENHAM</td></tr>
<tr><td>STATUS:</td><td>LAST UPDATING:
MAY 1979</td></tr>
</table>

CLASSIFICATION 1.2

CLASSIFICATION 1.2

TITLE 1 FLECHT. Low flooding rate test program (Full length Emergency Cooling Heat Transfer)	COUNTRY Belgium (U.S.A.) SPONSOR ORGANIZATION Westinghouse Nuclear Europe
TITLE 2	PROJECT LEADER SCIENTISTS
INITIATED (date) May 1974 — COMPLETED End 1976 STATUS PROGRESSING — LAST UPDATING May 9, 1975.	

1. GENERAL AIM

The general objective of the FLECHT test program is to obtain experimental data for use in evaluating the heat transfer capabilities of a PWR Emergency Core Cooling System during a postulated loss-of-coolant accident.

2. PARTICULAR OBJECTIVES

The objectives of the tests to be conducted in the modified FLECHT test configuration are to supplement the parametric effects studied in the original FLECHT program, and to provide heat transfer coefficient and entrainment data at flooding rates of 1 in/sec and below. The forced flooding tests will be conducted with rod bundles having a cosine and a skewed axial power profile.

3. EXPERIMENTAL FACILITY

The FLECHT-SET test facility will be modified to conduct forced flooding tests as shown in Figure 1.

The modified facility consists of :

a) The original FLECHT test section housing with baffle installed in the upper plenum exhaust to improve liquid carryover separation.

b) The 10 x 10 rod bundle and related existing instrumentation including the ANC liquid level transmitter.

c) The existing pressurized water supply accumulator and injection line with three rotameters injection rates from 0.5 to 12 in/sec.

d) A close coupled carryover tank connected to the test section upper plenum.

e) A commercially available steam separator with a capacity of 2500 lbs/hr, and a liquid collection tank to collect liquid entrained in the exhaust steam.

d) A close coupled carryover tank connected to the test section upper plenum.

e) A commercially available steam separator with a capacity of 2500 lbs/hr, and a liquid collection tank to collect liquid entrained in the exhaust steam.

f) Exhaust piping with a system pressure control valve and an orifice plate flow meter to mesure steam flow rate.

4. PROJECT STATUS

a) Progress to-date :

Modifications to the test facility have been completed, and shakedown testing has been started with the cosine axial power profile rod bundle.

b) Results : None

5. NEXT STEPS

Complete testing with a rod having a cosine axial power profile in April, 1975.

Complete testing with a rod bundle having a skewed axial power profile in November 1975.

6. RELATION WITH OTHER PROJECTS

This program is related to all other Emergency Core Cooling System Test Programs such as :

Delayed DNB
UHI

Blowdown, Refill and Reflood
FLECHT-SET

7. REFERENCE DOCUMENTS

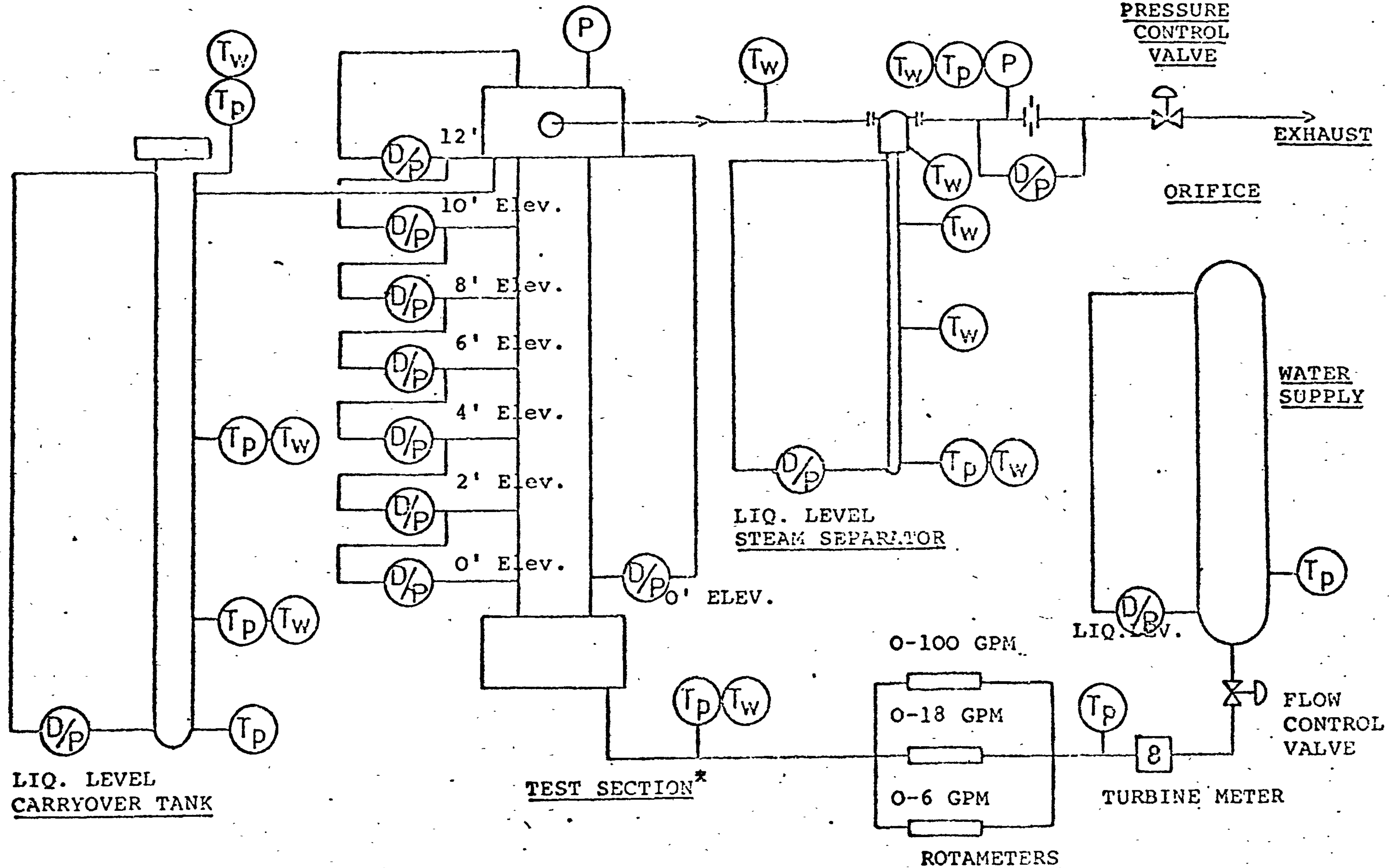
a) WCAP-7665 - PWR FLECHT Final Report, April, 1971.

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

8. DEGREE OF AVAILABILITY

Available upon request.

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



* ALL INSTRUMENTATION IS NOT SHOWN

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

FLECHT : FLECHT-SET

(Full Length Emergency Cooling Heat Transfer System Effects Tests)

1. GENERAL AIM

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

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

2. PARTICULAR OBJECTIVES

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

3. EXPERIMENTAL FACILITIES AND PROGRAM

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

The program is divided in 2 steps :

- PHASE A consisting of scoping tests (1 loop no steam generator)
- PHASE B including a more complete systems effect simulation (2 loop steam generator simulation).

4. PROJECT STATUS

4.1. Programs to-date

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

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

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

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

4.2. Essential Results

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

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

5. NEXT STEPS

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

6. RELATION WITH OTHER PROJECTS

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

7. REFERENCE DOCUMENTS

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

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

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

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

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

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

8. Degree of availability

Available upon request.

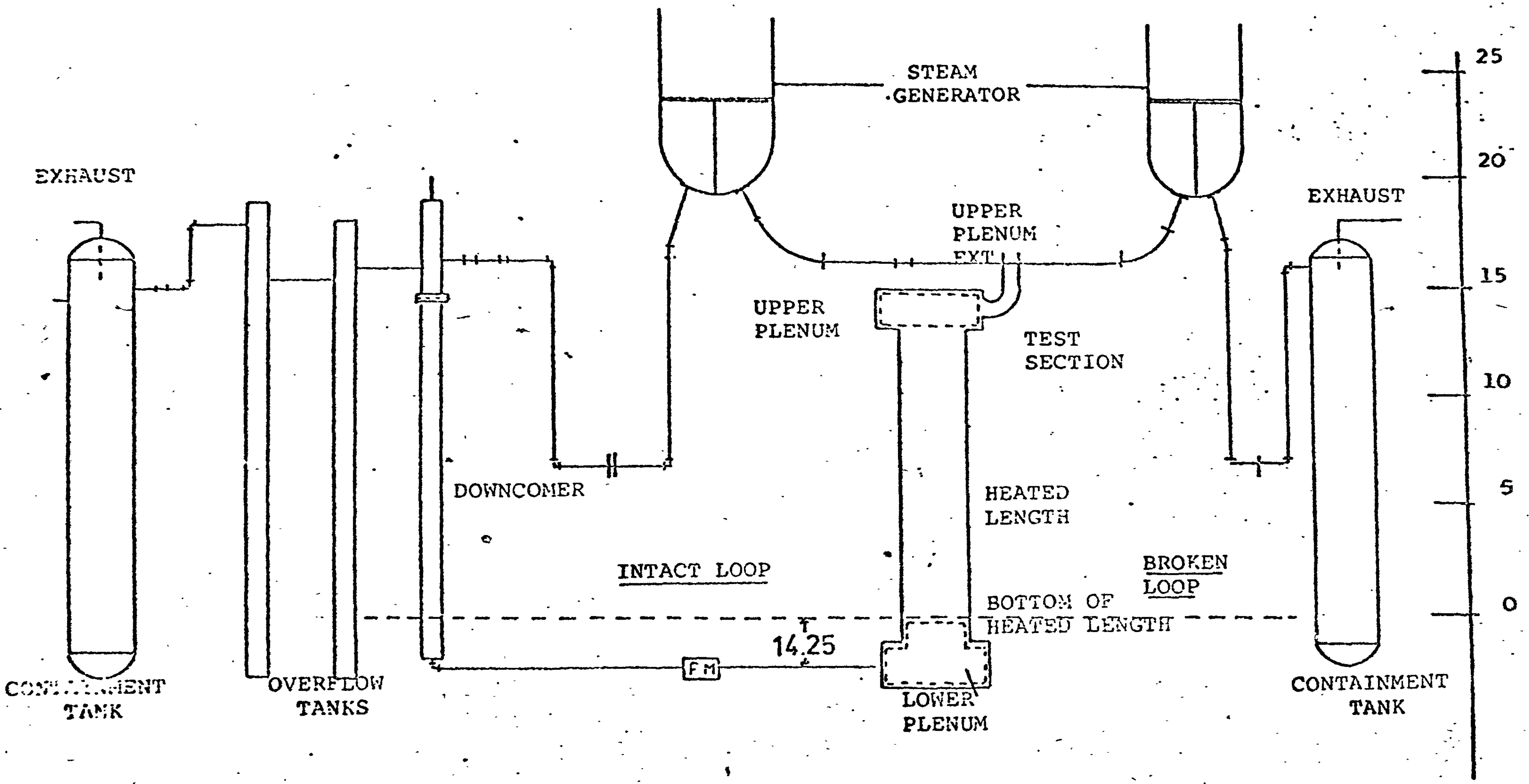


FIGURE 1

FLECHT-SET PHASE B

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

1. GENERAL AIM

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

2. PARTICULAR OBJECTIVES

The AEC interim criteria states in part :

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

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

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

3. EXPERIMENTAL FACILITIES AND PROGRAM

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

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

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

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

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

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

4. PROJECT STATUS

Progress to-date and essential results.

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

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

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

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

5. NEXT STEPS

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

6. RELATION WITH OTHER PROJECTS

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

TABLE 1

COLD LEG STEAM/WATER MIXING TESTS

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

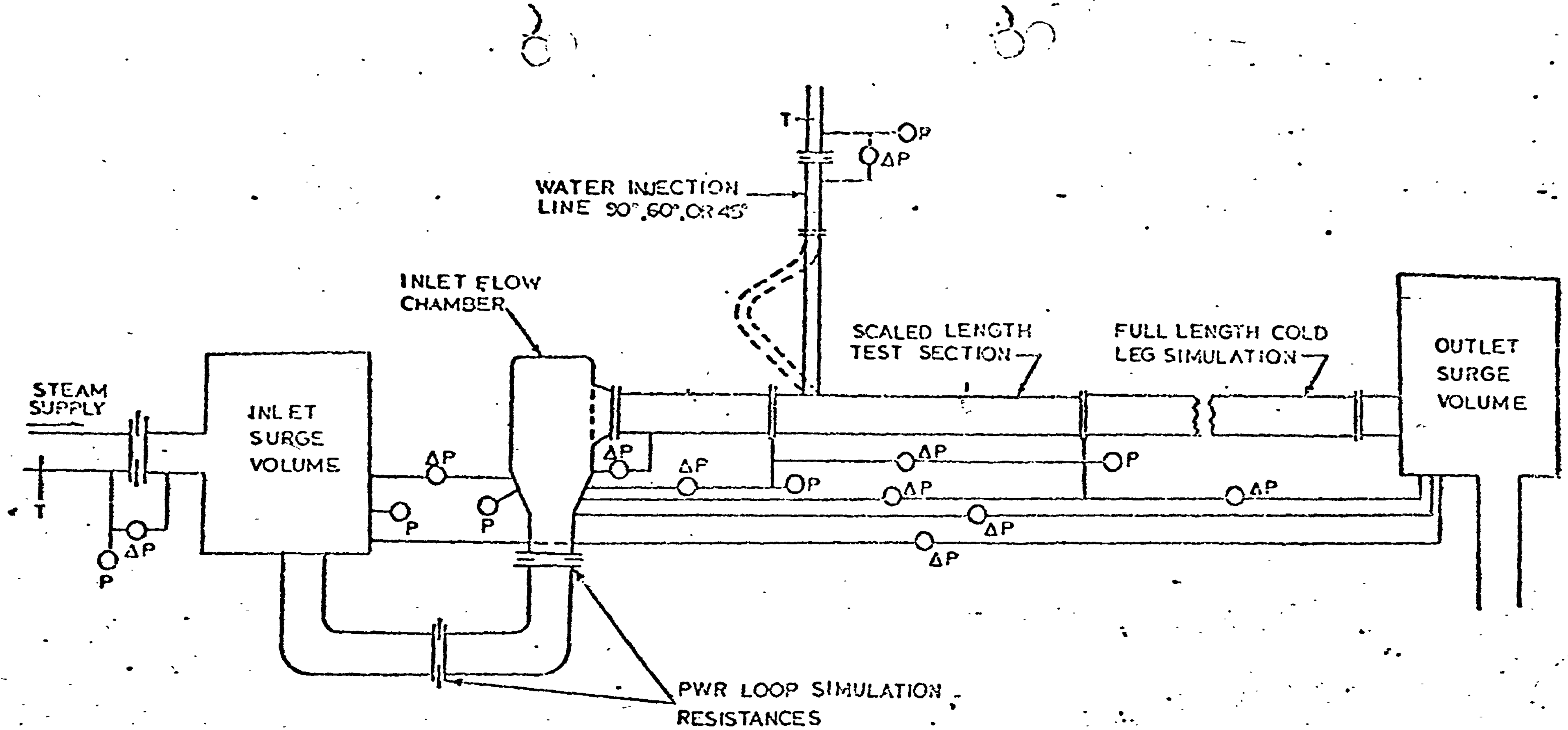


FIGURE 1
STEAM-WATER MIXING TEST CONFIGURATION
SHOWING PRESSURE AND FLOW INSTRUMENTATION

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	1.2	RS 0036 B

Vorhaben/Project Title	
Notkühlprogramm - Niederdruckversuche Wiederauffüllversuche mit Berücksichtigung der Primärkreisläufe Emergency Core Cooling Program - Low Pressure Experiments. Refilling Experiments with Simulation of the Circulation Loop	Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 513, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 1. 73	30. 6. 79	D. Hein
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 78	

1. General Aim

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

2. Particular Objectives

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

3. Research Program

The test facility is designed to cover the following parametric variations:

- Max. initial clad temperature: 500 to 800 °C
- System pressure: 1 - 6 bars
- Max. decay heat flux: 4 to 8 W/cm^2
- Time function of decay heat: const, ANS standard
- Reflood rates: 6 - 60 cm/sec
- Split of reflood rates top/bottom: 0/5, 1/4, 2/3, 3/2 4/1, 5/0
- Time function of injection rates: const, accumulator characteristic
- Break size: 0,25 to 2 F (double ended guillotine)
- Break location: hot leg, cold leg,
- Simulated pump resistance: locked rotor, free rotor
- Residual water in lower plenum: 0 to lower grid plate
- In loop seals: 0 to full

-2-

4. Experimental Facilities

In order to investigate the refill and reflood phase in a PWR including the feedback of the complete primary system, a test facility was built. Beside a 340 rod-testbundle it includes three scaled down primary loops with active steam generators.

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

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

5. Progress to Date

The position of the heating filament at two locations (0^{o} and 90^{o}) in 50 heater rods was measured by X-ray and recorded, and 25 were chosen for instrumenting with thermocouples.

The position of the spacer grid, the temperature measuring levels and the radial distribution of the instrumented heater rods, as well as the position of the control rod guide tube, were established in consultation with the GRS.

With the completion of the second test bundle, the following work was carried out:

- 160 heater rods were tested for leaks and , if necessary, rebrazed and redressed. Subsequently the ceramic caps were brazed on and welded
- The thermocouples were brazed in the gooves of the instrumented heater rods, the brazed locations dressed

and subsequently the thermocouples measured throughout and recorded.

- For the remainder of the delivered heater rods, the Helium leak test was carried out, and, if necessary, rebrazing was resorted too. Afterwards the end caps were brazed on, and the internal resistance measured and recorded.
- The spacer grids for the inner heated zone and the upper contact plate with the silver plate were completed.
- To permit the installation of the Film Probes and the Density Measurement Probes, the lead-ins to the conduit, core container and rod bundle container were constructed.
- The individual current leads for the heater rods were constructed. They enable the recognition of the failure of a single rod.
- The control rod guide tubes (unheated rods of 13.7 mm dia.) were completed. Six rods were prepared for the installation of the Steam-Probe, which will enable a wall temperature measurement at any given time on all 7 measurement levels.

Two rods with Core Liquid Level Detectors and 5 rods with core impedance probes were supplied by the NRC. Contract partner in the USA.

Details of further collaboration were discussed in a meeting with representatives from the USA. Although not anticipated by KWU, it appeared that the signals of the Liquid Level Sensors can be processed digitally on-line, rather than being first stored in an intermediate operation on analog tape, as in the LOFT Project. The data processing equipment necessary for it does not exist and as yet was not provided for in the preparations for Test Series II.

6. Results

Except for some failed thermocouples, the required 25 heater rods are instrumented. The 340 rods required for the bundle are completely installed. The spacer grids for the inner zone and the upper contact plate, including the silver plate, are completed. The NRC (Dr. Hsu) notified KWU that, for PKL II, only 5 (instead of 10) rods with Core Impedance Probes and only 4 (instead of 10) Film Probes have been completed by Oak Ridge. This makes necessary a new distribution of the 5 rods in the bundle.

7. Next Steps

The bundle assembly is being continued.

8. Relation with Other Projects

RS 0036 C, RS 287

9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	1.2	RS 287

Vorhaben/Project Title	Land/Country
Wiederauffüllversuche mit Berücksichtigung der Primärkreisläufe (PKL) Versuchsphasen I B und II Refilling Tests Considering the Primary Loops (PKL). Test Phases I B and II	FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 513, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 9. 77	31. 12. 80	D. Hein
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	31. 12. 78	

1. General Aim

Experimental investigation of the refill and reflood phase during a LOCA, using a sufficiently large model of the entire PWR primary loop.

2. Particular Objectives

Special aims of this program are

- to perform approx. 15 more test runs in the PKL test facility using the currently installed 340 rod bundle
- selection and upgrading of additional instrumentation, especially for the second bundle
- to replace the current bundle by the new one
- to perform approx. 30 more test runs with the new bundle.

Parallel and subsequent to the test runs, the results will be evaluated and presented in test reports.

3. Research Program

Performance of further refill and reflood tests.
Additional instrumentation provided for the second test bundle
Modifications and additions for the PKL test facility
Installation of the new bundle
Start-up
Performance of tests
Evaluation and documentation of test results.

4. Test Facility

The PKL test facility simulates as closely as possible the primary loops including active steam generators of a

1300 MW plant, scale 1 : 134 (referring to number of heater rods). The conceptual design of the test facility has the following specific features:

- exact simulation of the core geometry and heating of the bundle
- exact simulation of all reactor elevations, locations of feed nozzles leading into the primary loops and pressure loss sequences
- sufficiently good simulation of the circuit volumes and the thermal capacities of the primary- and the secondary sides.

The test results are to verify the computer codes used for emergency core cooling analysis. In this connection special attention should be paid to the results obtained not only from the German FLUT and WAK programs, but also from RELAP4 MOD6 and TRAC which are developed in the USA as well as the REFLOS program which was developed by GRS from the US code FLOOD4.

5. Progress to Date

The work completed during the reporting period is basically divided into two tasks: the preparation and completion of test series IB and the preparation of test series II.

Test series IB:

The plans for test series IB are to perform about 15 - 20 more tests with the first test bundle, following series IB (RS 36 B). On that occasion the following parameters are to be studied more thoroughly in tests with a rupture located in the cold leg:

- tests for the proper model simulation of hot leg injection in primary coolant loops by varying the internals in the upper plenum
- systematic variation of the distribution of flood water between hot-leg and cold-leg injection locations
- effects of a reduced containment backpressure of 1.2 bar (4.2 bar regularly, so far)
- reduction of rupture cross-section to 1/2 F (2 F regularly so far).

Should the bundle instrumentation still provide sufficient test data, then further tests with ruptures located in the hot leg are planned.

So far, the first-mentioned tests have been performed with varying internals and varying injection of the flood water in 7 test runs (5 of which with varying test conditions) (K14 series). The variation of internals was made for the following reasons:

- On account of the model simulation of the reactor in primary coolant loops (heights 1 : 1, but reduction of cross-sections at a scale factor of 1 : 134) the flood water injection is to be effected in such a way that both in the downcomer and in the upper plenum the processes to be expected for the reactor develop properly from a phenomenological point of view. Their influence on the refill and reflood process was examined by means of varying the internals and changing the water supply.
- As, apart from the TRAC computer program, only one-dimensional programs are available for verifying the tests, an attempt should be made (and better than in the IA test series) to achieve radially symmetric cooling conditions in the test bundle.
- By means of more instrumentation in the upper plenum and especially above the rod holding plate, the tests should furnish measuring data concerning events at the rod holding plate for the 2D/3D test project, which is yet to be planned, as the rod holding plate is the interface between the 2D (core segment) and 3D test projects (upper plenum).

For exclusive cold-leg injection a test was carried out in order to study the influence of one or two downcomer pipes on the test sequence. The results of this K5.4A test (with one downcomer pipe) are nearly identical with those of the K5.3A test (with two downcomer pipes).

Preparations for test series II:
Parallel to the work done on the second test bundle (RS 36 B),

the activities concerning the expanded measurements for PKL II and the test preparations have been combined here.

USNRC (US Nuclear Regulatory Commission) lend out measuring instruments for expanding and improving the test data within the scope of the PKL II test section. These are the following measuring systems:

- impedance probes in the bundle and in the upper plenum to determine the density and velocity of the medium (through transit time measurements)
- measurements of film thickness on the flow channel walls (test bundle)
- so-called liquid level detectors in the bundle and the upper plenum for measurements of water level and/or water distribution
- turbine probes in the upper plenum (velocity of fluid)
- two-phase mass flow measuring units (instrumented spool pieces) in the primary loops (4 measuring points)
- technical assistance with the construction of a superheated-steam probe
- optical viewing system using video tape recording for the upper plenum.

The detectors for the test bundle itself are attached in or on unheated rods (simulation of the control rod guide tubes) and have been supplied in part. To proof-real detectors in the test bundle (max. 900 °C rod temperature) a 9-rod test set-up is used. So far initial tests have been performed with so-called superheated-steam probes, which are meant to determine the superheating of the steam, if any, and with a high-frequency probe developed at Karlsruhe Nuclear Research Centre (KFK) for determining the portions of water and steam in the flow.

6. Results

For the tests of the first section of the IB series, KWU have compiled the measurement results, and GRS-M and KWU have finished the analyses. On the basis of these results the internals in the upper plenum are to be decided upon for the future tests.

7. Next Steps

- completion of the remaining tests of series IB
- disassembling of the first bundle, rearrangement of the test set-up for test series II and assembly of the second test bundle
- installation of the additional test instruments
- expansion of the data recording unit to about 500 measuring channels and an additional 23 channels with prompt interrogation (max. 3.2 Kc/s per channel).

8. Relation with Other Projects

RS 0036 C

9. References

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10. Degree of Availability

Berichtszeitraum/Period 1. 1. 78 - 31. 12. 78	Klassifikation/Classification 1.2	Kennzeichen/Project Number RS 184
Vorhaben/Project Title Untersuchungen zur Hydraulik des Flutvorgangs und zu bisher noch unberücksichtigten Einflußgrößen beim Wiederbenetzen Investigations on the Influence of Hydraulics during Reflooding	Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 513, Erlangen	
Arbeitsbeginn/Initiated 1. 10. 75	Arbeitsende/Completed 30. 10. 78	Leiter des Vorhabens/Project Leader D. Hein
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating 31. 12. 78	Bewilligte Mittel/Funds

1. General Aim

In order to improve the reflood models, the hydraulics during reflood will be studied in detail.

2. Particular Objectives

A detailed knowledge of the hydraulics in a channel during the reflood phase and the resulting flow pattern will improve the calculation of heat transfer in the unwetted area. With the help of this experiment, criteria will be worked out for the transition from vapour to fog flow and from fog flow to film boiling in order to get more information on the extent of the different heat transfer regions.

The coupling of the rewetting model, based on heat conduction in the wall, with the hydraulics of the channel is a prerequisite for a general applicability of theoretical calculations.

3. Research Program

Rewetting experiments will be carried out for tubes with different hydraulic diameters varying the parameters: initial wall temperature, inlet subcooling, feed velocity and system pressure. Also the ratio of stored heat to the water content within the channel will be varied.

A comparison will be made between Zircaloy and stainless steel clad in the annular test section for the progression of the rewetting front.

To improve the rewetting model information is needed on the rewetting temperature and on the effects near the rewetting front.

4. Experimental Facilities

For these experiments the test-rig used for the program RS 62 was modified in order to obtain more detailed information on the hydraulics during the reflood period. For these experiments constant inlet conditions will be stressed.

For special tests an annular test section with a quartz-glass tube for the outer wall will be used.

5. Progress to Date

Under the topic "Formation of new rewetting fronts", additional tests have been performed on the annular test section, using a 0.45 mm stiff wire welded on to the test tube. Also, the rewetting process was verified by the Finite-Element-Program, considering test related data.

In order to evaluate the influence of the energy storage capacity of the test tube insulation on the reflood phase, heat-up tests were performed. Beginning with a tube wall temperature between 200 and 400 ^{o}C and atmospheric pressure heating rate between 3 and 5 W/cm^2 was applied and the further temperature increase of the tube wall was measured as a funktion of time. The measurement results were evaluated and incorporated in the computer program HYDROFLUT.

In addition, evaluations were made of the steam temperature and the phase shift in the second test series, where the latter was determined from the momentum measurement at the test tube outlet.

Experimental calculations were performed with the computer program developed in project RS 184, dealing with the determination of flooding. These tests were to show whether or not the program would also yield satisfactory results under extreme conditions.

For that purpose parameter studies were performed with the computer program HYDROFLUT. The results of a bundle test from series RS 36 were verified by means of a program developed for single tube tests.

6. Results

Tests performed with the welded wire showed that in this special case with established nucleate boiling a new rewetting front occurs in the defective area. This was also confirmed by verification calculation, in which good agreement was obtained with the experimental values. The test calculations performed with the computer program HYDROFLUT showed good agreement with the experiments even with increased heat flux. It appears that the program for extreme experimental parameters yielded the turn-around temperatures ahd the quenching times to within 12 %.

The parameter study revealed good agreement between calculation and measurement over a wide range. Bundle tests can be verfied by the computer model provided the modified boundary conditions have been taken into account.

7. Next Steps

Project completed.

8. Relation with Other Projects

RS 36, RS 62

9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	1.2	RS 268
Vorhaben/Project Title Vorprojekt zur experimentellen Untersuchung der Einflüsse mehrdimensionaler Effekte beim Fluten Preliminary Project: On the Experimental Investigations of Multi-Dimensional Effects Influencing Flooding		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 52, Karlstein
Arbeitsbeginn/Initiated 1. 1. 77	Arbeitsende/Completed 31. 1. 78	Leiter des Vorhabens/Project Leader Dr. Melchior
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating 31. 12. 78	Bewilligte Mittel/Funds

1. General Aim

Preparation of a feasibility study for the "3D-experiment". This experiment will serve to study the 3-dimensional flow effects in the upper plenum of a PWR during the refill and reflood phase after a LOCA. Moreover computer programs describing these effects will be verified with the experimental results.

2. Particular Objectives

This study is expected to render results upon which a decision will be made on whether or not the "3D-experiment" will be performed.

3. Research Program

3.1 Specification of the planned test, estimate of usefulness by the architect engineer.

3.2 Systems-related technical design of the test stand (test stand concept).

3.3 Procedural preparation of the test stand concept.

3.4 Preparation of specifications for the measurements and data acquisition, conceptual design of the data acquisition system.

3.5 Evaluation of expenditures, dates and personnel requirements for the tests.

4. Experimental Facilities

Experimental Facilities are not required for the test.

5. Progress to Date

During a session held by the "SK Notkühlung" the make-up of the final report was discussed. The discussion resulted in extensive changes being made in the report. Especially with respect to the 3D-experiment, an overall revision was made as to the reason why this experiment will be performed and what the predicted benefit will be.

In regard to design data, the required steam supply for the test stand as a function of the fracture size was significantly extended. The problem of coupling the 2D- and 3D experiment was thoroughly discussed, taking into account the possibility of the two experiments not being performed parallel to each other; ideas developed by GRS were incorporated. Preliminary copies of the revised report were distributed to the SK-members, the BMFT, USNRC and JAERI.

6. Results

A technical solution of the 3D-experiment was prepared by which the thermohydraulic behaviour of a PWR upper plenum can be examined during the refill- and reflood-phase after a LOCA. Both hot and cold two ended ruptures can be examined, whereby each time 1 functioning and 1 ruptured loop can be simulated. The design pressure for the test stand is 10 bar, the necessary steam flow is 340 kg/s max.

The test stand will probably be erected at GKM (Mannheim).

7. Next steps

Project is completed.

8. Relation with Other Project

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9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	1.2	RS 268 A

Vorhaben/Project Title	Land/Country	
Vorprojekt zur experimentellen Untersuchung der Einflüsse mehrdimensionaler Effekte beim Fluten Preliminary Project: Experimental Investigations of Multi-Dimensional Effects Influencing Flooding	FRG	
	Fördernde Institution/Sponsor	BMFT
	Auftragnehmer/Contractor	KRAFTWERK UNION AG Reaktortechnik R 11, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 4. 78	30. 11. 78	F. Winkler
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	31. 12. 78	

1. General Aim

The study of an experimental investigation of the influence of multi-dimensional effects on flooding (3D-Experiment), conducted as a research task, is concluded. Because of the importance of basic individual points of the study and with regard to the desire and recommendations for changes, the discussions that have taken place within the Federal Republic of Germany and in the framework of trilateral cooperation have led to an expansion of the 3D-Experiment.

2. Particular Objectives

The objective of the planned work is therefore, through intensive work on individual, although essential construction and procedural details, to clarify still-existing problems, to test the technical feasibility of the recommended changes and program expansions, and to determine the accompaning increased costs.

3. Research Program

3.1 Design of the 3D test facility with downcomer (simulation of a 180^{o} sector).

3.2 Investigation of the possibility to fabricate the 3D test facility from carbon steel.

3.3 Planning of the water/steam mixer.

3.4 Conceptual planning for important parallel research. (Planning for the execution of the experiment is not a component of the supplementary contract. It is expected that a follow-on contract will be placed).

3.5 Investigation of the wall effects in the test facility (required by the simulation of a half of the upper plenum) on the experimental results.

3.6 Investigation of aspects of the downcomer simulation.

3.7 Investigation of the possibility of alternative emergency cooling systems.

3.8 Investigation and clarification of the modifications of the upper core support plate and upper tie plate.

3.9 Investigation of the adequacy of simulation of reactor conditions in the test facility in regard to the perforated baffle in the upper plenum.

3.10 Development of the flow diagram of the 3D test facility with a downcomer and simulation of a 360° sector.

4. Experimental Facilities

Experimental facilities and computer programs are not necessary for this study of the 3D Program.

5. Progress to Date

To 3.1 Development of the flow diagram and design of the test facility under the conditions of the end phase of the blow-down, refill and flooding phase for cold and hot leg breaks. Development of the regulating and control concepts for the test facility. Determination of the maximum steam requirements and the maximum exhaust steam flow rate in the test facility on the basis of agree-upon best estimate assumptions for the reactor accident.

Establishment of the controlling dimensions of the test vessel. Design of the test vessel with built-in components and important auxilary components.

Break-down of the test facility into systems and subassemblies. Appogiatura for the three-dimensional distribution of the test instrumentation.

Estimate of the maximum number of channels and signals frequencies to enable the design of the data logger.

To 3.2 Consideration of the material specifications for the test facility.

Development of a proposal for the surface treatment of the

material for the minimization of corrosion with the use of ferritic material.
Material specification for the individual components in the light of a specification of ferritic material for the test facility.

To 3.3 Development proposals for the specification of the core simulator.

To 3.4 Consideration of accompaning important structural investigations.

To 3.5 + 3.9 Estimation of the steam cross-flow in the upper plenum of a reactor during the refill and flooding phases, with cold and hot leg breaks.
Development of a proposal for the structural specification of the baffle in the upper plenum and the simulation of the baffle steam flow.

To 3.6 Consideration of the advantages of integrating the downcomer.

To 3.7 + 3.8 3.10 Constructive proposals were developed for alternative emergency cooling systems and other reactor geometries.
Flow diagram and thermohydraulic design of the test facility with and without simulation of the end phase of blowdown.

6. Results

To 3.1 The flow diagram and the technical design data of the 3D test facility were proposed. On technical and financial grounds the high steam demand during a test run is provided by Ruths-variable pressurizer. The exhaust steam flow through the hot and cold leg breaks as well as across the perforated baffle in the upper plenum is condensed in the containment or in a water tank. The steam or water mass injected into the core simulator in the refill and flooding phases is discharged under control as water out of the lower plenum. The required extent and three-dimensional distribution of the advanced instrumentation was established and a recommendation sent to the USNRC.

To 3.2 The use of ferritic materials is a major consideration in the light of the proposed conservation of resources. Problems could arise in regard to the adhesion of the magnetite layer as the result of the stresses caused by thermal shocks.

The use of ferritic materials is further endangered by the use of water with a high electrical conductivity, as required by the use of the conductivity probes.

To 3.3 Different development scenarios were put forth for the injection of steam and water in the core simulator.

To 3.4 Investigations of the following problems should be carried out:

steam and water injection in the core simulator,
controllable flow fluctuations in the steam and water supply lines to the core simulator,
regulation and control,
water spraying into the containment (condensation of steam)

To 3.5 + 3.9 The steam flow across the perforated baffle in the upper plenum reaches the highest value in the refill phase (hot leg break: 200 kg/sec). Also, in the flooding phase in most cases, the baffle-steam flow cannot be neglected.
The steam flow across the perforated baffle is regulated by steam inlet valves and exhaust steam valves.

To 3.6 The integration of the downcomer enables the investigation of water-steam counter current flow in the downcomer with the original geometry during the end-phase of the blowdown and refill phases. In addition, a test facility with a downcomer conveys the advantage that across the intact loops a flow path in parallel to that to the upper tie plate exists between the upper plenum and the core, so that in the test facility the pressure difference between the upper plenum and the core simulator acts automatically the same as in the reactor.
The existence of the downcomer permits the investigation of a possible hydraulic fluctuation in the region of the core-downcomer during the flooding phase.

To 3.7 + 3.8 Alternative emergency cooling systems including vent valves (BBR) and other reactor geometries (the upper plenum of the C-E design) can be investigated.

To 3.10 The flow diagram for the 360° concept with and without simulation of the end phase of the blowdown is being submitted.

7. Next Steps

The program is concluded.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period 1.1. - 31.12.1978	Klassifikation/Classification 1.2	Kennzeichen/Project Number RS 314
Vorhaben/Project Title Flutprogramm-Entwicklung Application and Development of Reflood Computer Codes		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH
Arbeitsbeginn/Initiated 1.10.1977	Arbeitsende/Completed 31.12.1980	Leiter des Vorhabens/Project Leader Prof. Hicken
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 31.12.1978	Bewilligte Mittel/Funds

1. General Aim

Parallel and complementary investigations to the experimental research program sponsored by BMFT as well as the development of computer codes for the analysis of the refill and reflood phases of LOCA.

2. Particular Objectives

2.1 Application of reflood codes within the frame of analytical work for the refill and reflood experiments 2D/3D, CCTF and PKL.

2.2 Development and improvement of reflood codes.

2.3 Implementation of foreign codes.

3. Research Program

3.1 Analytical assistance to 2D/3D.

3.2 Analytical assistance to CCTF.

3.3 Analytical assistance to RS 287 (PKL).

3.4 Check out of further codes.

3.5 Development of numerical methods of high efficiency.

3.6 Development and improvement of single models.

3.7 Development of the FLUT-pilotprogram.

3.8 Implementation of foreign codes.

4. Experimental Facilities, Computer Codes

The refill and reflood experiments within the frame of RS 287 (PKL) are performed by KWU, Erlangen. The transformation of the data stored on magnetic tapes into engineering units is made by a code. The evaluation and the presentation of the results will be carried out more and more by appropriate codes. At the present the code REFLOS /1/ which is utilized by GRS in the licensing practice, is used for the calculations. The computer code TRAC /2/, which was taken over from LASL in version 16.3, will be used for the calculation of particular problems of the refill and reflood processes.

5. Progress to Date

Ad 3.1 Continuation of preparatory work and contribution to the actual planning phase of the SCTF and UPTF: Comments on the program itself /3, 4, 5, 6, 21/, calculations and estimations of different parameters during the LOC phases end-of-blowdown, refill and reflood related to a 1300 MWe PWR /7, 8, 9, 10/, proposals, suggestions and calculations for the design of the SCTF /11, 12, 13, 14, 15/, and for the design of the UPTF /16, 17, 18, 19, 20/, proposals for the instrumentations /22/ and the test matrices /23, 24/.

Performance of a deepening training in the handling of the advanced code TRAC-P1 (16.3) and in combination with modifications of the grafics output file improvement of plotting routines in order to get a more detailed information from the calculated results.

Analysis of some results of TRAC calculations of the PWR reference reactor and the SCTF. Performance of first TRAC runs for a 180 and a 360 section of the upper plenum.

Participation in different working groups and national and international experts meetings.

Ad 3.2 No activities in this period.

Ad 3.3 Development and test of different routines of a code system for the automatical evaluation of PKL experimental data, e.g. interpretation of mass flows, removal of stored energy from steam generator internals and the heater rods, presentation of the spatial quench front propagation in defined levels. Performance of a post test calculation of the PKL test K1.3 by using the TRAC-P1 (16.3) code up to 227 s problem time. Performance of a comparative evaluation of the tests K1.1, K1.5, K1.7, K5,4a /25/. Performance of the analysis of the subseries K14 /26, 27/. Preparation of the analysis of the test K1.3, K2, K5a, K5.1b with respect to the influence of the power profile and of the tests K1.3, K1.4, K5a, K5.3a referring to the loopresistance influence. Participation in different working groups and experts meetings.

Ad 3.4 First analysis of the RELAP4-Mod 6 heat transfer models, especially with respect to an application to the refill and reflood problems.

Ad 3.5 Continuation of the work on the development of efficient high-order method for the solution of fluid dynamics equations /28, 29/, especially with respect to the procedure itself and to the coding. Alternative preparation of a Argonne version for the time integration procedure by Gear. Performance of comparative calculations with the codes ASWRP2-IMEX and TRAC-P1 (16.3) by using the DF-models /30, 31/.

Ad 3.6 Analysis of single models of the TRAC-P1 (16.3) code. Calculation of some 30 sample problems in order to check these models by the simulation of the reflood process in pipe - U-tube- and vessel-geometries using subcooled and saturated fluid /32, 33, 34, 35, 36/. Based on the detailed knowledge of the different TRAC models, proposals of models have been made to be built into the FLUT code.

Ad 3.7 Development of the basis of a two fluid model for the simulation and quantification of the processes occuring during the refill and reflood phases of a LOCA. The main points of the development have been the possible and meaningful appointments of initial and boundary values and the transition conditions from the single phase to the two phase fluid with special respect to the coarse computational mesh method /37, 38, 39, 40, 41, 42, 43/.

Ad 3.8 Performance of a series of well-known adaptions for the conversion of the CDC code TRAC-P1 (16.3) (standard FORTRAN single precision floating point) to IBM (standard FORTRAN-H-Extended with double precision floating point). Implementation and check on the Amdahl 470/V6 computer of GRS /44/. Implementation of RELAP4-Mod6 /45/.

6. Results

Ad 2.1 The results obtained from several runs with the code TRAC-P1 (16.3) are of limited evidence:

- The simulation of the reactor typical end-of-blowdown and refill phases brought problems referring to an application to SCTF.
- The usage of different nodalisations with the simulation of the PKL test K1.3 result in different values of the histories of cladding temperatures and quench times.

From the comprehensive comparative evaluation of the PKL test following results can be derived:

- The influence of different loop resistances on the quench behaviour of the highly heated up rods is minor in the case of a combined injection and is important in the case of only cold leg injection.
- A simulated radial power profile influences the quench behaviour in both cases - combined and cold leg injection - by a 5o % increase of the quenching time. In the cold leg injection case it causes an automatically shut-off by reaching the pre-given temperature limits 900 °C and 930 °C.
- The influence of different constructions within the upper plenum of PKL on the quench behaviour is decisive with respect to the precooling effect during the refill phase and minor during the reflood phase.

Ad 2.2 A comparison between the TRAC-P1 (16.3) DF-model and the two fluid model by simulating a vertical pipe results in a nearly identical axial void distribution and relativ velocities, which correspond in the case of using the DF-model with the empirical correlations. The two fluid model calculates slightly smaller values.

The most important results of the check of TRAC included models are:

- The reflood processes can be simulated by TRAC.
- The two fluid model has some advantages in the low pressure region of a LOCA.

First test runs with the actual version of the FLUT code have brought satisfactory results.

Ad 2.3 The to IBM converted program TRAC-P1 (16.3) received from LASL in March 1978 is implemented on the Amdahl 470/V6 and released since 27.6.1978.

7. Next Steps

Ad 3.1 Continuation of the analytical assistance to the planning and construction of UPTF and SCTF. Analysis of results obtained by the simulation of SCTF and the reference reactor with TRAC.

Ad 3.2 Begin with the analytical assistance to CCTF.

Ad 3.3 Simulation of a PKL test with TRAC-P1A. Continuation of comparative evaluations.

Ad 3.5 Optimalization of the ASWR procedure.

Ad 3.6 Further checks of TRAC models and preparation for take over into FLUT.

Ad 3.7 Implementation of prepared models in FLUT.

Ad 3.8 Take over and implementation of the version TRAC-P1A.

8. Relation to other Projects

2D/3D experiments.

9. References

/1/ E.J. Kersting: Rechenprogramm REFLOS. Programm zur Berechnung des Wiederauffüll- und Flutvorganges. GRS-A-163, September 1978

/2/ TRAC-P1, an advanced best-estimate computer program for PWR LOCA analysis. LASL, vol I, March 1978

/3/ E.J. Kersting: Stellungnahme zur "program description USNRC preliminary draft - October 1977", 9.2.1978

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/5/ E.J. Kersting: Coupling of the Japanese and German multidimensional facilities, January 1978

/6/ Beiträge zu Appendix III des "draft arrangement between BMFT, JAERI, USNRC"

/7/ E.J. Kersting: Durch LOCA-Analyse zu ermittelnde Parameter für die UPTF - Liste der Rechenläufe für Referenzreaktor, 17.8.1978

/8/ E.J. Kersting: Asymmetrische Effekte bei Reaktoranlagen mit alleiniger Kalteinspeisung, 12.9.1978

/9/ E.J. Kersting: Erforderliche Dampfmengen für die Simulation des Flutvorganges eines KWU - Reaktors der 1300MW Klasse, 13.9.1978

/10/ E.J.Kersting: Selection for desired TRAC-runs for reference reactor with combined injection, handout for 2D/3D meeting in Washington, 6.11.1978

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/13/ E.J. Kersting: Required steam and water flows for the 2D-test facility (SCTF), 29.8.1978

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/17/ E.J. Kersting, J. Keusenhoff: Stellungnahme zur Ausführung des 3D-Versuchsstandes und zur Übertragbarkeit der Ergebnisse, April 1978.

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/20/ E.J. Kersting: Überlegungen zum 360 Grad Konzept, Interessenlage der Vertragspartner, Sinn und Zweck der Simulation von end-of-blowdown, Vor- und Nachteile der verschiedenen Konzepte für einen 3D-Versuchsstand, 11.9.1978

/21/ V. Teschendorff: Kostenabschätzung für analytische Arbeiten zum 3D-Projekt, Tes-78-4, 6.6.1978

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/24/ R. Kirmse: Entwurf für die 3D - Matrix, Kim-78-1, 24.5.1978

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/26/ Tischvorlage für die Besprechung der SK Notkühlung AG "PKL-Einbauten im oberen Plenum", 27.11.1978
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/30/ P. Romstedt, V. Teschendorff: Vergleichsrechnungen TRAC-P1 (16.3)/ASWRP2 - Driftflux, Interne Mitteilung, 14.11.1978
/31/ P. Romstedt: Vergleichsrechnungen TRAC-P1 (16.3.)/ASWRP2-Driftflux, Interne Mitteilung, 11.12.1978
/32/ V. Teschendorff: Darstellung des Pools auf der oberen Gitterplatte durch TRAC, Tes-78-2, 24.5.1978
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/35/ V. Teschendorff: Benutzerhinweise zu den Komponenten von TRAC-P1 (16.3), Tes-78-10, 14.9.1978
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strömung, Buh-78-8, 9.6.1978
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/42/ W. Buhl: Transformation der Impulsgleichungen der beiden Fluidphasen auf Gleichungen für den Gesamtimpuls und die Relativgeschwindigkeit, Buh-78-10, 21.6.1978
/43/ W. Buhl: Einfache Modelltypen für die "rechten Seiten" - Konstitutive Gleichungen, Buh-78-12, 31.10.1978
/44/ A. Schmidt: Report on the Conversion of the LASL-code TRAC-P1 GRS-A-206 (September 1978)
/45/ A. Schmidt, K. Siegmund: Freigabe von RELAP4/Mod 6, update 4 und des zugehörigen Ploprogrammes sowie der benötigten Unterprogramme aus dem "environmental subroutine package", Interne Mitteilung 2.2.1979

10. Degree of Availability of the Reports

The notes and internal reports are not available. The results are presented in GRS-A-... reports. These reports are available through the Gesellschaft für Reaktorsicherheit (GRS) mbH, Forschungsbetreuung, Glockengasse 2, D-5000 Köln 1.

	Classification: 1.2
Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: NORHAV - RHC a core heat-up computer program	Organization: Risø National Laboratory
Initiated date: November 1971 Completed date: 1976 Status: completed	Scientists: J.G.M. Andersen H. Abel-Larsen Preben Hansen

1. General aim

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

2. Particular objectives

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

3. Experimental facilities and programme

4. Project status

1. Progress to date

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

2. Essential results

5. Next steps

Development of a flooding version of RHC, called NORCOOL-I.

6. Relation with other projects

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

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

b) The Danish transient subchannel computer program TINA and the one dimensional blow down code RISQUE under development at Risø.

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

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

7. Reference documents

Jens G. Munthe Andersen:

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

8. Degree of availability

	CLASSIFICATION: 1.2
TITLE (ORIGINAL LANGUAGE):	COUNTRY: DENMARK
	SPONSOR: Risø National Laboratory
TITLE (ENGLISH LANGUAGE): Inverted annular film boiling during the reflooding phase.	ORGANISATION: Risø National Laboratory
	PROJECT LEADER:
INITIATED: September 1977 / COMPLETED:	SCIENTISTS: Per Ottosen
STATUS: Progressing. / LAST UPDATING:	

1. General Aim

Heat transfer on vertical surfaces in tubes under inverted annular film boiling. Measurements of axial void fraction distribution.

2. Particular objectives

An experimental and theoretical work concerning inverted film boiling in glass tubes using N_2 as flow medium and steel tubes using water as flow medium has been started.

Later on a theoretical work based on the experimental results will be started.

3. Experimental facilities

A visual observable test section made of glass tubes has been constructed.

The mean void fraction can be measured by γ-ray absorption by an error of ± 1,5% void fraction.

4. Status

4.1. Progress to date

The first visual experiment using N_2 as a two-phase flow in a heated glass tube has been tried.

The instruments for the measurements are still being developed.

5. Next steps

Construction of a test loop, in which it is possible to establish inverted film boiling using water as medium.

145-1-07		1-2
TITRE Thermohydraulique du LOCA. Etude des interactions mécaniques et thermo-dynamiques dans l'injection de secours d'un réacteur PWR : Programmes EPIS I et II.		Pays FRANCE Organisme Directeur CEA/DgCS - EDF/SEPTEN
TITLE (Anglais) LOCA thermohydraulics. Steam-water mixing studies for PWR : EPIS I and II projects.		Organisme exécuteur CEA/DTCE (SACLAY) Responsable
Date de démarrage 01/01/75	Etat actuel en cours	Scientifiques
Date d'achèvement 31/12/80	Dernière mise à jour 01/79	

1. OBJECTIF GENERAL

Etude des phénomènes se produisant lors de l'injection d'eau de secours par accumulateurs et pompes au cours d'un accident de dépressurisation d'un réacteur pressurisé.

2. OBJECTIFS PARTICULIERS

Développer des modèles physiques pour interpréter les expériences.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

EPIS I : Etude de l'interaction mécanique par injection d'eau dans un débit d'air (échelle 1/11).

EPIS II : Etude des interactions mécaniques et thermodynamiques par injection d'eau dans un débit de vapeur (échelle 1/25).

4. ETAT DE L'ETUDE

1) Avancement à ce jour

Essais EPIS I terminés. Essais complémentaires EPIS I' terminés.
Essais eau-air sur géométrie EPIS II terminés.
Installation EPIS II opérationnelle, campagnes d'essais 1 et 2 terminées.

./.

2) Résultats essentiels

- EPIS I : Evolution des pertes de charge au niveau de l'injection en fonction des paramètres principaux :

  . angle de piquage de l'injection, rapport des vitesse de l'eau et de l'air, niveau de pression de la cuve .

  . Test de différents modèles physiques (en cours) qui permettraient de retrouver soit le ΔP total, soit la ligne piézométrique .

  . La force axiale sur la tuyauterie peut être négligée dans la modélisation sur géométrie EPIS II .

  . La connaissance de la ligne piézomètrique permet de calculer le taux de vide à l'aval de l'injection .

- EPIS II : Observation de phénomènes oscillatoires dans des domaines précis de températures et de titres .

  . Mesure des lignes piézométriques moyennes pour les régimes stationnaires et oscillatoires .

  . Première modélisation en stationnaire (dans la marge d'erreur des mesures) .

  . Premières bases d'un modèle d'oscillations .

5. PROCHAINES ETAPES

EPIS I : Continuer la modélisation, en particulier :

  . Etude de Θm (du modèle HEXECO) et de ses incertitudes .

  . Modélisation du noeud d'injection en visant l'extrapolation au cas du réacteur .

EPIS II : Campagnes d'essais avec instrumentation amont et aval plus fine (capteurs piézométriques, microthermocouples, mesure de taux de vide) .

Développement de modèles en stationnaire et en régime oscillatoire

7. DOCUMENTS DE REFERENCE

- "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" .
  1ère partie : Expérience EPIS I, rapport SEEN RT 76-014

- "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"
  2ème partie: EPIS I PRIME I, rapport SEEN RT 77-103

- EPIS II

  Compte rendu de la campagne d'essais n°1
  Rapport SEEN RT 77-117

- "Etude préliminaire à l'injection de secours"
  3ème partie: EPIS I PRIME II
  par CHAULIAC, LAMOUREUX - Rapport SEEN RT 78-153 A

- EPIS II

  Compte rendu de la campagne d'essais n°2
  Rapport SEEN RT 78-197

145-1-05		1-2
TITRE Thermohydraulique du LOCA : Etude expérimentale du refroidissement de secours des réacteurs à eau - Programme ERSEC.		Pays FRANCE
		Organisme Directeur CEA-Dg/CS EDF/SEPTEN
TITLE (Anglais) LOCA Thermohydraulics : Experimental investigation of water reactors safety injection : ERSEC project.		Organisme exécuteur M. COURTAUD STT-GRENOBLE
		Responsable
Date de démarrage 01/01/72	Etat actuel en cours	Scientifiques
Date d'achèvement 31/12/80	Dernière mise à jour 01/79	

1. OBJECTIF GENERAL

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

2. OBJECTIFS PARTICULIERS

Développement de modèles physiques pour l'interprétation des expériences.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

La boucle ERSEC comprend une section d'essai qui peut être :
soit un tube (Øint = 12 mm, l = 3,25 m), soit un boitier carré contenant une grappe de 36 barreaux, entourée des éléments permettant de fournir les conditions de l'essai. A la sortie de la section d'essai, le mélange eau-vapeur est séparé :

- l'eau est récupérée dans un réservoir où son volume est mesuré ;
- la vapeur est envoyée sur des vannes de régulation permettant de garder la pression constante, puis dans un condenseur.

Les principaux paramètres sont :

- la pression : 1 à 6 bars
- le flux de chaleur : 3 à 7 W/cm2
- le débit surfacique d'injection : 2 à 18 g/cm2 sec.
- la température initiale de paroi : 300 à 900°C.

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Le programme comprend :

. Des expériences de renoyage à débit constant en tube,

. Des expériences de renoyage à charge constante en tube

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

4 - ETAT DE L'ETUDE

1) Avancement à ce jour

- Une première campagne d'essais de renoyage à débit constant sur une grappe 36 barreaux PWR 17x17 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 : terminés.
- Essais de renoyage à débit constant sur tubes de différentes longueurs chauffants.
- Mise au point de la mesure de taux de vide dans l'écoulement amont du front de trempe par diffusion d'un faisceau de neutrons.
- Interprétation en cours des expériences : progression du front de trempe avec le code de conduction bidimensionnelle associé avec un modèle de coefficient d'échange suivant :

  à la loi classique de Forster Zuber en ébullition nucléée bornée par la relation de flux critique de Zuber, on ajoute un terme en

$$\frac{k \quad d\,T_p}{dz}$$

  qui traduit l'effet d'inertie thermique de la couche limite.

  Etude de l'écoulement amont.

- En cours : essais en grappes 36 barreaux avec minimisation des effets de boitier.

2) Résultats essentiels

La valeur de k a été corrélée pour l'ensemble des essais en géométrie tubulaire.

Développement de modèles physiques représentant le rayonnement en aval du front de trempe.

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

Une méthode de mesure de taux de vide dans les écoulements basse pression, basse vitesse a été mise au point.

5 - PROCHAINES ETAPES

- Essais en grappe (influence des oscillations de débit)
  Ces essais sont effectués à débit forcé avec des oscillations d'amplitude et de fréquence égales à celles observées dans les expériences globales (SEMISCALE, PKL).

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- Expériences fondamentales en vue d'étudier l'écoulement amont (mesures de la ligne piezomètrique et du profil axial du taux de vide) pour l'ajustement du modèle théorique "codes avancés".
- Interprétation des échanges thermiques en zone asséchée à l'aide du code FLIRA.

7 - DOCUMENTS DE REFERENCE

- "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 renoyage utilisés dans l'étude du refroidissement de secours des réacteurs à eau - Choix d'un modèle".
  P. CLEMENT Note DTCE-STT 507.
- "FLIRA : Reflooding calculation model following an accidental primary fluid loss".
  JP. L'HERITEAU and D. MENESSIER
  European two Phase Flow Meeting Haïfa (June 1975)
- "Modélisation thermohydraulique des écoulements et des échanges de chaleur au cours du renoyage d'un coeur de réacteur à eau pressurisée".
  P. RAYMOND
  Thèse de 3ème cycle (Nov. 77)
- "Modeling of quench front progression and heat transfer by radiation during reflooding of a tubular test section".
  P. CLEMENT, R. DERUAZ
  European two phase flow meeting - Erlangen (Juin 76)
- "Modeling of heat transfer by radiation during the reflooding phase of LWR".
  R. DERUAZ, B. PETITPAIN
  Specialist's Meeting on the behaviour of water reactor fuel elements under accident conditions.
  SPATING (Norway) (Sept. 76)
- "Interprétation des essais ERSEC sur le refroidissement de secours des réacteurs à eau pressurisée au moyen du code FLIRA".
  C. REVIGLIO
  Thèse de 3ème cycle (Nov. 77)
- "Some aspects of reflooding studies in France (1977)
  N. TELLIER
  4th meeting of the CSNI-ECCS ad hoc group
  GRENOBLE, Juin 1977

./.

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

- "Essais préliminaires de mesure de taux de vide par diffusion ou transmission de neutrons"
R. FREITAS et AL
Compte rendu d'essais TT/SETRE-78/22 - B/RFr.

- "Heat Transfer Modeling of the quench front during reflooding phase of a LOCA"
P. CLEMENT, P. REGNIER
European Two Phase Flow group Meeting Paper E5 (STOCKHOLM, Juin 1978)

	CLASSIFICATION: 1.2
TITLE (ORIGINAL LANGUAGE): Studio teorico-sperimentale della termoidraulica connessa alla refrigerazione di emergenza per allagamento dal basso.	COUNTRY: Italy
	SPONSOR: Politecnico di Torino
TITLE (ENGLISH LANGUAGE): Theoretical and experimental study of reflooding thermohydraulic	ORGANISATION: Politecnico di Torino (*)
	PROJECT LEADER: M. De Salve
INITIATED: January 1976 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: June 1979	

(*) Istituto di Fisica Tecnica ed Impianti Nucleari.

1. General aim and particular objectives.
   This experimental and theoretical study is to improve the knowledge of the heat and mass transfer mechanism during reflooding.

2. Particular objectives.
   To find, for different linear thermal capacities and lenghts in tubular and annular test section, the influence of initial thermal energy and of the linear thermal power on the chronological occurrence of heat transfer regimes and heat flux versus flow rate and inlet subcooling.

3. Experimental facilities and programme.
   An experimental facility with an inner heated annular test section and tubular 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 and high flooding rates, high initial wall temperatures ($T \simeq 800°C$) and high and small inlet subcooling. Some tests have been performed. Bottom flooding tests have been performed in a tubular test section with variable inlet subcooling, heated power, flow rate.

4. Progress to date.
   Many tests in both rod annular and tubular test sections have been performed (I.D.8 mm and O.D. 10 mm) in the following ranges: initial wall temperature from 600 to 800°C; flow rate from 5 to 180 g/cm^2s; inlet subcooling from 80 to 10°C.
   The experimental results show the dependency of the quenching temperature and rewetting front velocity from the flow rate, initial wall temperature, inlet water subcooling and the linear thermal power. Liquid carry-over is collected.

5. Next steps.
   Theoretical models have been analysed and an attempt to evaluate the heat transfer coefficients in all regions during rewetting is going on.
   A theoretical approach to couple the conduction to hydrodynamically controlled model will be performed. Experimental tests to study the rewetting with constant head flow

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Theoretical and experimental study of reflooding thermohydraulic	1.2

rate will be performed.

6. Reference documents and degree of availability.
All the results will be available.

- M. De Salve
Aspetti termoidraulici nel ribagnamento di un elemento riscaldante ad elevata temperatura.
PT IN 99, settembre 1977, Politecnico di Torino.

- M. De Salve
Esperienze di allagamento di un elemento riscaldante ad elevata temperatura.
PT IN 101, novembre 1977, Politecnico di Torino.

- M. De Salve
Modelli matematici per lo studio del ribagnamento di elementi riscaldati ad elevate temperature. Parte I (Modelli monodimensionali).
PT IN 102, febbraio 1978, Politecnico di Torino.

- G. Del Tin, M. De Salve, B. Panella
Heat transfer during rewetting by bottom flooding.
ENS/ANS International Topical Meeting on nuclear reactor safety, Brussels, October, 16-19,1978.

- G. Del Tin, M. De Salve, B. Panella
Geometry and heat capacity effects on heat transfer during rewetting by bottom flooding.
2nd multi-phase flow and heat transfer symposium workshop, Miami-Beach, Florida, U.S.A., April,16-18,1979.

- G. Del Tin, M. De Salve, B. Panella
Heat transfer during rewetting by bottom flooding in tubes of different thickness.
European two-Phase Flow Group Meeting, Ispra, June,5-8, 1979.

(Politecnico, Istituto di Fisica Tecnica ed Impianti Nucleari,
Corso Duca degli Abruzzi 24, I-10124 Torino)

	CLASSIFICATION: LWR 1.1 & 1.2
TITLE (ORIGINAL LANGUAGE): Einfluß der DWR-Umwälzschleifen auf den Blowdown (LOBI Projekt)	COUNTRY: FRG C.E.C. SPONSOR: FRGMRT C.E.C.
TITLE (ENGLISH LANGUAGE): Influence of the PWR loops on the blowdown (LOBI project)	ORGANISATION: C.E.C. J.R.C.-Ispra PROJECT LEADER: W. L. Riebold
INITIATED: Dec. 1, 1973 COMPLETED: Nov. 30, 1981 STATUS: Completion of shake-down tests LAST UPDATING: May 1979	SCIENTISTS: Eder, Fortescue, Fritz, Kolar, Larsen, Mörk-Mörkenstein, Ohlmer, Piplies, Städtke

PROJECT TITLE: Reflood Behaviour of PWR Fuel	CSNI INDEX NO: 1.2
SPONSORING COUNTRY: United Kingdom	ORGANISATION: Atomic Energy Establishment, UKAEA, Winfrith
DATE INITIATED: 1978 DATE COMPLETED: Continuing	PROJECT LEADER: K G Pearson

DESCRIPTION

1. General Aim

To improve understanding of the heat transfer and hydraulic processes during PWR reflood following LOCA in order to improve the effectiveness of Emergency Core Cooling.

2. Particular Objectives

To provide experimental data needed to improve models of reflood phenomena, especially rewetting of the fuel. Particular aspects of concern are:-

(a) Influence of clad properties and local two phase heat transfer on rewetting.

(b) Effects of cluster distortion, for example clad ballooning, on heat transfer and on quenching.

3. Experimental Facilities

(i) REFLEX Rig. A single tube reflood test rig with highly insulated test sections representing both undistorted and ballooned fuel.

(ii) CREATE (Cluster Rig Electrochemical Analogue Tests). This is used to obtain detailed distribution of heat transfer coefficients in ballooned clusters.

(iii) Emergency Core Cooling Test Facility: A test rig capable of mounting large full length electrically heated bundles (eg 8 x 8) for emergency cooling experiments at pressures up to 70 bar.

4. Project Status

REFLEX is being used for single tube bottom flooding tests. Electrochemical analogue results have been obtained for a 67 pin ballooned cluster.

5. Next Steps

Single tube and 3-D test sections simulating can ballooning are being designed for REFLEX. Indirectly heated ballooned fuel pin simulators are being developed for large scale cluster tests.

CLASSIFICATION 1.2

Title 1: P.W.R. Refill Studies

Title 2:

Initiated: 1st November 1975

Completed:

Status: Progressing

Last Updated

Country: United Kingdom

Sponsor: UK - NII

Organisation: Strathclyde University

Project Leaders : H C Simpson, D H Rooney

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1. General Aims:

To simulate the refill process in a P.W.R. downcomer and study its effectiveness.

2. Particular Objectives:

To produce a theoretical model, or correlation, defining the refill process enabling the limiting conditions to be predicted.

3. Experimental Facilities and Programme:

Work to be carried out in three phases. Phase 1 is developed annulus with tangential water injection. Phase 2 is developed annulus with normal water injection. Phase 3 is 1/10 scale model of P.W.R. downcomer. All test sections transparent, fluids steam and water, pressures just above atmospheric. Measurements to be taken include steam and water flowrates, pressures, temperature distributions. Cine photography to capture liquid bridging effects.

4. Project Status:

Phase 1 data being collected.

5. Next Steps:

Production of theoretical model

6. Relation with Other Project:

Similar in some respects to the Wallis work at Creare.

Classification 1.2 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: 1.1.2 1.2
TITLE (ORIGINAL LANGUAGE): WATER COOLED REACTOR DEPRESSURIZATION STUDIES:- CSNI STANDARD PROBLEM CALCULATIONS FOR ECCS.	COUNTRY: UK
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/UKAEA
	PROJECT LEADER: G M JORDAN
INITIATED: 1974 / COMPLETED:	SCIENTISTS: A A DEBENHAM
STATUS: Continuing / LAST UPDATING: MAY 1979	

1. GENERAL AIM

		Classification 1.2, II.1
Title 1	Thermohydraulic Safety Studies Heater Rod Cluster Rig	Country U.K.
Title 2	Effects of Fuel Pin Ballooning on Reflood Heat Transfer	Sponsor Organisation CEGB
Initiated 1.6.78 Status Continuing	Completed Last updating	Project Leaders S.J. Board S.A. Fairbairn

The aim of this project is to study some of the effects of fuel pin ballooning on emergency cooling heat transfer. Measurements will be made of heat transfer around ballooned pins in an otherwise undistorted pin array. Electrically heated rods (∿ 1 metre heated length) will be used as fuel pin simulators. These rods will be supported vertically in a cylindrical silica sleeve (to provide a flow visualization facility) at near atmospheric pressure. Pin ballooning will be simulated by attaching metal sleeves to the heater rods. Initially, the sleeves will be attached at the same axial level to look at the effects of a coplanar blockage over a fraction (∿ 30%) of the area of the rod cluster. Direct measurements will be made of superheat in the steam and steam-water flows used to cool the rods. Rod temperatures will be measured by internal thermocouples.

CLASSIFICATION 1.3

Berichtszeitraum/Period 1.1.78 - 31.12.78	Klassifikation/Classification 1.3	Kennzeichen/Project Number 06.01.04(PNS 4231)
Vorhaben/Project Title Theoretische Untersuchungen zum Brennstabverhalten beim Kühlmittelverluststörfall Theoretical Investigations of Fuel Behavior under Accident Conditions		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe GmbH (KfK) Projekt Nukleare Sicherheit (PNS) IRE, INR, IKE Stuttgart
Arbeitsbeginn/Initiated 1.1.73	Arbeitsende/Completed 31.12.80	Leiter des Vorhabens/Project Leader Dr. R. Meyder, Prof. H. Unger
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds Dr. H. Borgwaldt

1. General Aim

he aim of this project is the development of a code system (SSYST) and associated codes in order to model and calculate the behaviour of Zircaloy clad fuel rods in different phases of a LOCA. Especially the effect of ballooning and its consequences are studied and described with particular emphasis. Also, the theoretical investigation of the influence of ballooning on the effectiveness of emergency core cooling is a major aim of the planned activities.

2. Particular Objectives

Development of the modular code system SSYST and its complementation by associated codes. SSYST allows to simulate the interaction between heat conduction in a fuel rod, heat transfer in the gap, swelling and ballooning of the clad, pressure in coolant and fuel rod as well as the thermo- and fluiddynamic conditions in the coolant channel and the primary system of a LWR. Additional theoretical investigations are concerned with cooling conditions in deformed areas of fuel elements and the interactions between rods.

3. Program

The development covers three topics

- Development and verification of a computer code for single rod analysis of a LOCA
- Investigation of interacting fuel rods, behaviour of bundles
- Investigation of geometrical configurations for longterm coolability.

4. Computer codes

Code system SSYST. Associated codes: COMETHE IIIJ, RELAP4/MOD6, (COBRA IV), ZYBDA.

5. Progress to date

Code development.

To save computer costs the IBM-version of the SSYST code was analysed with respect to consumed computing time and improved where it seemed to be effective. All important modules have been adapted to this accelerated version, so that it is in use now for routine LOCA analysis. Also in the context of system improvement a first version of the module VARIO (VARiablen-Initialisierung zur Objektzeit) has been completed. It will bring an essential support for users in specifying an SSYST-input.

For an improved simulation of the primary system RELAP4/MOD6 was transferred from EG & G to Karlsruhe. The compilation of the received source deck with our FORTRAN-H-EXTENDED compiler showed up a number of errors which have been corrected. For plotting RELAP results a link was made to REGENT. The 8 test sample problems supplied by EG & G have been verified. The implementation work was completed and the RELAP4/MOD6 code system is now available for further checkout activities and first applications. For a better determination of the initial conditions of a rod COMETHE IIIJ, from Belgonucleaire was implemented and coupled to SSYST. To model the active link between heat transfer in the fuel rod and the thermodynamics in the subchannel the module ZETHYD for the blowdown phase has been attached to SSYST in Karlsruhe. The equivalent module for the refill and reflood phase, ZETHYF, has been developed continuously.

To represent the effect of azimuthally varying strains shown up by in- and out-of-pile experiments the module AZI was set up. It models the effect of excentricity between pellet and cladding within 180° of a rod slice and consequently calculates local strains and temperatures. The excentricity between fuel and cladding can be varied during a transient. Azimuthal variations of coolant heat transfer and radiation heat transfer to the environment are also included. Also for a better determination of final strains the NORA model, for Zircaloy creep at high temperatures, was extended to describe the effect of oxygen on strain rates. For burst strains the data base ZYBDA was set up and filled with more than 300 well defined experiments. These data have been analysed in a first step.

Application of SSYST.

In the field of probabilistic approach to fuel rod behaviour under LOCA conditions input-generation and data-transfer logics throughout the SSYST modules PROFAN have been highly automized. A set of 70 SSYST runs has been performed and first probability density functions for maximum clad strain and peak cladding temperature of a hot fuel rod have been determined. Recalculation of the experiments LOC-11, REBEKA and COSIMA was performed with great effort.

Cooling of deformed bundles.

The investigation of the coolability of strongly deformed zones considered radial heat transfer in the bundle and the potential of cooling by an axial steam flow through the residual channels.

6. Essential results

The reformulation of the data transfer in the kernel of SSYST and in the module ZET-1D resulted in a reduction of computing time of about 60 per cent for typical LOCA runs.

The formulation of the heat transfer and energy equation in the module ZETHYF has been improved especially by a more detailed modeling of the droplets in the dispersed flow regime.

The verification of the NORA model (deduced from 1D experiments) with tube data showed the necessity of modeling also the radial stress component. Inclusion of oxygen influence however showed only a small influence for loadings typical for a 2-F-LOCA, but for long lasting transients its hardening effect becomes important. The analysis of the burst data showed a quite good agreement of rupture resp. burst strains between 1 D creep and tensile tests and burst experiments.

In application of SSYST on experiments it was found that the quality of the recalculation is strongly dependent on the quality of the boundary conditions. So for the recalculation of REBEKA experiment 8 % strain were calculated but the experiment shows 6-30 %. With the recalculation of LOC-11 experiment (PBF, Idaho) the calculation showed rod burst but in the experiment about 2 % were measured only. Parallel calculations to these problems with FRAP-T indicated similar results as SSYST. So we can conclude that the boundary conditions were not precise enough. This shows up an important problem in verifying those codes. Apparently the precision of data or the reconstruction of the situation in the experiment has greater variations in temperature for example, than is necessary to rise the creep rate for Zry from being almost zero to large values. This sensitivity showed up also in calculations for COSIMA experiment and in the runs made for the response surface of clad final strain after a LOCA. In these calculations for typical reactor fuel rods it turns out that large strains have a low probability. Important is that most of the deformation takes place in the refill phase and the clad freezes rapidly in the flooding phase. This is due to the increased inner gap and the increased outer heat transfer coefficient. In evaluating this result it should be kept in mind that the code does not take into account a super heated vapour which might be there in realistic situation.

7. Plans for the near future

The release of SSYST-2 and the development of a burst criterion for Zry cladding will be the most important tasks for the next future.

8. Relation with other Projects

This project is part of the major project PNS 06.01 of the Kernforschungszentrum Karlsruhe. It is supported by the other parts of the major project as well in the development of models as in their verification.

9. References

Borgwaldt, H., Meyder, R., Unger, H.:
Beiträge zum 1. Halbjahresbericht 1978 des Projektes Nukleare Sicherheit des Kernforschungszentrums Karlsruhe; KFK Bericht in Vorbereitung

Borgwaldt, H., Meyder, R., Unger, H.:
Beiträge zum 2. Halbjahresbericht 1978 des Projektes Nukleare Sicherheit des Kernforschungszentrums Karlsruhe; KFK Bericht im Vorbereitung

10. Degree of availability

Unrestricted distribution

Berichtszeitraum/Period 1.1.78 - 31.12.78	Klassifikation/Classification 1.3	Kennzeichen/Project Number 06.01.07 (PNS 4236)
Vorhaben/Project Title Untersuchungen zum Brennstabverhalten in der Blowdown-phase eines eines Kühlmittelverluststörfalles (COSIMA-Programm) Investigations of the Fuel-Rod-Behavior during the Blowdown-Phase of a Loss-of-Coolant Accident (COSIMA-Program)		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS) IRE/IT
Arbeitsbeginn/Initiated 1.7.1972	Arbeitsende/Completed 1980	Leiter des Vorhabens/Project Leader Dr.G.Class, IRE;K.Hain, IT
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

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 theoretical models (SSYST).

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

COSIMA experimental operation was continued during the period of reporting with 57 blowdown tests performed. All the envisaged types of fuel rod simulator, i.e. the simulators capable of ballooning containing Al_2O_3 and ThO_2 ring pellets, respectively, and the instrumented simulator for measurement of the heat transfer coefficients, have now undergone complete testing and have meanwhile been inserted in the main tests. In the main tests mostly a 2F cold leg rupture was simulated and the claddings were exposed to internal gas pressures of up to 124 bar.

Another central subject of activities consisted in the advancement of the special methods for non-contact measuring of the cladding tube temperature and for measurement of the transient two-phase mass flow.

Theoretical work was continued to recalculate the COSIMA tests by means of the RELAP4-002 (17) computer code.

6. Results

From the COSIMA main tests several types of testing for simulation of PWR primary loop ruptures on the cold and hot sides, accompanied or not by cladding tube deformation, as well as tests with the WUS (measurement of heat transfer coefficients) simulator have been made available for evaluation.

In direct comparison measurements the WUS simulator instrumented by thermocouples was used to check the accuracy of non-contact measurement of the cladding tube temperature by means of pyrometers. The accuracies of measurement of the pyrometers and of the thermocouples embedded in the cladding tube proved to be comparable. Work on the optimization of the true-mass flowmeter (TMFM) resulted in a further improvement of the accuracy and availability during measurement of transient two-phase mass flows. Independent of the steam quality and of the mass flow, the inaccuracy of measurement lies within $\pm$ 3 %, related to the nominal flow rate.

After some modifications of RELAP4-002 (17) satisfactory agreement was obtained between the measurement and the computation /1/. The following changes will be given as examples:

- The DNB criteria for pressures above 70 bar had to be corrected to higher heat fluxes.
- In case of stagnant flow a "free convection and radiation" mode of heat transfer was introduced for a high volume fraction of steam.
- The interpolation towards superheated steam of the heat transfer coefficients at high volume fraction of steam was made to depend on the steam quality; the application of the film boiling correlations was restricted to steam qualities of more than 0.15.
- The material data for water vapor were taken from MAPLIB.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	1.3	06.01.09 (PNS 4238)

Vorhaben/Project Title	
Untersuchungen zur Wechselwirkung zwischen aufblähenden Zircaloy-Hüllen und einsetzender Kernnotkühlung (REBEKA-Programm) Studies of the Interaction between Ballooning Zircaloy Cladding and the Emergency Core Cooling (REBEKA-Program)	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KfK Projekt Nukleare Sicherheit Institut für Reaktorbauelemente

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1973	1981	K. Wiehr
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. <u>General Aim</u>

The general aim of the project is the development of experimental information about the ballooning of zircaloy claddings during the refill and flooding phases of a hypothetical loss-of-coolant accident which is to verify and further developing the SSYST code system.

2. <u>Particular Objectives</u>

The deformation behavior of zircaloy claddings is studied under various transient boundary conditions in single rod and bundle experiments. For this purpose, full length fuel rod simulators with axial power profiles and arranged in bundles under approximately representative thermohydraulic flooding conditions are being used. The project serves the particular objectives outlined below:

- Assessment of the ballooning mechanism.
- Assessment of the influence of emergency core cooling on the ballooning.
- Studies of the thermal and mechanical interactions between adjacent rods during ballooning in a rod bundle.
- Generation of information about possible failure propagation.
- Studies of the extent and distribution of cooling channel blockages.

3. <u>Research Program</u>

The experimental program is carried out in a number of consecutive series of experiments starting from single rod experiments in a steam atmosphere and finishing with bundle flooding experiments of bundle assemblies including 7x7 rods (49 rod bundle).

All experiments are accompanied by basic experiments on separate effects and by theoretical studies.
The experiments are carried out as parameter experiments. The experimental parameters are based on the assumed power generation and the thermohydraulic boundary conditions accompanying a loss-of-coolant accident:

- rod power	10 - 25 W/cm
- axial power profile	step and cosine shaped, respectively
- internal rod pressure	20 - 100 bar
- systems pressure	1 - 4.5 bar
- cladding temperature at onset of flooding	700 - 900°C
- flooding rate (cold)	1 - 30 cm/s
- flooding water temperature	25°C - saturation.

4. Test Facilities

Test rigs are available for single rod experiments using shortened fuel rod simulators in air and steam atmospheres, respectively, and there is also a test loop for bundle experiments involving flooding of full length fuel rod simulators with axial power profiles.
In the single rod experiments on shortened fuel rod simulators it is possible to simulate temperature transients of the type calculated in the refilling and flooding phases of a loss-of-coolant accident for rods of various power ratings by combining the external heat transfer conditions with the rod power of the fuel rod simulator. - In the test loop for full length bundle experiments under flooding conditions a representative cladding temperature curve is generated largely automatically as a consequence of the good simulation quality of the fuel rod simulator and of representative emergency cooling conditions.

The deformation of the zircaloy cladding tubes is recorded by X-ray filming and evaluated. Some 130 measuring data (temperature, pressure, level, power, etc.) are recorded 10 times per second each with a cycle frequency of 10 kHz by means of a fast data acquisition system and the CALAS process computer.

5. Progress to Date

In the period under review most of the effort was concentrated upon the implementation of the following activities.

- Completion of bundle tests 2 and 3 with flooding in a 5x5 rod configuration with full length fuel rod simulators of an axial power profile to study the influence of cooling on deformation
- Evaluation of bundle test 2 and comparison with the results of bundle test 1
- Theoretical studies on the influence of cooling on temperature differences on the cladding circumference
- Investigations on the influence of surface mounted clad thermocouples on the accuracy of the temperature measurement and the rewetting behaviour of the cladding tubes
- Design work to extend the REBEKA test loop and data acquisition system for 7x7-rod bundles.

6. Results

Two full-length bundle tests with flooding (test 1 and 2) were evaluated to study the influence of cooling on deformation. The flooding from bottom was initiated at a different timing following a steam flow from top to bottom during the heat-up phase. In test 1 with relatively early flooding at 760°C maximum cladding temperature, most deformation occured during the reflooding phase under pronounced two-phase cooling. In test 2 with relatively late flooding at 860°C maximum cladding temperature, all deformation was generated during the heat-up phase under moderate cooling of steam flow from top to bottom.

In both tests an axial shift of the maximum strains was observed and is evidence that an axial cladding temperature profil exists. In test 2 this is a consequence of a superheating of the single-phase steam cooling during deformation in the heat-up phase. The axial deformation profile of test 1 in which the deformation occured under two-phase flow during the reflood-phase suggests that there is no thermodynamic equilibrium between steam and water during the reflooding phase and a substantial superheating of the steam which results in a axial cladding temperature difference between the grid spacers. Such axial temperature profiles promote a localization of maximum strains on the hot area and prevent axially extended ballooning.

In test 1 the distribution of the wall weakening of the zircaloy cladding tubes is in no way uniform and suggests a pronounced temperature nonuniformity on the cladding circumference with the consequence of relatively small strains and a small coolant channel blockage ratio of only 25%. In contrast to test 1 the bundle cross section of test 2 exhibits a relatively uniform circumferential wall weakening of the cladding tubes as a consequence of fairly uniform temperature distributions on the cladding circumference which resulted in a coolant channel blockage ratio of 60%.
This suggests that the increasing cooling efficiency during the reflooding phase enhances existing azimuthal temperature differences and contributes to a limitation of the circumferential strains and the resulting coolant channel blockage in a rod bundle. This experimental evidence has been proved by a theoretical analysis.
The surface mounted thermocouples used to measure the cladding temperature are not able to determine the true temperature owing to the fin cooling effect and the attachment on the surface. Comparisons between fin and embedded thermocouples of guard fuel rod simulators with Inconel claddings have shown that the temperature recorded during reflooding by a fin thermocouple is lower by up to 40°C.
Comparing the quench times indicated by surface mounted and embedded thermocouples of guard fuel rod simulators gives no evidence that the rewetting behaviour of the Inconel cladding is influenced by fin thermocouples.

7. Next Steps

- Further single rod tests
- Detailled evaluation of bundle tests 1, 2 and 3
- Performance of bundle test 4 comprising a passive cold control rod guide tube in the centre
- Extension of the REBEKA-test loop and the computer controlled data acquisition system for 7x7-bundle configurations
- Theoretical work

8. Relations with Other Projects

KfK: PNS 4231, 4235, 4236, 4237, 4239
KWU: RS 107, RS 36
USNRC: MRBT program

9. References

/1/ PNS-Halbjahresbericht 1977/2, KfK 2600, April 1978

/2/ K. Wiehr, F. Erbacher, H.J. Neitzel: Out-of-pile-Bündelversuche zum Brennstabverhalten in der Flutphase eines hypothetischen Kühlmittelverluststörfalles. Reaktortagung 1978, Hannover, 4.-7. April 1978

/3/ F. Erbacher, H.J. Neitzel, K. Wiehr: Studies on Zircaloy Fuel Clad Ballooning in a LOCA, Results of Burst Tests with Indirectly Heated Fuel Rod Simulators. 4. Conference on Zirconium in the Nuclear Industry, June 27-29, 1978, Stratford on Avon, England

/4/ PNS-Halbjahresbericht 1978/1 KfK 2700, September 1978

/5/ F. Erbacher: Verhalten der Brennelemente beim Kühlmittelverluststörfall und Wechselwirkung mit der Kernnotkühlung, KfK 2691, September 1978

/6/ I. Gaballah: Ein Beitrag zur theoretischen Untersuchung der Zweiphasenströmung mit Phasenwechsel in einem Kühlkanal eines LWR-Brennstab-Bündels beim Kühlmittelverluststörfall, KfK 2657, September 1978

/7/ I. Gaballah: Theoretische Untersuchungen zur Gasströmung in aufblähenden LWR-Brennstäben, KfK 2656, September 1978

/8/ F. Erbacher, H.J. Neitzel, K. Wiehr: Interaction between Thermohydraulics and Fuel Clad Ballooning in a LOCA, Results of REBEKA Multirod Burst Tests with Flooding, Sixth Water Reactor Safety Research Information Meeting, November 6-9, 1978 Gaithersburg, MD, USA

10. Availability of Reports

Available as KFK reports or conference reports.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	1.3	RS 185 A

Vorhaben/Project Title	
Parameteruntersuchungen über die Beeinflussung der Hüllrohre durch Nachbarstäbe beim Kühlmittelverluststörfall Investigations on the Influence of Neighbouring Fuel Rods during LOCA	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik B 221, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 11. 77	31. 3. 79	H.-J. Romeiser
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 78	

1. General Aim

The object of the program is to investigate under what conditions mechanical and thermal interaction between two neighbouring test fuel rods may occur, and, how the deformation behaviour may be affected.

2. Particular Objectives

The specific objective is indicated by the following special questions:

- Are the basic results of single rod tests with respect to burst pressure, burst temperature and burst strain modified by multirod test results concerning the possible influence of neighbouring rods?
- Does the contact of neighbouring rods have an influence on the time to burst, burst location, burst temperature and burst strain? Is there a propagation effect?
- What influence do the coolant conditions (normal convection/forced air) have on the deformation behaviour of the test fuel rods in the multirod geometry?
- What can be said concerning distribution of cladding deformation in multirod geometry?

These questions are supposed to be answered by tests with systematic variation of the parameters affecting the deformation behaviour of the fuel rods. These are essentially the rod internal pressure (effectively the pressure difference resulting from the respective fuel rod internal pressure and coolant pressure) and the test temperature.

These parameters can be adjusted separately for the two interior test rods, so that the following cases can be investigated:

- One pressurized test rod in combination with a neighbouring rod which suffers no deformation and which is subjected to a temperature level equal, higher or lower to that one of the test rod.
- Two pressurized rods which are subjected to the same or different internal pressure and/or temperature level.

The experiments will be run in air with normal convective cooling (for comparison with single rod tests) as well as with forced air cooling (influence of cooling).

The correlations for the analytic description of deformation and bursting, obtained from the single rod tests, will be compared with the results of the multirod tests and modified if necessary.

3. Research Program

To 3.1 Experimental Procedure

The multirod test assembly comprises 12 rods in a 3 x 4 matrix, in which the two central rods can be pressurized internally. Each rod contains an internal tungsten heater. The pressurization is performed separately for the two central rods. Temperature measurement is carried out by the means of thermocouples, which are distributed axially and radially in the bundle and spot-welded to the cladding surface.

The multirod assembly is constructed in that way, to allow the installed heaters to be reused for several more experiments. The test assembly is heated to a starting temperature of 350 °C and held at this temperature for about 30 minutes. In this time period both interior rods capable of deformation are pressurized. After achievment of temperature equilibrium the test assembly is heated at a rate of 10 - 20 % K/s

a) until burst of the test rod (transient test)
b) to a predetermined maximum temperature and held at this temperature level until one or both pressurized test rods have burst (creep test)

During the experiment the following parameters are measured and recorded:

- cladding temperature of the test rod
- temperature of the surrounding rods
- test rod internal pressure

The experiments are carried out with the following parameter ranges:

a) transient tests:

heating rate:	10 - 20 °K/s
internal pressure:	50 - 80 bar

b) creep test:

heating rate:	10 - 20 °K/s
internal pressure:	50 - 80 bar
max. temp.:	700 - 900 °C
holding time at max. temp:	≤ 90 s

3.2 Test evaluation

a) determination of the extent of deformation
b) determination of the cladding deformation at one or more cross-sections of the multirod assembly
c) examination and, if necessary, modification of the empirical correlations developed in the scope of RS 177 program on the description of deformation and burst behaviour
d) evaluation of the burst deformation.

4. Test Facilities

The test arrangement consists of the following:

a) An indirectly heated test assembly with 3 x 4 rods. The two interior rods can be pressurized with Helium.
b) Power supply for heating the test assembly
c) Facility for Helium pressure production
d) Control device, measurement equipments and pressure and temperature recording systems.

5. Progress to Date

To 3.1 Calculation of the air flow rate necessary to achieve the α-numbers (about 30 $W/cm^2 \cdot K$) characteristic for the refill phase.

Completion of a multiple rod test with two different pressurized test rods (50 and 65 bars). The test rods contained Al_2O_3-pellets as a loading and insulation material, whereas Al_2O_3-tubes have been used in previous rods. Assembly of an experimental arrangement for performance of two rod-internal pressure tests in a 3 x 4 multi rod geometry with forced air cooling:

- testing of a new sample holder for the adjustment of a defined constant gas inventory
- fabrication of the air shroud
- fabrication and testing of the supplementary heatable gas reservoir
- set up and testing of the air compressor
- fabrication of spacer grids and cladding samples
- assembly of the 3 x 4 multirod array.

Performance of heating experiments in 3 x 4 multirod geometry with forced air cooling, for achievement of the required α-numbers.

Improvement of the burst facilities by installation of supplementary water cooling for the sample holder and the electrical connections.

Fabrication of extended test rods and internal heaters to enable extension of heated length in the multirod tests. Fabrication of a correspondingly extended air shroud.

Completion of the first scoping test with forced air-cooling and with two test rods capable of deformation.

The following test conditions were established:

- transient test
- cold air shroud
- heating rate of about 10 ^{o}K/s
- internal pressure: P_1 = 65 bar
  P_2 = 50 bar
- α-number: about 30 W/cm^2 · K

6. Results

To 3.1 The most important results of the heat transfer calculation for a heated test section of about 500 mm length and with the normal rod-to-rod distance of 14.3 mm are listed in the following table. The hydraulic conditions used for the calculation are given too.

type of flow	laminar					turbular			
air velocity, m/s	0,5	1	2	3	5	10	20	30	50
Reynolds Number	193	386	772	1158	1930	3860	7720	11580	19300
Nusselt Number	4,21	4,65	5,34	5,89	6,75	14,4	26,53	36,82	54,85
α-Number W/m^{2}K	9,1	10,1	11,6	12,7	14,6	31,3	57,4	79,7	118,7

laminar: $Nu_o = \sqrt[3]{3,66^3 + 1,61^3\ Re \cdot Pr \cdot di/l}$

turbular: $Nu_o = \dfrac{\xi/8\ (Re-1000) \cdot Pr}{1 + 12,7 \sqrt{\xi/8}\,(Pr^{2/3} - 1)} \cdot \left[1 + \left(\dfrac{di}{l}\right)^{2/3}\right]$

The calculation indicated that, for an α-number of about 30 $W/cm^2 \cdot K$ an air velocity of at least 10 m/s is necessary.
Accordingly, turbulent flow is indicated with a calculated Reynold Nomber of about 3860.

With the new test rod holder the total gas inventory of the fuel rod can be maintained at about 40 cm^3. The additional volume existing outside the test rods and pressurizing capillary tubes can be heated up to 350 °C. The pressurizing capillary tubes are separately connected to the two pressurized test rods. With the spacer grids used, the test rods can be installed without damage to the cladding surface.

It has not yet been possible to evaluate the completed scoping test. The following preliminary findings can be started: In spite of a low α-number (about 30 $W/m^2 \cdot K$), a displacement of the maximum temperatures in the direction of the upper spacer grid was observed during the experiment. One burst location is situated directly below, the other directly above the upper spacer grid.

The magnitude of the burst deformation has not yet been determined; however, it has been observed by visual inspection that no contact of the neighbouring rods has taken place.

From this finding, one can conclude that the magnitude of deformation is smaller than 30 %.
Burst of the test rods accured at different times, according to their internal pressure.

To 3.2 The experimental results from the multirod test did not show the anticipated hot spots on the cladding outer surface of the pressurized rods, as a result of the Al_2O_3 pellet loading, expected to lead to a nonuniform temperature distribution on the cladding circumference.

This finding may be explained by the small diametral gap of the Al_2O_3 pellets and/or the low heating rate followed by the holding period at maximum temperature.

7. Next Steps

To 3.1 Performance of scoping tests with forced air cooling of the multirod test assembly with two test rods capable of ballooning.

The following test conditions will be established:

- transient test
- cold air shroud
- heating rate of about 10 K/s
- internal pressure (P_1/P_2): (50/65); (65/65) bar
- α-number: 30 $W/cm^2 \cdot K$

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	1.3	RS 309
Vorhaben/Project Title Innendruckversuche an Einzelstabproben zur experimentellen Absicherung des Brennstabverhaltens in der Notkühlanalyse Single Rod Internal Pressure Tests to Experimentally Verify Fuel Rod Performance for Emergency Cooling Analysis		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik B 221, Erlangen
Arbeitsbeginn/Initiated 1. 11. 77	Arbeitsende/Completed 31. 3. 80	Leiter des Vorhabens/Project Leader H.-J. Romeiser
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 78	Bewilligte Mittel/Funds

1. General Aim

Further experimental parameter investigations of clad swelling under approximated LOCA conditions in single rod geometry.

2. Particular Objectives

2.1 Investigation of the specific- and overall influence of test parameters on shape of ballooning and scattering of test data obtained up to now:

- Cladding tube geometry (wall thickness variation)
- axial and azimuthal temperature variations
- cladding tube properties (mechanical properties, structure and microstructure)
- superposition of influencing factors

2.2 Further investigations covering an extended refill phase are planned on the effects of:

- low heating rates
- prolonged holding times
- steam atmosphere
- pre-oxidation

The test results obtained are to verify over an extended parameter range the present empirical correlations of the deformation and burst behaviour analysis, and to improve knowledge of the natural scattering of results.

3. Research Program

Consistent with the key points, the program will be carried out in the following test groups. That this program does not overlap with the scope of tests planned under PNS (project nuclear safety) was confirmed with GfK.

3.1 Investigations of the reasons causing scattering of the deformation data

- influence of cladding-geometry
- influence of thermal eccentricity
- influence of cladding properties

3.2 Investigations on the "2nd peak" (refill phase) over an extended parameter range

- with various heating rates during transient burst tests
- with various holding times during creep-burst tests
- with various, preceeding thermal conditioning of the cladding corresponding to the blowdown phase of a LOCA by means of transient- and creep-burst tests.

3.3 Investigations of steam influence

Construction, assembly and test of an equipment for internal pressure tests (transient- and creep-burst tests) with directly heated cladding samples in steam atmosphere.

Performance of transient- and creep-burst tests to evaluate relationships during swelling and bursting of cladding tubes in steam.

Evaluation of the influence of cladding tube pre-oxidation on the deformation- and burst behaviour of cladding tubes. Determination of correlations existing between burst temperature, burst strain, internal pressure, heating rate and holding time by means of transient- and creep-burst tests in steam.

3.4 Investigations of combined effects

Transient- and creep-burst tests are planned to be performed on indirectly heated samples in steam with superimposed individual effects, such as steam atmosphere, cladding tube geometry, thermal eccentricity and cladding tube properties in order to verify the combined influence of these individual effects on the deformation and burst behaviour of cladding tubes.

3.5 Analytical Work

The models for the deformation and burst behaviour of the directly heated single rod will be checked by means of the test results and will be adjusted to the actual test conditions in order to achieve a more realistic description of the deformation and burst behaviour.

4. Experimental Facilities

Tests in air will be performed in the same test equipment as was used for RS 107, though modified to meet the specific test requirements. The device consists of burst equipment, heating transformer, programmable temperature control:

electrical requirements:	500 A/60 V
pressure range:	100 - 150 bar
length of test specimens:	50 cm

Tests in steam will be performed in a test device consisting of steam generator transformer, programmable temperature control and burst equipment.

steam generation:	5 kg /h max.
steam temp.	600 °C max.
steam pressure:	5 bar max.
sample length:	40 cm

5. Progress to Date

To 3.1 Experiments on the influence of the azimuthal temperature distribution

Two methods for the setting of azimuthal temperature differences on the cladding tube samples subjected to direct heating were tested:

a) Attachment of a series-connected additional heating rod parallel to the sample at a distance of 14.3 mm. By reducing the wall thickness of the additional heater, higher temperatures could be set on the additional heater while the amperage of the current remained the same. Three additional heating conductors with different reductions of the wall thickness were used (0.05, 0.15 and 0.25 mm).

b) The sample was heated on the one side with the additional heating rod and cooled with air on the opposite side.

Creep-burst tests with defined azimuthal temperature differences and a photographic recording of the deformation and burst process were performed. The following were selected as test parameters:

Maximum temperature	800 ^{o}C
Internal pressure at 350 ^{o}C	50 and 65 bar
Azimuthal temperature difference	40 ^{o}C to 100 ^{o}C.

The measuring and evaluation method for the determination of the time dependant strain was tested.

It was possible to improve the burst equipment by heating the additional heater separately (additional transformer) with its own temperature control.

Pre-oxidation of Zircaloy-4 cladding tube samples

For the preparation of pre-oxidized Zircaloy cladding tube samples the cladding tubes were oxidized in a tube furnace in an argon/oxygen atmosphere of approx. 70 to 80 vol% argon and 30 to 20 vol% oxygen at a temperature of 400 - 420 ^{o}C. Oxide layers of approx. 5 μm and 30 μm thickness are to be achieved. The preliminary determination of the thickness of the oxide layers will be met by the gravimetric method.

Establishment of an α'-structure in the Zircaloy cladding tubes

On Zircaloy tubes forming part of normal deliveries from manufacturers A and C, an α'-structure was built up by fast cooling from the β-phase range (approx. 1020 - 1050 ^{o}C). The cooling rate in the (α+β) phase range was approx. 60 - 80 ^{o}C/s.

Production of samples with α and recrystallized structures

Metallographic examinations of the heat-treated Zircaloy tubes were performed to characterize the structures produced.

Investigation of the effect of differences in the wall thickness on the azimuthal temperature profile

A computational estimate and verification tests were performed to determine the effect of the azimuthal differences in wall thickness of the cladding tubes (eccentricity) on the azimuthal temperature profile in the case of directly heated samples.

To 3.2 Transient burst tests with directly heated samples with low heating rates

Transient burst tests were performed on Zircaloy-4 cladding tubes made by manufacturers A and C under the following test conditions:
Direct heating, heating rate 5 K/s, internal pressure ~ 80 bar at 350 °C, free internal volume approx. 10 cm^3 (approx. 8 cm^3 sample, 2 cm^3 fittings), manufacturer "A" with raw material from suppliers No. 1 and No. 2, manufacturer "C" with raw material from supplier No. 1, loading as usual (ceramic pellets and rod), atmosphere inside/outside: air.

To 3.3 Steam burst equipment

A temperature correction for internal pressure tests in a steam atmosphere at different steam flow rates was determined with the aid of cladding tube temperature measurements. The temperature was measured with spot-welded Pt-Pt-10-Rh thermocouples. For improved heat insulation the sample was filled with Al_2O_3 powder. The tests were carried out with direct heating. The steam data are summarized below:

Test No.		1	2	3	4	5
Steam temperature	°C	Test in air without steam	150	150	150	150
Steam pressure in the autoclave	bar		4.7	4.4	4.3	3
Steam flow rate	l/h		7.5	10.5	13.5	19.0

To 3.4 Transient burst tests with internally heated samples

The equipment for the performance of tests with indirectly heated single rods was enlarged by a cylindrical mirror reflector (brass cylinder, hard chrome plated and metallized inside) and with an automatic unlocking device for the axial restraint equipment (solenoid valve and compressed-air cylinder).

The tests were started for the extended systematic investigation of the effect of the heating rate and internal pressure on the ballooning and burst behaviour of internally heated single rod samples with eccentric Al_2O_3 pellets, a thermal reflector and with axial restraint. Tests with the following test conditions and parameters were performed:

Heat-up rate of the internal heater	2, 10 and 28 °C/s
Internal pressure at 350 °C cladding tube temperature	50, 70, 90 and 110 bar
Free volume (cold)	33 cm^3
Atmosphere inside/outside	air/helium
Cladding tube dimensions:	
outside diameter	11.18 mm
wall thickness	0.68 mm

6. Results

To 3.1 Tests on the effect of the azimuthal temperature distribution

Differences in temperature across the cladding tube circumference of approx. 40 °C (at 800 °C) could be achieved by the additional heater series connected and situated parallel to the sample. ΔT_{az} is limited by the thermal conductivity and the removal of heat by convection and radiation. An increase in the temperature of the additional heater does, therefore, not lead to a further increase in the ΔT_{az} but to an increase in the temperature of the overall sample circumference.

It was by simultaneous heating of the one side and cooling of the opposite side that azimuthal differences in temperature exceeding 200 °C could be achieved.

At a temperature difference of 40 - 50 °C the sample came into contact with the additional heater arranged parallel to it during the ballooning process. This led to a short circuit at the point of contact between the heater and the sample and possibly distortion to misleading test result. For this reason the additional heater is heated in further tests by means of a second transformer independently of the sample. At the same time this measure allows a better control of the temperature of the additional heater and a better reproducibility in setting defined azimuthal temperature differences on the sample.

It was possible to evaluate the available photographs with a measuring accuracy of ± 10 μm.

The burst strain decreases as the temperature difference increases. With $\Delta T_{az} > 50$ °C the $\varepsilon_{\beta} < 40$ %.

Pre-oxidation of Zircaloy cladding tube samples

After an oxidation period of 900 h, oxide layer thicknesses of approx. 3 μm were achieved.
After an oxidation period of approx. 1950 h, oxide layers of approx. 6 μm thickness were achieved.
After an oxidation period of approx. 6000 h, oxide layers of approx. 21 μm thickness were achieved.

Production of samples with an α and recrystallized structures

Investigation of the effect of differences in the wall thickness on the azimuthal temperature profile.

The foll. values ΔT_{max} were estimated as upper limit for azimuthal temperature differences caused by eccentricity:

Δs in m	10	30	50	70	90	110
ΔT_{max} in K			14	19		
ΔT_{ε} in K	2	6	10	13	17	21

This list also contains values for the temperature difference ΔT_{ε} required to compensate the stress effect on the strain rate ($\dot{\varepsilon}/_{s=max} = \dot{\varepsilon}/_{s=min}$).

As ΔT_{max} and ΔT_{ε} are approximately of the same order of magnitude it should be verified whether such temperature differences actually occur.

For this purpose samples were heated up to 800 °C and the temperature was measured at the points of maximum and minimum wall thickness and at two points of medium wall thickness displaced by 90 °C, respectively.

Sample No.	Δs µm	$T/_{s=max} - T/_{s=min}$ K	$\Delta T_{measured}$ K	$\Delta T_{calculated}$ K
73438	60-70	2-6	8-9	approx. 18
73438/2	67	4-7	8-11	approx. 18
72290	6	5-7	13-15	

The temperature difference measured with repeated heat-up of a sample showed deviations up to 4 K max.

An unequivocal relation of temperature and wall thickness could not be ascertained.
In contrast, on the sample with low eccentricity, greater differences in temperature were measured than on the parallel sample with a relatively large eccentricity (67 µm). This leads to the conclusion that the temperature difference caused by a variation in wall thickness up to 70 µm is within the temperature measuring accuracy.

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Recrystallization of Zircaloy-4 cladding tube samples

The microsections of the recrystallized cladding tube samples show complete recrystallization without grain growth. The grain size is about Nos. 11 to 11 1/2 to ASTM E112.

To 3.2 Transient burst tests with directly heated samples

The average burst temperature and average burst strain for the samples of the different tube manufacturers with raw material from different suppliers are listed in the following table:

Tube manufacturer	Supplier of raw material	Average burst temperature °C	Average burst strain %
A	No. 1	823	110
A	No. 2	817	101
C	No. 1	832	136

To 3.3 Steam burst equipment

During the testing of the equipment the individual components were functioning properly. The thermo-couples spot-welded to the sample remained attached in their position during the period of steam flow. During a period of operation of 15 minutes, the steam pressure in the steam generator decreased from approx. 4.5 bar to 2.5 bar, the straight-through valve being fully open, and the saturated steam temperature dropped in accordance with the pressure. The superheater remained constant at the set temperature of 400 °C. The steam was condensed and intercepted. The flow rate was approx. 30 kg/h.

The average temperature gradient in the cladding tube wall in the temperature range of interest of 800 °C is 14 °C without steam and 17 °C with steam (flow rate of approx. 5 kg/h). These measurements have shown that temperature measurement in steam does not require any correction. The slight difference in temperature is caused by the increased power output necessary in the steam atmosphere. It is within the measuring accuracy of the temperature measurement.

The axial temperature profile is affected by the steam. A maximum temperature is formed which is displaced towards the steam outlet.

To 3.4 Transient burst tests with internally heated samples

The effect of the heating rate on the burst strain varies: Between 2 and 10 °C/s there is no significant effect. The burst strain determined is in the temperature range of 700 - 900 °C between 20 and 50 %. At 28 °C/s the burst strain is less and is between 10 and 30 %. The plot of the burst strain versus burst temperature shows a flat maximum for all three heating rates. At 2 and 10 °C/s this maximum is situated between 770 and 830 °C, at 28 °C/s between 830 and 860 °C.
The heating rate, at identical internal pressure, shows a clear influence on the burst temperature. Lower heating rates lead to lower burst temperatures than higher heating rates. Between 2 and 28 °C/s a difference in the burst temperature of approx. 100 °C arose in the pressure range of 50 to 90 bar.

The samples which burst in the temperature range of the α-phase exhibit the well known effect of bowing of the sample independently of the heating rate and in spite of axial restraint. Samples, however, which burst in the $(\alpha+\beta)$ phase range do not display any bowing.

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The occurrence of several small bulges across the heated length of the sample is characteristic for these tests. This was observed for all heating rates.

The results are in good agreement with the results of the ORNL tests.

7. Next Steps

To 3.1 Performance of directly heated creep-burst tests in air with an azimuthal temperature difference of up to approx. 150 °C. Recording of the deformation and burst process by photographic methods.

Performance of creep-burst tests with direct heating in air to determine the influence of the cladding tube geometry.
Oxidation of the samples will be continued until an oxide layer thickness of 30 µm.

To 3.3 Performance of transient burst tests in steam.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period 01.01. - 31.12.78	Klassifikation/Classification 1.3	Kennzeichen/Project Number 06.01.06 (PNS 4235.1)
Vorhaben/Project Title Untersuchungen zum mechanischen Verhalten von Zircaloy-Hüllrohrmaterial Investigations of the Mechanical Behavior of Zircaloy Cladding Material		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KfK Projekt Nukleare Sicherheit (PNS) IMF/II
Arbeitsbeginn/Initiated 01.01.1973	Arbeitsende/Completed 1979/1980	Leiter des Vorhabens/Project Leader M. Boček
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Investigation of the plastic behavior of Zircaloy-4 (Zry-4) during different reactor incidents, especially LOCA-typical temperature- and stress-transients and in a LOCA-typical environment.

2. Particular Objectives

Determination of a mechanical equation of state, containing all the parameters which influence the plastic strain.

3. Research Program

3.1 Tensile testing and creep testing of Zry-4 at high temperatures.

3.1.1 Influence of texture on plastic properties.

3.1.2 Influence of grain structure and phase composition on plastic properties.

3.1.3 Influence of ZrO_2-coating on plastic properties.

3.2 Burst tests with Zry-4 cladding.

3.2.1 Isothermal tests.

3.2.2 Transient tests.

3.2.3 Combined experiments (integral tests).

3.3 Examination on irradiated material.

3.4 Destructive postexamination of in-pile ballooned rods.

4. Experimental Facilities

Ad 3.1 Tensile testing will be performed in an INSTRON closed-loop machine.

Ad 3.2 For burst tests in vacuum tubes will be pressurized in a radiant furnace. For integral experiments (steam environment) cladding with internal heaters will be used.

5. Progress to Date

Ad 3.1 a) Tensile experiments on Zircaloy-4 with temperature and load ramps respectively.

b) Calculation of the life time for structures subjected to superimposed temperature-load ramps.

c) Tensile creep of Zircaloy-4 in an oxidizing atmosphere.

Ad 3.1.3a) Investigation of the influence of ZrO_2-layers on cavitation in Zircaloy.

b) The elastic behavior of pressurized ZrO_2/Zry-4 composite cladding (oxide layer on the outer surface).

Ad 3.2 a) Attempt to quantify the geometry of blown cladding tubes.

b) The determination of the time to rupture of Zry-cladding subjected to non steady loading conditions (failure criterion).

c) Isothermal and non-isothermal burst experiments in the "TUBA"-equipment.

Ad 3.4 Destructive postexamination of in-pile ballooned rods.

6. Results

Ad 3.1 a) In the 1st SAR(semiannual Report) 1978 [1] results are reported about calculations of the life time of structures plastically deformed by applying temperature or load ramps respectively. The calculations were based on the assumption that the life fraction rule (LFR) is obeyed. To check the validity of the assumption for the case of Zircaloy-4, appropriate tensile experiments have been performed. These investigations have shown that principally it is possible to predict on the basis of the LFR - at least for the upper α-Phase range of Zircaloy-4 - the failure temperature as well as the failure stress as functions of the loading conditions.

b) Assuming the validity of the LFR the life time was calculated for structures subjected to superimposed linear (in time) temperature and load ramps. As an essential result it follows therefrom that according to the ratio stress rate/temperature rate between two cases can be distinguished. For small stress rates the life time is determined solely by the temperature ramp whereas for high stress rates both the ramps influence the life time. In praxis this bears consequences for Zircaloy cladding subjected to LOCA-typical loading conditions.

c) After reconstruction of creep apparatus tensile creep experiments in air atmosphere were initiated on Zircaloy-4. The intention is to examine the influence of a steady oxidizing atmosphere on creep. These experiments complete a series of similar preceding examination in vacuum, the aim of which was to verify the constitutive models for inelastic behavior of Zircaloy.

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Ad 3.1.3 a) It was the aim of this investigation to explain the influence of ZrO_2-coatings upon the ductility of Zircaloy in the upper α-phase region (see 1st. SAR 1977, [2]). It is shown that the effect is due to a radial compression of the metallic substrate when the cylindrical ZrO_2/Zry-4 composite is loaded by tension. This stress component which suppresses the growth of cavities in grain boundaries is caused by the different contractile properties of both the components of the composite. The model describes quantitaitvely the influence of cracks in the oxide layer upon the decisive radial stress component.

b) In correlation to the Point a) above the question arose whether the increase of ductility observed on Zircaloy cladding with oxide layers is due to the same effect as responsible for the increase of ductility of tensile specimens. For this purpose calculations were performed showing that a compressive radial stress component in the Zircaloy can be generated when a tubular ZrO_2/Zry-4 composite is internally pressurized. Presently investigations are on the way to determine the influence of elastic constants, cladding geometry and oxide thickness upon the stress components in the composite.

Ad 3.2 a) Using an integral mean mean value $\bar{\varepsilon}_\Phi$ for the axial distribution of the tangential strain together with the maximum value for the tangential strain $\varepsilon_{\Phi,max}$ a value W_Z is defined in the way, that

$$W_Z = 1 - \frac{\bar{\varepsilon}_\Phi}{\varepsilon_{\Phi,max}} \geq 0 \qquad (1)$$

Two boundary values exist for W_Z:
$W_Z \to 0$ for an axially uniform tangential strain distribution and $W_Z \to 1$ for very localized balloons. W_Z-values have been computed for Zry-4 cladding tubes blown in the FR-2 loop. These are compared to the actual deformation conditions. This indicates the possibility of using this procedure to predict the resulting "failure type" of blown cladding.

b) Using the life fraction rule in the infinitesimal form, relations have been derived for the burst temperature and bust stress of material deforming under monotoneous temperature or stress ramp respectively. From there it follows for a temperature transient creep test at constant load F

$$\frac{P \cdot c \cdot \tau_o}{T_o} + 1 = \left[\left(\frac{\tilde{T}_B}{T_o} \right)^2 e^{P(1-T_o/\tilde{T}_B)} \right]_{F,c} \quad \text{for } T_o \neq \tilde{T}_B \qquad (2)$$

or for a stress transient creep test at constant temperature T

$$\tilde{\sigma}_B = \{ \sigma_o^{\,n} \left[b\tau_o \, (n+1) + \sigma_o \right] \}^{\frac{1}{1+n}}_{T,b} \qquad (3)$$

the symbols represent: $\tilde{T}_B$, $\tilde{\sigma}_B$ the rupture temperature and the rupture stress resp. for the corresponding transient tests; T_o, σ_o the starting temperature and the starting stress resp.; τ_o the time to rupture at T_o and σ_o in the ISO-test; $c = {}^{dT}/dt$, $b = {}^{d\sigma}/dt$ the heating- and stressing rate resp. (in the eq. (2) and eq. (3) resp. both are considered as constants); n the stress exponent; P a parameter derived from ISO-stress rupture test (Larson-Miller Parameter). The corresponding rupture times are

$$\tilde{\tau}_{F,c} = \frac{\tilde{T}_B - T_o}{c} \quad \text{and} \quad \tilde{\tau}_{T,b} = \frac{\tilde{\sigma}_B - \sigma_o}{b}$$

The comparison of experimental values with those obtained from the eq. (2) and eq. (3) resp. leads to the conclusion, that the model assumptions are in the present case essentially obeyed. Thus not the strain determines time to fracture but the stress and temperature.

c) First burst tests have been performed in the "TUBA"-equipment. These tubes can be blown in a gradient free temperature field. Some of the results are allready included in [1] (see section 2).

Ad 3.4 The test specimen A 2.1, A 2.3 and B 1.2, which had not been pre-irradiated, from in-pile-experiments in the FR-2 loop were subject to a destructive postexamination. The aims of this examination are the estimation of the cladding temperature on the basis of the structural appearance, the measurements of the circumferential strain, the change in wall-thickness and the change in area of the cladding cross sections on exposed parts of the rod.

In additon to this, mikrohardness measurements over the cross section had been made, to determine the oxygen uptake of the Zry-cladding material during the transient.

7. Next Steps

Ad 3.1 Tensile experiments for superimposed ramp loading conditions.

Ad 3.1.3 Examination of the cavitation behavior of Zircaloy and the fracture morphology of oxide layers for internally pressurized ZrO_2/Zry-4 composite cladding.

Ad 3.2 Examination of dependencies of W_Z upon test conditions. Burst test to verify failure criteria. Restart of burst experiments in the "TUBA" equipment. Evaluation of burst experiments performed in the "FABIOLA" equipment. Comparison of deformation data with those for model calculations.

Ad 3.3 Evaluation of mechanical recovery experiments performed on tensile specimens irradiated in the FR-2.

Ad 3.4 Postexaminations of in-pile ballooned cladding.

8. Relation with other Projects

PNS 4235.2 (06.01.06/02A)
PNS 4235.3 (06.01.06/03A)
PNS 4235.4 (06.01.06/04A, 06.01.06/05A)
RS 107

9. References

[1] M. Boček et al. in 1st. PNS-Sem.-Annual Report 1978 (German with English abstracts), KfK-2700, Nov. 1978

[2] M. Boček et al. in 1st PNS-Sem.-Annual Report 1977 (German with English abstracts), KfK-2500, Dec. 1977, P. 270

10. Degree of Availiability of the Reports

Unrestricted distribution.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78-31.12.78	1.3	06.01.06(PNS 4235.2)
Vorhaben/Project Title		**Land/Country**
Untersuchung zur Hochtemperatur-Wasserdampf-Oxidation an Zircaloy-Hüllrohrmaterial		FRG
		Fördernde Institution/Sponsor
		BMFT
		Auftragnehmer/Contractor
Investigation of the High Temperature Steam Oxidation of Zircaloy Cladding Tubes		Kernforschungszentrum Karlsruhe PNS/IMF 2
Arbeitsbeginn/Initiated	**Arbeitsende/Completed**	**Leiter des Vorhabens/Project Leader**
1.1.1973	1979/80	Dr. S. Leistikow
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

Cladding Material Behaviour under Accident Conditions

2. Particular Objectives

Investigation of the High Temperature Steam Oxidation of Zircaloy Cladding Tubes.

3. Research Program

Study of Zircaloy 4/Steam Oxidation Kinetics and of Oxidation Related Change in Mechanical Properties.

4. Experimental Facilities

Experimental set-ups for isothermal and temperature-transient oxidation reactions. Facilities for isothermal and temperature- and pressure-transient stress-rupture testing.

5. Progress to Date

Final isothermal and temperature-transient testing. Further post-test evaluation and documentation. Isothermal/isobaric creep-rupture testing of capsules in argon and steam (unoxidized and preoxidized condition). Temperature-transient/isobaric and temperature-/pressure-transient testing in steam according to LOCA-transients.

6. Results

The calculation and comparison of the heat production caused by the oxidation reaction $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$ at short isothermal exposures (900-1200 $^\circ$C) and by decay heat showed that during a LOCA oxidation produces only at the very first beginning a significant part of the total amount of heat. Its proportion is quickly diminishing. However, LOCA typical transients (because of their reduced time-at-temperature), preformed oxide scales, and the fact that only 75 % of the oxygen is reacting to ZrO_2 (the other part is dissolved in the metallic matrix) are the reasons that in fact heat production is still overestimated.

Zircaloy 4 cladding tubes of Oak Ridge National Laboratory (ORNL) were tested in steam at 1000°C during 1 - 90 min to compare the applied test methods with ours and to explain minor divergencies in the proportion of ZrO_2/α-layers as function of exposure time. Also for this material, an intermediate decrease of the proportion of the ZrO_2/α-Zr(O) layer thicknesses was found, followed by an drastic increase when the breakaway of the ZrO_2 scale occurred. Other reasons for small divergencies were evaluated, for instance temperature and time measurements, metallographic evaluation of layer thicknesses, consequences of one-sided (ORNL) and double-sided oxidation.

Other experiments were directed towards the reproduction of the so-called anomalous oxidation effect caused by the hysteresis of the monoclinic to tetragonal ZrO_2 - phase transformation and its feed-back on oxidation kinetics. During temperature transients this effect was reproduced qualitatively and-in case that a reduced gain of oxygen was observed-explained by the predominant existence of the monoclinic oxide phase.

Isothermal/isobaric capsule testing was continued and the creep-rupture functions were measured for 600, 650, 700, 750 and 850°C in argon and steam. In addition preoxidized capsules were tested in steam. The total number of creep-rupture experiments performed during the last years under isothermal/isobaric conditions now sums up to about 420. These tests showed below 800°C a moderate prolongation of time-to-rupture if the tests were performed in steam (or after preoxidation in steam) instead of argon. Also slightly reduced maximum circumferential strain could be measured.

At 800°C, 32-71 bar five creep curves were measured showing the transition to tertiary creep at $\varepsilon \leq 10$ % and maximum circumferential elongations after rupture of $\varepsilon_R = 70-100$ %. The specimens were evaluated by metallography and scanning electron-microscopy in the ruptured and nonruptured condition. It could be confirmed that there is a pressure dependence of the superficial oxide crack pattern: at equal strain high internal pressure caused the formation of a high crack density of small width, at lower pressure a low density of broad cracks could be observed.

The stress-rupture function of Zircaloy 4 at 1000°C in air showed - that compared to the behaviour in argon - no strength increasing effect was exerted by the quickly growing, but frequently cracking oxide scale (consisting of ZrO_2 and ZrN). Also ductility was reduced slightly.

Temperature-transient/isobaric tests were performed for testing - at a given internal pressure - the influence of various heat rates and of the blowdown-peak on time and temperature at rupture. During these tests no remarkable influence of the heating rate (5 and 10°/s) and of the blowdown-peak on temperature-at-rupture

950°C could be detected.

7. Next Steps

Metallographic evaluation of temperature-transient oxidation of preoxidized tube sections. Isothermal/isobaric creep-rupture tests at 750 - 900°C, 50 - 120 bar. Creep tests at 800 and 1000°C in air and comparison with those in steam. Temperatur-transient/isobaric creep-rupture tests (ramp tests) at higher heating rates (10 - 30°/s).

8. Relation to other Programs

Other programs within KfK-PNS, KWU, NRC.

9. References

1) Semi-Annual-Rep. KfK-PNS 1/78 and 2/78.

2) S. Leistikow, G. Schanz u. H. v. Berg
Kinetik und Morphologie der isothermen Dampf-Oxidation von Zircaloy 4 bei 700 - 1300°C
KfK 2587 (1978)

3) S. Leistikow u. R. Kraft
Kriech-Berst-Untersuchungen zum Kühlmittelverlust-Störfallverhalten von Zircaloy 4-Hüllrohren in Argon und Wasserdampf
Proc. Reaktortagung Hannover April 1978, S. 549-552

10. Degree of Availibility of the Reports

Part of open literature

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	1.3	06.01.08 (PNS 4237)
Vorhaben/Project Title Untersuchungen zum Brennstabverhalten in der 2. Aufheizphase eines Kühlmittelverluststörfalles - In-pile-Versuche mit Einzelstäben im DK-Loop des FR2 Investigations on Fuel Rod Behavior in the 2nd Heatup Phase of a LOCA In-pile Experiments with Single Rods in the DK Loop of the FR2 Reactor		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KfK - Projekt Nukleare Sicherheit Abt. Ingenieurtechnik
Arbeitsbeginn/Initiated: 1972	Arbeitsende/Completed: 1981	Leiter des Vorhabens/Project Leader: B. Räpple/E.Karb
Stand der Arbeiten/Status: Continuing	Berichtsdatum/Last Updating: December 1978	Bewilligte Mittel/Funds

1. General Aim

Investigation of the behavior of LWR fuel rods during a loss-of-coolant accident (LOCA). Provision of experimental data for the verification of computer codes.

2. Particular Objectives

Experimental investigation of the influence of a nuclear environment on the mechanisms of fuel failure. Performance of transient in-pile tests with single rods in the DK loop of the FR2 reactor, simulating the cladding temperature history of the second heatup phase of a LOCA.

3. Research Program

A total of approx. 45 nuclear tests are planned with unirradiated as well as with previously irradiated fuel rods. Irradiation in the FR2 reactor, steps of burnup: 0/ 2,500/ 5,000/ 10,000 / 20,000/ 35,000 MWd/t_U. Range of internal pressures: 25 to 120 bar.

25 Reference tests with electrically heated fuel rod simulators (BSS).

Both nuclear and reference tests are performed with the test rods in the in-pile section of the FR2 DK loop.

4. Experimental Facilities, Computer Codes

4.1 Test Facilities

4.1.1 Test Loop

The DK loop is operated with superheated steam of 60 bar, at 300 - 350 ^{o}C in the test section, and with a mass flow of 120 kg/h. The in-pile test section is located within a pressure tube in a vertical test hole of the FR2 core. The test transient is initiated by interrupting the coolant flow and depressurizing the coolant channel. When the desired cladding temperature is reached, the specimen power is reduced by reactor scram or interruption of the power supply respectively.

A special waste-gas system is available for the fission products escaping from the burst nuclear rod; it decontaminates the waste gases from volatile halogens and retards noble gases; solid and liquid isotopes are retained by filters.

4.1.2 Preirradiation and test rod assembling

The test rods are irradiated in the FR2 up to the desired target burnup in instrumented 6-rod bundles similar to FR2 fuel elements. Instrumentation provides data for determination of burnup and for safety monitoring.

Assembling of the irradiated test samples, in particular the spot welding of thermocouples onto the cladding is done remotely in the hot cell within the FR2 containment, using a device designed especially for this purpose.

4.1.3 Post-Test Examination PTE

Upon completion of the test the specimen is subject to examination in the neutron radiography facility (NERA) of the FR2. After decoupling from the sample support rod, the test rod is transported to the Hot Cells for nondestructive and destructive PTE.

4.2 Computer Programs

Comparative calculations of the thermal conditions during pre-irradiation in the FR2 reactor: Codes CARO (KWU) and COMETHE.

For pre-test calculations: Special versions of the codes RELAP, WALHYD (IKE) and STATI (KfK), which model the particular conditions of the test section for the nuclearly as well as for the electrically heated rods.
Post-test calculations: SSYST.

5. Progress to Date

5.1 The experimental program was continued according to schedule with a total of 11 nuclear tests:

With 5 rods each of series G 1 and G 2/3, all irradiated to 35,000 MWd/t_U, the tests with high burnup fuel were completed.

One nuclear test with an unirradiated rod, B 1.7, was performed.

5.2 A total of 5 tests with electrically heated simulators were completed with various objectives.

5.3 The first in-pile burst test with an electrically heated simulator, BSS 12, was made successfully.

5.4 The in-pile section of the test loop, which had been in the reactor core since 1969 had to be replaced when reaching its maximum tolerable irradiation dose. After a successfull trial with simulator BSS 14 the regular operation was resumed.

6. Results

6.1 Burst pressures and burst temperatures found during the test series G 1 and G 2/3 (burnup 35,000 MWd/t_U) did not show an influence of the nuclear parameters. Like the failure data of the B tests (unirradiated rods) and of the F tests (20,000 MWd/t_U) the G data lie well within the scattering band of the burst pressure and temperatures obtained out-of-pile by various investigators.

As experienced with the F rods, neutrografies of the G rods show that the fuel pellets, already cracked during previous steady-state irradiation, disintegrated during the transient or during subsequent handling at those rod sections where major clad lifting or ballooning occured.

6.2 Measured with 4 TC's at the same axial location, 5 cm below the upper edge of the fuel stack, and 90 ^{o} apart the maximum circumferential temperature variation was found to be approx. 40 K (steady-state as well as transient).

6.3 The in-pile burst test with BSS 12 confirmed the selected design of the electrically heated simulator. The burst data (54,2 bar, 842 ^{o}C) as well as size and shape of the balloon are similar to those of the nuclear tests.

6.4 Several BSS tests were used to calibrate an improved TC attachement method. Spot welding the 1 mm OD thermocouples to the cladding at the cylindrical part of the platinum jacket resulted in a reduction of the measuring error from previous 75 $\pm$ 35 K to now 10 $\pm$ 10 K.

6.5 An analysis concerning the proper plenum size for short length test rods was completed. It shows that best simulation is obtained with a test rod plenum having approximately the same volume as a full length standard fuel rod (internal report 9.8).

6.6 Post-test examination of the test samples continued. Out of 26 rods transferred to the hot cells 13 are examined fully, nondestructive testing of 9 rods was completed.

6.7 Six internal reports were distributed (besides contributions to semi-annual and annual reports) documenting test data and PTE results (9.7, 9.9).
An invited paper was presented at the 6th Water Reactor Safety

Research Information Meeting of the USNRC (9.6).

7. Plans for the Near Future (1979)

7.1 FR2 irradiation of rods for series C (target burnup 2,500 MWd/t_U) and series E (10,000 MWd/t_U).

7.2 Performance of test series C and of approx. 4 Reference Tests with electrically heated simulators.

7.3 Hot Cell post-test examination and evaluation of the results will be continued.

8. Relation with other Projects

KfK: PNS 06.01.04 (4231), 06.01.05 (4233), 06.01.06 (4235)
06.01.07 (4236), 06.01.09 (4238), 06.01.10 (4239)

USNRC: PBF-Program, MRBT-Program

9. References

/1/ PNS semi-annual report KfK 2600, 1977/2 April 78 pp. 87-98 / 373-382

/2/ PNS semi-annual report KfK 2700, 1978/1

/3/ IT annual report KfK 2595, 1977, pp. 15-17

/4/ E. Karb: Ergebnisse von In-pile-Experimenten im FR2 zum Brennstabverhalten bei Kühlmittelverluststörfällen.
In KfK 2570, pp. 195-217, Dez. 1977

/5/ E. Karb: Results of the FR2 Nuclear Tests on the Behavior of Zircaloy Clad Fuel Rods. 6th WRSR Inf. Mtg., Nov. 1978, Gaithersburg, Md, USA

10. Degree of Availability

Unrestricted Distribution

147-1-04/4163-2 + 3		1-3
TITRE Expériences de dépressurisation et de renoyage d'une grappe combustible PWR : Programme PHEBUS		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) Experiments on depressurization and reflooding of a PWR fuel assembly : PHEBUS project .		Organisme exécuteur CEA/DSN/SRS/SES Responsable
Date de démarrage 01/01/76	Etat actuel en cours	Scientifiques
Date d'achèvement 31/12/83	Dernière mise à jour 18/12/78	

1. OBJECTIF GENERAL

Le programme PHEBUS se propose 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.

II - OBJECTIFS PARTICULIERS

A - Réalisation d'une installation d'expérimentation permettant de :

- reproduire les conditions d'environnement d'un réacteur PWR pour une grappe expérimentale d'une taille de 25 crayons.

- simuler les différentes conditions de dépressurisation étudiées dans l'analyse de sûreté d'un réacteur PWR.

- simuler les différents types d'injection de sécurité qu'il est prévu d'utiliser à l'occasion d'un tel accident.

B - Programme expérimental, comprenant :

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

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

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

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

C - Interprétation des résultats

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

III - INSTALLATION EXPERIMENTALE

L'installation comporte :

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

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

  - le circuit d'eau déminéralisée qui refroidit le coeur à partir d'une réserve de 500 m^3 servant de volant thermique et permettant de conduire un essai à pleine puissance pendant 20 à 30 minutes.

- une boucle d'essai avec :

  - une cellule en pile dans l'axe vertical du coeur, contenant la perche d'essai qui abrite la grappe de crayons combustibles à tester, de 0,80 m de longueur active.

  - un ensemble de circuits hors pile contenus dans un caisson de 450 m^3 pouvant tenir 1,5 bars de surpression et rassemblant les fonctions de :
    . maintien des conditions nominales de fonctionnement d'un PWR.
    . déclenchement de la dépressurisation (brèche de dimension et de position variables).
    . mise en oeuvre de l'injection de secours, dont on peut faire varier certains paramètres.
    . mesure des paramètres, et prélèvement d'échantillons.

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

IV - PROGRAMME D'ESSAIS

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

- la catégorie I se compose des paramètres que l'on fera varier indépendamment : la taille de la brèche, sa localisation, la puissance linéique maximale des crayons combustibles et leur pression interne.
- la catégorie II se compose des paramètres qui seront étudiés en faisant varier un seul paramètre de la catégorie I, à savoir : la pression d'injection, le débit d'injection, la température de l'eau d'injection.
- enfin, des essais réalisés dans des conditions bien précises pour étudier des phénomènes particuliers et notamment les phénomènes aux limites concernant l'intégrité des éléments combustibles seront appelés essais de la catégorie III : l'influence des points d'injection, le comportement des crayons combustibles dans les conditions limites permises par les critères et dans la mesure du possible, l'état des crayons en fin de dépressurisation.

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

V - ETAT D'AVANCEMENT

Le réacteur a divergé le 9 août 1978.

Les essais neutroniques menés par le SEN se sont terminés le 1/12/78.

Suite aux résultats des premiers essais d'ensemble, il a été nécessaire de réaliser des modifications importantes sur les tuyauteries HP du circuit d'essai entre juillet et fin novembre 1978.

Après cette modification, les essais à froid à 270 bars ont eu lieu le 6/12/78. Ils se poursuivront à chaud jusqu'à la fin février 1979.

Les essais en puissance sont liés à une autorisation du SCSIN. Ils devraient avoir lieu jusqu'à 10 MW à partir de mars 1979 avec vérification du fonctionnement général sur perche prototype monocrayon.

La montée en puissance jusqu'à la puissance nominale avec perche prototype 25 crayons se situerait alors début juin 1979.

La campagne d'essais du programme commencerait au début du 3e trimestre 1979.

- Les calculs préliminaires permettant de préciser le programme des essais sont en cours au moyen des codes RELAP, CERES, FRAPT, FLIRA.

Le détail des programmes est en cours d'élaboration et d'étude par la commission ad hoc. Les cinq premiers essais sont présentement définis.

VI - RELATIONS AVEC D'AUTRES ETUDES

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

Cette étude est liée à l'étude 147-1-01.

VII - DOCUMENTS DE REFERENCE

DSN/SETS 91-73 du 21.3.73 Programme Phébus.
Plaquette Phébus IPSN-DSN 1977 (en cours de refonte)
DSN/SES 76-78 Programme de démarrage.

147-1-05		1-3
TITRE INTERPRETATION DES ESSAIS ET CALCULS RELATIFS AU PROGRAMME PHEBUS		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) TESTS INTERPRETATION AND CALCULATION RELATED TO PHEBUS PROJECT		Organisme exécuteur CEA-DSN/SEAREL Responsable
Date de démarrage 1.07.78	Etat actuel en cours	Scientifiques
Date d'achèvement 31.12.85	Dernière mise à jour 1.1979	

1. OBJECTIF GENERAL

- Mise à jour et préparation des moyens de calcul .
- Calculs prévisionnels et interprétation des essais au fur et à mesure de leur réalisation .

2. OBJECTIFS PARTICULIERS

2.1. - MISE AU POINT DES MOYENS DE CALCUL

Il s'agit de rendre opérationnels, au fur et à mesure de leur disponibilité, les codes qui peuvent permettre l'interprétation des essais PHEBUS .

Les calculs seront réalisés d'abord avec les codes de première génération de type RELAP, FRAP, puis avec les codes de deuxième génération lorsque ceux-ci seront disponibles et suffisamment validés par ailleurs .

2.2. - CALCULS PREVISIONNELS

Ces calculs pourront être faits :

- soit pour définir les paramètres de l'essai ;
- soit, les paramètres de l'essai étant figés, pour aider à sa réalisation (tout essai devra être précédé de ce calcul prévisionnel final) .

2.3. - INTERPRETATION DES ESSAIS

Cette interprétation doit comprendre :

- une comparaison des calculs prévisionnels et des résultats expérimentaux,
- la réalisation de nouveaux calculs pour ajuster les modèles sur les résulats expérimentaux,
- une analyse des résultats de comparaison calcul-expérience .

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

. Réacteur PHEBUS - voir fiche 147-1-04

4. ETAT D'AVANCEMENT

Cette étude fait suite à l'étude intitulée:

- Programme PHEBUS: moyens de calcul et calculs préliminaires (fiche 147-1-01)

Dans l'étude précédente de nombreux calculs préliminaires ont été effectués pour la phase de dépressurisation avec le code RELAP 4 mod 3 . Des calculs de renoyage ont été faits également avec RELAP FLOOD . L'ensemble de ces calculs constitue une base importante dans le travail de mise en place des outils d'interprétation des essais PHEBUS .

5. PROCHAINES ETAPES

5.1. MISE AU POINT DES MOYENS DE CALCUL

Cette mise au point sera faite dans un premier temps à partir de la version RELAP 4 mod 5 pour les phase de dépressurisation et de remplissage. Dans un deuxième temps on mettra au point le calcul de la phase de renoyage avec la version RELAP 4 mod 6. C'est cette version qui sera enfin utilisée pour le calcul de l'ensemble des phases de l'essai .

Cette mise au point comprendra l'utilisation de tous les résultats d'essais effectués hors pile : essais de perte de charge, de débit brèche, de renoyage ERSEC ... .

Les différentes étapes de cette mise au point sont les suivantes:

1. Mise à jour du jeu de données
2. Mise au point de l'initialisation du calcul
3. Calcul brèche
4. Mise au point du calcul dépressurisation
5. Mise au point du calcul de remplissage
6. Mise au point du calcul de renoyage

7. Calcul de la pression du réservoir de décharge (à l'aide du code PAREO)
8. Calcul combustible
9. Calcul d'une caractéristique type LOCA .

5.2. CALCULS PREVISIONNELS

Ces calculs seront faits en fonction des besoins liés au déroulement du programme expérimental . Les essais actuellement prévus, et pour lesquels des calculs prévisionnels sont programmés, sont:

. essais hydrauliques et prototypes 1 crayon)
. essais hydrauliques et prototypes 25 crayons) 1er semestre 1979
. premiers essais 1 crayon du programme PHEBUS : 2ème semestre 1979

5.3. INTERPRETATION

Des interprétations simplifiées sont faites pour les essais prototype du fait des problèmes d'instrumentation . Les interprétations approfondies commenceront avec les premiers essais 1 crayon du programme PHEBUS (2ème semestre 1979) .

6. RELATIONS AVEC D'AUTRES ETUDES

. Programme expérimental PHEBUS (voir fiche 147-1-04)
. Etudes combustibles (145-2)

	CLASSIFICATION: 1.3
TITLE (ORIGINAL LANGUAGE): Ribagnamento delle barre di combustibile durante la fase di E.C.C.S.	COUNTRY: ITALY SPONSOR: C.N.R.
TITLE (ENGLISH LANGUAGE): Rewetting of fuel rods during E.C.C.S.	ORGANISATION: CALABRIA UNIVERSITY PROJECT LEADER: V. MARINELLI
INITIATED: January 1977	COMPLETED: December 1980 STATUS: In progress	LAST UPDATING: June 1979	SCIENTISTS: G. OLIVETI, A. SABATO

1.- General aim
Study of cooling of fuel rods during the E.C.C.S.

2.- Particular objectives
Optimization of engineering correlations and models to predict the thermal behaviour of rods during E.C.C.S., development of a computer code for rewetting calculations under the reflooding mode of E.C.C.S.

3.- Experimental facilities and programme
Experimental apparatus for bottom flooding in rod-annular geometry at low pressure, starting from different levels of temperature, and, successively, experiments of spray cooling.

4.- Project status
A survey of literature on the subject has been done and a preliminary computer code has been developed, in order to see the present capability of predictions against available literature data of flooding.
Additionally a study has been done on the interaction of the vapour phase and the liquid phase in countercurrent flows during sprays, and has been checked against experimental data on a 9 rods bundle, giving encouraging agreement. This study has been verified also with experimental data obtained in round pipes and ducts of different shapes.

5.- Next steps
The Experimental setup is in construction and the models are in progress.

6.- Relation to other projects and Codes-None.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Rewetting of fuel rods during E.C.C.S.	1.3

7.- Reference documents

A. Sabato
Tesi di Laurea: Modelli di ribagnamento per le barrette di combustibile di LWR nel caso di incidente
Jan. 1978, Politecnico di Torino.

V. Marinelli - G. Oliveti - A. Sabato
Counter current flow of air and water in a 9-rod bundle
Atto del Dipartimento 23, Dpt. of Mechanical Engineering, Università della Calabria, June 1978.

V. Marinelli - G. Oliveti - A. Sabato
Calculations of countcurrent flow of liquid and gas in the flooding regime
Atto del Dipartimento 28, Maggio 1979, presented also: European two phase flow group meeting at ISPRA, June 5-8 1979.

8.- Degree of availability
Full availability.

	CLASSIFICATION: 1.3
TITLE (ORIGINAL LANGUAGE): Investigation LWR fuel behaviour under LOCA conditions	COUNTRY: Italy
	SPONSOR: Italian Government
TITLE (ENGLISH LANGUAGE):	ORGANISATION: JRC Ispra
	PROJECT LEADER: (°)
INITIATED: June 1975 / COMPLETED: 1982	SCIENTISTS:
STATUS: In progress / LAST UPDATING: July 1979	

(°) JRC Ispra-ESSOR Division
CNEN Italy - Thermal Reactor Department

General Aim

In-pile investigation of LWR fuel behaviour (rods and bundles) in simulated LOCA conditions using an experimental loop, named SARA, being built for the ESSOR reactor. The primary objectives of the program are:
- Rod, especially clad, deformations during wide range of LOCAs.
- Interactions caused by deformations:
  - Between rods
  - Between deformations and coolant
  - With possible rod failure propagation.
- Bundle coolability and thermal response during reflooding
  - S/C blocking due to ballooning
  - S/C blocking due to brittle fragmentation.

Particular Objectives

- Detailed design, fabrication and installation of the loop.
- Specification of the LOCA simulation test program.
- Execution of the test program in its SUPER-SARA version (SUPER-SARA test program) as a proposed program of the E.E.C. (see point 5.).

Experimental Facilities and Program

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 thermo-hydraulic performance of the loop in the nuclear mode will be guaranteed by carrying out prior tests in an electrically heated parallel twin test section, using the same basic loop. The loop is designed to cover a wide LOCA parameter range

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Investigation LWR fuel behaviour under LOCA conditions	1.3

and to drive PWR rod bundle assemblies of up to 36 rods (17x17 type) of 2m lenght (2.3 MW heat rejection). This length is considered quite sufficient for the correct simulation of reflooding heat transfer and coolability for the badly deformed bundle at the end of the simulated LOCA.

Project Status

- Detailed design completed, fabrication underway.
- Test matrix covering "large", "medium", and "small" break LOCAs being elaborated.
- Theoretical scoping and precalculation of tests underway but much new modelling of processes to be done.

Next Steps

End '79, decision by the Council of Ministers of the European Communities on the funding of the ESSOR reactor and SSTP from 1980 on.

	CLASSIFICATION: 1.3
TITLE (ORIGINAL LANGUAGE): Falling film shrouds rewetting	COUNTRY: Italy SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE):	ORGANISATION: CNEN PROJECT LEADER: G. Farello
INITIATED: March 1978 / COMPLETED:	SCIENTISTS: G. Ferrari
STATUS: In progress / LAST UPDATING: June 1979	

1) General Aim

Measure of the rewetting velocity and of the sputtering characteristics.

2) Particular Objectives

Types of flow (rivulet etc.) and droplet characteristics measurements in dependence of flowrate, subcooling, initial wall temperature, etc.

3) Experimental Facility

"Ad hoc" test section with visualization devices.

4) Project Status

A first set of measurements has been completed.

5) Next Steps

A second test section has been designed and proposed for the construction.

6) Contact Person

G. Farello, CNEN, CSN Casaccia, CP 2400, I-00100 Roma.

	CLASSIFICATION: 1.3
TITLE (ORIGINAL LANGUAGE): STUDIO DEL FENOMENO DI RIBAGNAMENTO DI SUPERFICI AD ALTA TEMPERATURA CON PARTICOLARE RIFERIMENTO ALLE GUAINE E AI CANALI DEL COMBUSTIBILE NUCLEARE	COUNTRY: ITALY SPONSOR: CONSORZIO NUCLITAL
TITLE (ENGLISH LANGUAGE): STUDIES ON THE REWETTING OF HIGH TEMPERATURE SURFACES, WITH PARTICULAR REFERENCE TO FUEL CLADDING AND FUEL CHANNELS	ORGANISATION: UNIVERSITA' di PALERMO x PROJECT LEADER: E. OLIVERI
INITIATED: 1978 / COMPLETED:	SCIENTISTS: F. CASTIGLIA, S. TAIBI, G. VELLA
STATUS: IN PROGRESS / LAST UPDATING: JUNE 1979	

x ISTITUTO DI APPLICAZIONI E IMPIANTI NUCLEARI

DESCRIPTION:

The aim of this study is:

1- to compare critically the existing analytical one- and two-dimensional models on the rewetting of high temperature surfaces, qualifying them on the base of existing experimental data;

2- to gain detailed information on the physics of rewetting, singling out the main physical parameters involved in the phenomenon and try to improve the existing theoretical models;

3- to carry out both a conceptual study and a preliminary project of some basic experiences, where the different effects will be taken separatly into account, in order to test the effectiveness of the various theoretical models.

(Istituto di Applicazioni e Impianti Nucleari, Università, Parco d'Orleans, I-90128 Palermo)

	CLASSIFICATION: 1.3
TITLE (ORIGINAL LANGUAGE): Heat transfer and rewetting in PWR reflood (HTH 6.2.1)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Winfrith)
	PROJECT LEADER: J Fell (Winfrith)
INITIATED: March 1978 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: March 1979	

Background

There is a requirement for an improved understanding of the heat transfer and hydraulic processes taking place during the reflood stage of a PWR/LOCA in order that more realistic predictions of fuel behaviour may be made for deformed cladding.

Objectives

The experimental objective is to develop a detailed understanding of the fluid flow and heat transfer processes involved in PWR reflood, leading to the development of predictive models which can be applied to undistorted and distorted fuel bundles.

The investigations will be limited to conditions of fixed coolant input; the analysis of system effects does not form part of the present programme.

A number of new experimental facilities are involved:-

1. A single tube rewetting rig (REFLEX).

2. A perspex 76-pin air flow model.

3. A perspex 76-pin electrochemical analogue model.

4. A full length 64-pin bundle which will probably be mounted in either the 9 MW heat transfer rig or the HPSC rig.

Programme

1. One dimensional reflood heat transfer studies using tubular test sections including simulations of deformed clad constrictions.

2. Basic three dimensional studies of the effect of partial blockages, by-pass and phase separation.

   a. Deformed clad cluster hydraulics using an air flow model.

   b. Deformed clad cluster heat transfer using an electrochemical analogue.

   c. Reflooding of interconnected deformed and non-ballooned subchannels.

3. Rewetting studies including the influence of clad, filler and gas gap.

4. Confirmatory large scale deformed cluster reflood heat transfer and hydraulic performance test in 8 x 8 bundles of full length pins.

5. A supporting development programme will be directed towards the production of well instrumented model fuel pins suitable for use in (4) above and novel instrumentation such as water droplet size measurement and steam superheat measurement.

	CLASSIFICATION: 1.3
TITLE (ORIGINAL LANGUAGE): Clad deformation and multi-pin interaction effects in a LOCA (FS 1.1A)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Springfields)
	PROJECT LEADER: D O Pickman (SNL) J B Sayers (Harwell)

INITIATED: March 1978	COMPLETED:	SCIENTISTS:
STATUS: In progress	LAST UPDATING: March 1979	

Background

In a loss-of-coolant accident the deformation of the fuel rod cladding is an important factor in determining the efficiency of the core quenching system by reflooding. It is known that, under certain conditions of temperature and pressure differential, Zircaloy cladding can strain by large amounts (> 25%).

Objectives

To determine the strain behaviour of cladding under LOCA conditions, and to establish whether any large scale deformation will occur which could impair cooling by reflooding. If this is the case, to study and recommend remedial measures.

Programme

1. Laboratory Studies

Previous studies of the deformation behaviour of cladding[1] will be extended:

a. Directly heated cladding will be tested under conditions (loss of heat by convective cooling) appropriate to reactor behaviour.

b. The presence of adjacent rods will be simulated.

c. Cladding will be tested with internal heaters more closely representing fuel.

d. The destabilising effects[2] of local temperature variations and the anisotropic mechanical properties of the cladding will be studied.

e. Interactive effects in multi-rod assemblies will be studied by tests in isothermal conditions (muffle furnace), tests on directly heated arrays, and tests on arrays with internal heaters.

FS 1.1A contd

2. In-pile studies (LOCT experiment)

Following completion of the feasibility study(3) for a single rod LOCT (Loss of Coolant Test) experiment in the DIDO high pressure water loop, the next phase comprises a design study intended to produce a fully worked up scheme by the end of 1979.

In the design study some variations in design are proposed stemming in part from an improved understanding of the thermo-hydraulics through use of later versions of the RELAP computer code. Provision is also being made for the introduction of supplementary coolant, post blowdown, thus giving a measure of convective cooling to the fuel rod. The active part of the test section will be designed to accommodate a single pin with a length comparable to the PWR axial grid spacing. The pin will be surrounded by an electirc heater shaped to represent the neighbouring fuel pins. The heat generation, heat loss, and rise in temperature will be designed to be similar to that which might be experienced by one section of a PWR bundle. The objective is to attain an operating envelope covering clad temperatu in the range 600-800°C. However it is planned that the first experiment should be radiatively cooled in order to establish the experimental method in the most expeditious way. An out of pile test section would be linked to the high pressure water loop with an electrically heated test rod thus enabling the whole sequence of operations to be rehearsed and also to give direct comparison of an electrically heated pin with a fuel rod. Operating procedures will be designed to be fully automated and micro-processor controlled.

On the assumption that completion of the design study indicates that a worthwhile series of experiments can be mounted the forward programme involves modifications to the loop starting in 1980, recommissioning the loop to operate at full PWR pressure, improvements to the safety instrumentation and construction of a detailed safety case for the experiment. The first in-pile tests would be in 1982.

Facilities

PROPAT rig (single rod tests) FLEET-2 (multi-rod tests) at Springfields.
High Pressure Water Loop in Harwell DIDO reactor.

Reference Documents

1. HINDLE E D. Zircaloy fuel clad ballooning tests in steam. ND-R-6(S) and 1st Supplement.

2. ROSE K M. Stable and unstable deformation of Zircaloy fuel pin cladding in a LOCA. ND-M-151(S).

3. B MARLOW, C LANG and C J CRABBE. Design feasibility for controlled-loss-of-coolant studies using the high pressure water loop in DIDO reactor (LOCT test). Paper at international colloquium on irradiation tests for reactor safety programmes Petten. June 1979.

	CLASSIFICATION: 1.3
TITLE (ORIGINAL LANGUAGE): Modelling of fuel element behaviour during transients (FS 4.1)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Springfields)
	PROJECT LEADER: D O Pickman (Springfields)
INITIATED: March 1978 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: March 1979	

Background

During a loss-of-coolant accident or other transient the combination of a decrease in clad strength and an increase in differential gas pressure may cause the cladding to swell and partially block the coolant channel. The assessment of the likelihood of such an event involves both experimental and theoretical work. This present job is concerned primarily with the theoretical side, although contact will be maintained with certain experiments, particularly the PROPAT* experiment at UKAEA Springfields Laboratories and any UK or overseas irradiation experiments.

Objective

The transient codes CANSWEL, MABEL will be developed and will be compared with FRAP-T and HOTROD to give improved models of clad deformation and heat transfer. The MABEL code is being jointly developed with AEEW Winfrith (see project HTH 6.1.1). Particular attention will be given to the prediction of strain distribution during clad deformation in a LOCA, rod/rod interaction and gap conductance modelling. The codes will be assessed against any available irradiation experiments or pressurised laboratory tests.

Facilities

The computers installed at Springfields, Harwell and Risley will be used for this work.

Reference Documents

1. JONES P M. CANSWEL-1 A computer code to predict SGHWR fuel pin behaviour in a loss-of-coolant transient. First technical progress report. TRG Report 2841(S).

2. JONES P M, GITTUS J H and HINDLE E D. CANSWEL A computer model of clad behaviour during a loss-of-coolant accident. TRG Report 2901(S).

* LOCA clad deformation studies (Project FS 1.1A)

	CLASSIFICATION: 1.3
TITLE (ORIGINAL LANGUAGE): Rewetting, reflooding and clad deformation. MABEL development (HTH 6.1.1)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Winfrith)
	PROJECT LEADER: J Fell (Winfrith)
INITIATED: March 1978 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: March 1979	

Background

There is a potential for large clad strains to develop during a LOCA. This could result in axially extended deformation with consequent impairment of the heat transfer to the coolant and thus severe overheating of the fuel. A specialised code (eg MABEL) is required to examine the detailed behaviour of single fuel rods taking into account local azimuthal effects, subchannel blockage, etc. It takes its boundary conditions from a core-wide code such as RELAP.

MABEL is being developed in collaboration with UKAEA Springfields Laboratories, in three stages: MABEL-1 with a concentric pellet model and the coolant represented by input heat transfer coefficients: MABEL-2 with a 2D pellet model to represent azimuthal effects and coolant subchannel effects included: MABEL-3 with a more sophisticated model based upon the experience gained from MABEL-2.

Objectives

1. To develop the MABEL series of codes for PWR LOCA clad-ballooning studies.

2. To carry out analytic studies to validate the assumptions and modelling in MABEL; to update the model in the light of experimental work at AEEW or elsewhere.

3. To carry out PWR calculations as required.

Programme

Completion of MABEL-1	Autumn 1978
" " MABEL-2	Autumn 1979
" " MABEL-3	1980

Facilities

MABEL will be programmed for the IBM 370 Computer.

Reference Documents

1. R W Bowring and C A Cooper. "MABEL-1: A code to analyse cladding deformation in a Loss-Of-Coolant Accident". AEEW-R-1215

2. CORE MELTDOWN

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.-31.12.1978	2	RS 263/11

Vorhaben/Project Title	
Analytische Tätigkeiten der GRS im Rahmen des Reaktorsicherheitsforschungsprogramms des BMFT - Kernschmelzen Analytical Acitvities of the GRS in the Frame of the BMFT Research Program on Reactor Safety - Core Meltdown	Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.4.1977	31.12.1979	Dr.Kreusenhoff/Dr.Scharfe
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

Parallel and additional analytical studies for the BMFT-sponsored experimental research. Further development of computer codes concerning reactor safety.

2. Particular Objectives

Review and analysis of research reports and results of the core meltdown project. Review of LMFBR-related experiments with debris beds and molten pools.

2.1 Assessment and evaluation re RS 166
2.2 Assessment and evaluation re RS 72 a/b
2.3 Assessment and evaluation re RS 73
2.4 Assessment of the research projects dealing with steam explosions (RS 76; RS 76 A; RS 2o6)
2.5 Assessment of the research projects dealing with the interaction between molten mass and reactor concrete (RS 154; RS 183; RS 237)
2.6 Review of the meltdown process and behaviour of molten material after hypothetical accidents (LMFBR).
2.7 Review of Fuel-Coolant-Interactions
2.8 Review of behaviourand cooling of fuel debris beds.
2.9 Review of behaviour of a molten pool (LMFBR)
2.1o Review of interactions between core melt and core-catcher

3. Research Program

3.1 Assessment of the results obtained within RS 166 with a view to their relevance to risk statements. Examination of the operability of the BETON computer program.

3.2 Examination of the operability of the BILANZ computer program in connection with risk statements.

3.3 Review of the inherent value and significance of MELSIM for the analysis of the meltdown and slumping process. Comparative calculations with the BOIL computer program.

3.4 Assessment of the results obtained within the research projects dealing with steam explosions and review of their applicability and transferability to reactor-specific conditions.

3.5 Assumption, review and assessment of the analytical methods as well as the computed results for a substantiation of the statements made in continued risk studies.
Evaluation of the experimental results of RS 154 with a view to the H_2 and steam release rates as well as the meltdown behaviour of the concrete for reactor-speific data.

3.6 Review of acticities concerning the meltdown and the behaviour of molten materials in the core.

3.7 Critical assessment of experiments and analytical methods to investigate fuel-coolant-interactions. Investigation of the applicability to reactor conditions.

3.8 Critical assessment of experiments concerning fuel debris beds using simulant materials as well as reactor materials.

3.9 Review of experiments. Implementation of the computer code THEKAR and parameter calculations.

3.1o Critical assessment using simulant materials as well as reactor materials.

4. Experimental Facilities, Computer Codes

Ad 3.9: The code THEKAR was obtained from TU Hannover. The version THEKAR R has been implemented.

5. Progress to Date

Re 3.3 Review of the inherent value and significance of MELSIM for the analysis of the meltdown and slumping process. Comparative calculations with BOIL.

Re 3.4 Assessment of the results obtained within the research projects dealing with steam explosions and review of their applicability and transferability of reactor-specific conditions.
The amount of the molten mass and the kind of failure of the lower core support grid are essential boundary conditions for the occurrence of a steam explosion.
Participation in the preparation of a strategy for the necessary development of corresponding slumping models.

Re 3.5 Assumption and implementation of the BETZ computer program developed within the frame of RS 183. A parameter study was carried out to cover the variables of major influence.
Preparatory action for the assumption of the KAVERN program.

Ad 3.6 The transition phase which describes the graual non-energetic meltdown of the core has been analysed by reviewing the literature.
The analysis of theoretical and experimental investigations concerning the streaming of molten materials in narrow channels has been initiated.

Ad 3.7 Recent French CORECT tests - utilizing up to 8 kg of molten fuel - and recent Sandia in-pile-tests with carbide and oxide fuel have been reviewed.
The summary of a literature review concerning fuel-coolant interactions was published in a report /1/.

Ad 3.8 The literature and experiments concerning debris bed behaviour has been reviewed. Conditions leading to dryout have been studied. Different dryout correlations form three research centers, Argonne National Laboratory-ANL, University of California-UCLA, Sandia Laboratories were compared. Remelting of dry debris beds was considered. A preliminary report /2/ was published.

Ad 3.9 The present status of molten pool analysis has been studied. Mechanisms affecting heat transfer characteristics of molten pools have been established, including radiative heat transfer. The influence of fuel crust behaviour on the damage potential of the melt has been assessed.
The computer code THEKAR R has been implemented. A preliminary report /2/ was publisched.

Ad 3.1o According to the program schedule work on this topic will begin in 1979.

6. Results

Re 3.3 One of the results of the investigation of the entire core meltdown accident relevealed that Phase 4 is the decisive one for questions relating to containment integrity and fission product release. Thus, the detailed treatment of Phase 1 is shown to be less important than parameter studies and a delimitation of the possible scattering range of the results. This range of results is a consequence of the scattering of input data and boundary conditions which is due to both the statistical approach and the experiments involved. However, for the implementation of such parameter studies it will be sufficient and even advantageous to use a less involved and less sophisticated program.

All in all, the results obtained with the simpler BOIL program are not only comparable to those of MELSIM but even easier to interpret. Neither program will supply a criterion for the core slump. However, the exact time of the core slump is only of secondary importance for course and development of the entire core meltdown accident. The integral value of the steam production rate can be determined, and the exact time history is insignificant as far as contiued calculations are concerned. With regard to the hacrogen release an integral value will also be sufficient. Here, however, the value will be underdetermined because of the assumptions made in MELSIM /3/.

Re 3.4 The slumping process may be studied with the aid of relatively simple models. The models will not require any refinement unless and until the first few results have been obtained. A substantiation of the statements by way of small-scale experiments is only possible to a smaller extent and not necessary at present.

Re 3.5 The BETZ computer program deals with the -hase of concrete destruction. As the program is now available, it determines the concrete melting rate, the temperature of the molten mass as a function of time and the various heat fluxes. Thermal conduction in the concrete is not taken into consideration. BETZ is the predecessor of the detailed KAVERN program that is still in the development stage, and it is part of the superordinate BILANZ program that, once completed, is hoped to link all subroutines without necessitating manual intervention.

A parameter study investigated the essential input data with a view to their influence on the molten concrete masses.

The result reveal that there is very little dependence on the input data. In other words, the destruction of the concrete is mainly a function of the heat sources inherent in and the heat removal from the molten mass.

The kind of thermohydraulic model that was used was found to be the most essential factor influencing the formation of the melting cavern and thus the melt-through times of the inner beilogical shield.

In addition, the effect of a sump water contact on the time history of Phase 4 was determined /4/. Calculations with thwo heat transfer models lead to comparable results the heat transfer at the interface between molten mass and concrete. Depending on the model, the melt-through time of the inner biological shield is approx. 4 or 5.6 hours.

As compared with the reference case, a 5o % increase in radiant energy released from the surface of the molten pool will prolong the melt-through time of the inner biological shield by a maximum of 1.1 h (= 2o %). If the radiant energy release becomes zero (closed athermanous crust) the melt-through time will be shortened by approx. 1.7 h as compared with that of the reference case (5.6 h).

The melt-through time and thus the point of time of a possible sump water contact will change considerably if there is no locally constant heat flux transferred to the concrete, as has been presumed in the calculations so far. Experiments which are being carried out with regard to the heat transfer at the interface between molten mass and concrete (RS 166) suggest that the transferred heat flux max depend on the angle of inclination of the interface. In the case of a linear increase in the heat flux with the angle of inclination (doubling of heat flux in the case of a vertical interface), the melt-through time will be reduced ny approx. 5o %. A greater effect upon the time history of Phase 4 of a hypothetical core meltdown accident will result if there is a contact between molten mass and sumpf water. The heat transfer to the sump water is expected to the effected by way of stable film boiling. A BETZ calculation reveals that almost the entire energy content of the molten mass will serve to vaporize the sump water so that the molten mass will solidify approx. 2.8 h after the occurrence of sump water contact. The molten mass will not penetrate the concrete any further provided there is still some sump water left. At present, BETZ w811 furnish only a relatively coarse description of the interaction between molten mass and sump water. Because of the important influence of the sump water contact upon the course of the core meltdown accident it will be necessary the further improve the program.

Ad 3.6 The study of freezing models and of the ANL-thermit-experiments shows that plugging above the active core region cannot be described statisfacorily. Related investigations should be continued.

Ad 3.7 The mechanical energy release of the CORECT tests was less than 4 J/g-fuel. The Sandia in-pile-tests showed a maximum pressure peak of about 1ooo bar. This was probably caused by single-phase superheating of liquid sodium, and not by sodium vaporization.
The theories which try to model the mechanism of fuel-coolant-interactions were reviewed. Wehn they are applied to conditions which are expected during hypothetical accidents they show rather low probabilities for energetic fuel-coolant interactions. But final conclusions cannot be made at present.

Ad 3.8 Based on experiments with simulant materials, different correlations for fuel debris bed dryout were published by ANL, by UCLA and by Sandia Laboratories. Although the correlations show a certain quantitative agreement of the dryout threshold predictions, there are substantial differences in modeling the dryout process. Consequently, a reliable assessment of the importance of characteristic bed parameters - e.g. particle deameters and bed porosity - cannot be made as yet. The first in-pile-experiments at Sandia showed dryout thesholds which could be explained qualitatively by existing correlations. At dryout threshold power levels the dry parts of the beds still had a remarkable heat removal capability leading to rather small temperature transients.

Further in-pile-experiments at Sandia as well as experiments with electrical heating of UO_2-particles under sodium at ANL are planned to improve qualitative and quantitative understanding of debris bed behaviour. The main objectives are to investigate the influence of several parameters, e.g. higher power levels, additional steel particles, depth-dependent porosity.

Ad 3.9 At various research centers numerical calculations and experiments with simulant materials at low temperatures produced heat transfer correlations for nonboiling homogeneous pools with simple boundary conditions. The results show quite reasonable agreement. Functions were found correlating mean Nusselt numbers with the Rayleigh number. Marked local maximum heat fluxes were observed at the upper side walls of the pool.
The results cannot be applied directly to accident conditions since they do not consider some important aspects:

- Boiling in the pool and interactions of different materials will affect convection and heat distribution.
- Due to the high temperatures thermal radiation will contribute significantly to the heat transfer.
- The damage potential of the pool is higly dependent on the properties of the fuel crust at its boundaries.

Experiments with molten UO_2 at the ANL showed results which did not fit to the previous correlations, whereupon an additional radiative heat transfer model remarkably improved the agreement between calculations and test results.
Experiments with molten UO_2 will be continued at the ANL.
At Sandia, in-pile investigations with molten UO_2 are being prepared.

7. Next Steps

LMFBR: Review of current research concerning core meltdown, fuel debris bed behaviour, molten pool behaviour and coremelt-concrete interactions will continue.

8. Relations with Other Projects

9. References

/1/ V. Javeri
Zur thermischen Reaktion zwischen geschmolzenem Brennstoff und Natrium im schnellen natriumgekühlten Reaktor
GRS-A-235

/2/ H.Löffler
Das Verhalten von Brennstoffpartikelschüttungen und Kernschmelze nach einem hypothetischen Störfall in einem schnellen natriumgekühlten Reaktor.
GRS-A-242

/3/ Auftragsbericht GRS-A-139
K.Bracht, J. Keusenhoff
Beurteilung und Auswertung von Forschungsergebnissen des Projektes Coreschmelzen

/4/ Auftragsbericht GRS-A-221
K.Bracht
Der zeitliche Ablauf der Betonzerstörungsphase bei einem hypothetischen Kernschmelzunfall: Berechnete Auswirkung verschiedenartiger Einflußgrößen

1o. Degree of Availability of the Reports

The above cited references are available upon request from Gesellschaft für Reaktorsicherheit (GRS)/Forschungsbetreuung, Köln.

Berichtszeitraum/Period 1.1. - 31.12.1978	Klassifikation/Classification 2.1	Kennzeichen/Project Number 06.01.12/01A (PNS 4321)
Vorhaben/Project Title Experimentelle Untersuchung der Abschmelzphase von UO_2-Zircaloy-Brennelementen bei versagender Notkühlung Experimental Investigations of the Meltdown Phase of UO_2-Zircaloy Fuel Rods under Conditions of Failure of Emergency Core Cooling		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe GmbH. Projekt Nukleare Sicherheit / IT
Arbeitsbeginn/Initiated 1973	Arbeitsende/Completed 1978	Leiter des Vorhabens/Project Leader Dr. S. Hagen
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General aim

Experimental Investigations of the melting process including the re-solidification of the melt on colder parts for single rods and bundles with spacers, ballooned cans and absorber rods. The influence on the melting process from different parameters like surrounding atmosphere and temperature gradients will be investigated.

2. Particular objectives

Investigation of the influence of preoxidation, ballooning and absorber-rods on the melting and refreezing behaviour of bundles.

3. + 4. Research program and experimental facilities

They are described in earlier reports.

5. Progress to date

1978 have done meltdown experiments in steam on bundles of 3 x 3 rods with additional fiber ceramics isolation.
The bundle consisted of a central solid pellet rod, which was surrounded by ring pellet rods with 6 mm tungsten heater. The preoxidation of the cans was done at 550 °C. The ballooned cans were produced in the REREKA-facility.

For the absorber experiments the original absorber rods in 30 cm length were used in the centre of the bundle. According to the different types of absorber rods in the reactor we have done experiments with "black" absorber rods, consisting of Ag/In/Cd-alloy (80/15/5), "grey" absorber rods made of Inconel 600, the burnable poison borosilicate glas and empty chrome-nickel-steel guide tubes.

6. Results

The experiments on the refreezing behaviour confirm, that in all bundles heated in steam the melt produced first is refreezing during moving down into the colder part of the bundle. Thus the molten material developing later is forming a lump between the rods.
The investigations of the composition of the melt for one of the bundles showes an increase of the Uranium concentration from the lower to the upper part in the region 20 - 50 w %. Melt refreezing first in the lower part has the lower concentration of Uranium.
For the investigation of the influence of preoxidation we have used cans with an oxid layer of 20 μ. The bundle was heated to an surface temperature of the central rod of 1950 ^{o}C. We could find no difference in the meltdown and refreezing behaviour compared to not preoxidized rods.
In the experiments with ballooned rods we could find no preferred meltdown of the ballooned regions.

The beginning of material transport by melting of the absorber rods is determined by the chrome-nickel steel tubes. The Ag/In/Cd-alloy melting at 800 ^{o}C and the Borosilicate glas softening at the same temperatur are inclosed inside the tubes and restricted to its original region. Only when 1400 ^{o}C is reached, than the chrome-nickel steel tube failes and the liquified material is pouring out. The first eruption of Ag/In/Cd-alloy is quite violent, in consequence of the strong vapour pressure of indium. This primary melt and the molten material of absorber, can and guide tube, coming down when the hot temperatur region is spreading out, is refreezing in the lower region of the bundle in the form of a lump.

A similar behaviour - with the exception of the violent first outburst of the molten material - we have found for the Borosilicate glas, Inconel 600 and the empty tubes. In all experiments in the lower region of the bundle there has formed a lump of frozen material. Regarding the fuel-element one has to remember that only half of the fuel rods is in close neighbourhood to an absorber rod.

7. Next steps

The experiments are completed at the end of 1978. But the experimental arrangement is kept available for operation, if there is the necessity for further experiments in the region of core meltdown.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	2.1	RS 205

Vorhaben/Project Title	
Nachrechnung von Stabexperimenten und Absicherung von MELSIM (RS 205 - I. 1.5, Jahresbericht A 77) Calculation of Fuel Pin Meltdown Experiments and Application to MELSIM	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Institut für Kernenergetik und Energiesysteme der Universität Stuttgart

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
June 1, 1976	June 30, 1979	Dr.F.Schmidt/Prof.H.Unger
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

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" (KfK).

The present research project is aimed at the calculational verification of the results obtained at the KfK. An experimentally verified model of the meltdown process is to be developed later in order to describe the integral course of the core meltdown. This is of importance for the slumping modul of MELSIM.

2. Particular Objectives

The computer model is aimed at the solution of the following problems

- Fuel pin heatup in oxidizing (H_2O, air), reducing (H_2) and inert atmosphere
- Oxidization of the clad (zirconium) dependent on a differing oxigen supply, H_2-generation
- Influence of the interaction between UO_2 and molten zircaloy on the fuel pin behavior
- Pin failure depending on the radiation history
- Meltdown process in bundles of fuel pins
- Influence of the tungsten heater on the results of the single pin experiment

- Formation of channel blockage
- Geometric interactions during the slumping process

3. Research Program

3.1 Calculation of the Pin-Heatup Until Clad and Fuel Interact

The pin-heatup has to be simulated according to the different experimental conditions in different atmospheres. Temperature distribution, Zr-H_2O-reaction including heat production, formation of a zirconium oxide layer and the formation of H_2 has to be taken into account.

3.2 Simulation of the Pin-Behavior up to Clad-Melting-Temperatures

The interaction between Zr and UO_2 has to be described by a model.

3.3 Simulation of the Pin-Failure and the Meltdown Process

Pressure differences between the interior of the pin and the coolant have to be calculated and the behavior of the molten material has to be described.

3.4 Supporting Calculations with Respect to Rod-Bundle-Experiments

Bundle experiments are carried out to support the development of MELSIM. Therefore these experiments have to be calculated and the results of theory and experiments have to be analyzed. They have to be investigated with respect to consequences for the development of MELSIM's slumping modul and the modelling of the meltdown of the reference reactor.

4. Experimental Facilities, Computer Codes

No experiments are carried out within this project, the necessary experimental information is supplied by KfK (PNS 4240). Computer codes developed at IKE, modelling fuel pin behavior (e.g. STT, ZET-1D, ZET-2D, WUEZ) as well as the modular program system MELSIM are employed to the extend necessary. The experimental and theoretical knowledge is used to develop an improved computer code.

5. Progress to Date

to 3.3 Pin failure and melting behavior have been modeled, programmed and tested.

to 3.4 The analysis of further results of fuel rod bundle meltdown experiments has been continued. A two-dimensional model for the investigation of angular effects has been completed. The modeling of failure and meltdown of pins in bundle geometry has been continued. A description of the meltdown process in the entire reactor, based on experimental results has been started.

6. Results

to 3.3 Calculations using the module STAMEL were in good agreement with experimental results.

to 3.4 The pins fail at a temperature of $\approx$2100 K. The material remains within the core region. The melt freezes as soon as it reaches areas of the rods with a surface temperature of appr. 50 K below the melting temperature.

7. Next Steps

to 3.4 Further information for relevant meltdown parameters and for the improvement of the program for the integral meltdown experiments is provided. The fuel rod bundle model is verified by means of a further experiment. The final evaluation of the experiments is completed.

8. Relations with Other Projects

The experimental data for the performed and intended investigations are provided by KfK (Research Project PNS 4240). There is a strong dependence on the experimental program. In return, the experimental program of KfK will be supported by the theoretical investigations.

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	2.1	RS 211

Vorhaben/Project Title	Land/Country
Analyse der Zwischenphase Kernversagen - Schmelzsee, Bestimmung der Energiebilanzen nach hypothetischem RDB-Versagen Investigation of the Phase Between Failure of the Core and Assembling of the Molten Material in the Pressure Vessel; Integration of the Program MELSIM 1 into BILANZ	FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Institut für Kernenergetik u. Energiesysteme der Universität Stuttgart

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.7.76	31.3.78	Prof.Dr.Unger/DP Körber
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	December 1978	

1. General Aim

The physical behavior of a reactor core under hypothetical core meltdown conditions is to be investigated starting after the failure of the lower pin-supporting structure of the core and ending if the core is molten and assembled on the lower plenum of the pressure vessel. The vaporization process of the water contained in the lower plenum will be calculated as well as the heatup of the dry reactor vessel assuming different configurations of the core debris which fall into the plenum. A computer program (LÜCKE) will be developed. In order to obtain improved information on the heat balances involved in core meltdown, this program will be integrated into the energy balance program BILANZ I (PWR) and BILANZ II (BWR) of KWU together with the computer code MELSIM-1.

2. Particular Objectives

This project is closely connected to the research project RS 183 (energy balances after hypothetical pressure vessel failure) of KWU. Within this framework, KWU and IKE will couple the program systems listed above.

The particular objectives are as follows

- Investigation of the core meltdown accident by means of MELSIM-1 until the core falls into the lower plenum of the pressure vessel. Integration of MELSIM-1 in BILANZ I and II.

- Investigation of the behavior of the remaining core after partial failure of the lower core supporting structure.

- Modelling of the sequences of events in the reactor pressure vessel until the molten core is assembled in the lower plenum of the pressure vessel

- Completion of the computer program LÜCKE which simulates these events.

- Coupling of LÜCKE with the program system BILANZ - MELSIM-1 and investigation of the entire accident sequence.

3. Research Program

3.1 Search and processing of the PWR and BWR data required for computer calculations which are carried out with MELSIM-1

3.2 Integration of MELSIM-1 in BILANZ
- Coupling of MELSIM-1 and BILANZ
- Heatup and slumping of the remaining core after a first partical failure of the pin-supporting structure.

3.3 Development of a simple model to describe the sequence of events from the failure of the core supporting structure until the formation of core melt at the bottom of the pressure vessel
- Analysis of the accident sequence
- Development of the computer code LÜCKE
- Performance of the calculations

3.4 Integration of LÜCKE into BILANZ

3.5 Simulation of the accident in applicating the complete model to a standard PWR- or BWR-type and evaluation of the results.

4. Experimental Facilities, Computer Codes

Within the research project, the computer program LÜCKE is developed. It will be integrated together with MELSIM into the code BILANZ of KWU.

5. Progress to Date

to 3.5 Completion of the calculations with the code system developed.

6. Results

Calculations, performed for LWR's with the code system MELSIM-1 show that in the lower areas of the core a substantial amount of heat is released by means of the Zr-H_2O-reaction.

During the heatup- and meltdown phase, approx. 10 % of the total Zr-inventory of the core are oxidized. If the PWR-core is initially dry, core collapse takes place appr. 1400 s post accident initiation. At almost the same time the inner part of the supporting structure fails. 2600 s later failure of reactor pressure vessel takes place.

Calculations with MELSIM-1 and CONZU show that the containment withstands the pressure load at least for the time-span investigated.

7. Next Steps

Completed

8. Relation with Other Projects

There is a strong dependence on the project RS 73 (Development of the Computer Code MELSIM) and a close coupling to the investigation program RS 183 (Energy Balances after Hypothetical Failure of the Reactor Vessel). Further on this project is connected to the RS 316 program (Development of a Core Melting System on the Basis of RSYST).

9. References

Final Report BMFT-RS 211

10. Degree of Availability of the Reports

Available by GRS, Cologne

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	2.1	RS 316
Vorhaben/Project Title Beiträge zur Entwicklung von MELSIM-1 und zur Kopplung von Teilprogrammen zum Kernschmelzen auf RSYST-Basis Contributions to the Development of MELSIM-1 and to the Coupling of Core Meltdown Programs based on RSYST		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Institut für Kernenergetik u. Energiesysteme der Universität Stuttgart
Arbeitsbeginn/Initiated 1.11.77	Arbeitsende/Completed 31.3.80	Leiter des Vorhabens/Project Leader Prof.Dr.H.Unger/Dr.W.Gulden
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

- Contributions to the development of the program system MELSIM-1, testing and docomentation
- extension of the program system RSYST for the integrated description and simulation of the core meltdown accident. Development and test of a core meltdown program system, based on RSYST, which is suitable to simulate the entire course of the core meltdown, using the relevant core meltdown programs available.

MELSIM-1, which is able to describe the meltdown process extensively and detailed, is improved with respect to data handling and computer time by means of a change in the mathematical description of the meltdown process.

For the research project "core meltdown" of the BMFT, computer programs (MELSIM-1, LÜCKE, KAUHZ, CONZU and KAVERN, e.g.) have been developed, which simulate the four phases of the core meltdown accident (core heatup, evaporation of the residual water, heatup of the reactor pressure vessel and concrete penetration).

In order to treat the entire accident sequence, an integral program system is required, which automatically links the information without any losses or approximations between the programs describing parts of the sequence.

For the development of MELSIM-1 the use of RSYST /1/ has been of advantage. The high flexibility of RSYST enables integration, coupling or exchange of other programs or modules. Of further advantage is the fact, that programming for standard problems, e.g., can be done in such a manner, that users without particular experience with RSYST are able to handle the entire core meltdown program system.

2. Particular Objectives

2.1 Development of MELSIM

- Improvement of the models in the core heatup module
- steam generation rate flattening during core collapse
- improvement of the decay heat models and of the data exchange between different modules
- adaption of the modules to the improved data structure
- simplification of in- and output, test and documentation

2.2 Integral Description of the Core Meltdown Accident, Based on RSYST

- Analysis of the uncoupled programs and their interaction in the integrated system
- programming of the integrated core meltdown code system
- adaption of codes which already exist
- test and documentation of the entire integrated system

3. Research Program

3.1 Core Meltdown Code System MELSIM-1

Reduction of computer time, test and documentation.

3.2 Use of RSYST for an Integral Description of the Accident

An analysis of the programs and their interaction in the entire program system is performed. The data interfaces are fixed, additional modules are programmed.

3.3 Adaption of Programs Available

Assistance is given to KWU (see No. 8) in order to adapt their available programs to RSYST.

3.4 Testing of the Program System

The system is tested in its parts - by this contractor only with respect to MELSIM-1 and LUECKE - and integrated.

3.5 Evaluation of the Results and Final Report

4. Experimental Facilities, Computer Codes

Computer codes, developed by the contractor, are used, particularely RSYST, as well as MELSIM-1 and LUECKE. They can be applied, according to the demands, in parts or entirely in the module sequences required. The external programs, e.g. KAUHZ, CONZU and KAVERN will be supplied by KWU.

5. Progress to Date

to 3.1 Activities to save computer time and test runs have been carried out. The models for core heatup and -slumping have been concentrated in the module HEIZ. Those modelling boiling of residual water (KOCH) and of the heatup of core supporting structure (UMGEBU and UMGEBR) have been extended. The data transfer structure has been improved.

to 3.2 and 3.3 The analysis of core meltdwon programs resulted in the definition of interface data and the conception of two auxiliary modules STRAM and AKKU. STRAM realizes a generalized system timestep control and AKKU automatically accumulates important transient results on the database.

6. Results

to 3.1 The models of core-heatup and core-slumping were combined in order to reduce the data-input and -output. The model for the core-supporting structure includes the determination of the fall-down-times of core regions which failed.

to 3.2 and 3.3 After the integration of the modules STRAM, AKKU, CONZU, KAUHZ and WAVER into the core meltdown system KESS, first results for the first two phases of the core meltdown accident simulation were obtained.

7. Next Steps

to 3.1 The improvement of the models and data organisation of the core meltdown code MELSIM-1 and LUECKE with respect to the core meltdown system will be finished.

to 3.2 The modules STRAM and AKKU will be tested and improved. Programs for analyzing the 3rd and 4th phase of a core meltdown accident will be integrated into the system KESS.
and
3.3

8. Relation with Other Projects

Close cooperation with KWU (Project No. BMFT - RS 183).

9. References

/1/ Ruehle, R.
RSYST, an Integrated Modular System with Data Base for the Automatic Calculation of Nuclear Reactors (in German).
Report Nr. 4-12, Jan. 1973 of the Contractor

10. Degree of Availability of the Reports

Available at the GRS, Cologne

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01. - 31.12.1978	2.1	RS 166
Vorhaben/Project Title Theoretical and Experimental Investigation of the Thermohydraulic Behaviour of a Molten Core in the Reactor Vessel and on the Concrete of the Basement Theoretische und experimentelle Untersuchung des Verhaltens eines geschmolzenen Kerns im Reaktorbehälter und auf dem Betonfundament		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Institut für Verfahrenstechnik der Uni. Hannover Callinstraße 36 3000 Hannover 1
Arbeitsbeginn/Initiated May 1975	Arbeitsende/Completed February 1979	Leiter des Vorhabens/Project Leader DM 1.279.100,00
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General aim

The aim of this research project is to investigate the thermal interaction of the molten core material with various components of the reactor. Computer codes are established to describe the attack of the melt to the reactor concrete and the heat transfer between the melt and the boundaries of several reactor components. In order to test the reliability of the codes and to get necessary empirical information, a number of experiments with modeling fluids are performed.

2. Particular objectives

The investigation of the penetration of molten core material into concrete is of particular interest. In order to get this information, the heat transfer between the melt and the concrete must be determined by considering the behaviour of the molten concrete and the vapour and gas liberated from the concrete.

Another object of the research project is the investigation of the heat transport in a stratified system of two immiscible fluid layers, which is formed when the melt contains a metallic and oxidic phase.

Furtheron the heat transfer, the crust-formation and the fragmentation of a molten metal layer, cooled from the top with boiling water, is of particular interest. This case is important to investigate the phenomena occuring at a subsequent flow-in of water on the top of a pool of molten core material.

3. Research program

3.1 Investigation of the heat transfer from an internal heated fluid to an inclined gas liberating wall with the help of modeling experiments.

3.2 Numerical calculation of the formation of a cavity into concrete by molten core material. The heat transfer between melt and concrete, determined in 3.1, is used in this calculation.

3.3 Numerical calculations to determine the shape of rising bubbles and the heat flux to the bottom induced by this bubbles as a function of different parameters.

3.4 Investigation of the heat transport from two stratified immiscible fluid layers with internal heat sources to the boundaries at the top and the bottom of the 2-layer-system. This investigation is done with the help of numerical calculations as well as with a heat transfer model and with experiments.

3.5 Experimental investigation of the heat transfer and the thermal interaction between a pool of molten metal and an overlying layer of boiling water under the conditions of rising gas bubbles from the bottom of the pool. The gas liberation simulates as in 3.1 the gas formation due to the evaporation of water out of the concrete.

The aim of these experiments is to investigate the behaviour of the melt at the surface in order to find out if a formation of an insolating crust between water and melt is possible and to get informations about the fragmentation of the melt.

4. Experimental facilities and computer codes

To 3.1 The experimental investigation of the heat transfer between an inclined gas liberating wall and an internal heated fluid was performed with the help of the holographic interferometry. The construction of the test chamber and the used optical set up were described in an earlier report /1977/.

To 3.2 The penetration of the molten core material into concrete can be predicted by the code BETSI. This code computes the local melting rate of the concrete using an empirical correlation for the heat transfer between melt and concrete. The variation of the mean temperature of the melt is calculated with an energy balance.

To 3.3 To calculate the thermohydraulic behaviour around rising gas bubbles formed at the interaction front between melt and concrete the computer code BETON 3.6 was established. The code can handle two separated phases including the effect of their surface tension.

To 3.4 For the numerical calculation of the heat transfer in a 2-layer-system of immiscible fluids with internal heat sources, the code BETON 3.5 was established. Finally the application of the code was expanded to the conditions when in both layers internal heat sources exist.

To 3.5 The principle design of the test chamber, used in experiments, is shown in figure 9. The pool material is heated by a number of electrical heating tubes. To simulate liberation of gas at the melting front a series of nozzles are positioned in the bottom plate through which gas can be blown in. To determine the heat, transported from the melt to the water, a thermocouple is installed from above to measure the temperature in the melt as function of position and time. For observing and recording the interaction between melt and water photographically, the test chamber and the protection box are equipped with windows.

5. Progress to date

To 3.1 A large number of experiments are performed with various gas flow rates and inclination angles at different distances of the air injection nozzles.

To 3.2 The computer code BETSI was continuously completed with the results of 3.1.

To 3.3 The computer code BETON was completed and results are correlated to 3.1.

To 3.4 A heat transport model for a 2-layer-system was established.

To 3.5 During the period of this report suitable test chambers were constructed and a number of experiments with different initial and boundary conditions were performed.

6. Results

To 3.1 The correlation of the mean Nu-number, describing the heat transfer between an internal heated fluid and a gas liberating inclined wall, yields the following equation:

$$Nu = K\ (X)\ Re^{0,5}\ Pr^{0,42} \quad (1)$$

The coefficient K in equation (1) is a function of the inclination angle X, presented by the values in table 1 for different nozzle distances.

Table 1

X	0	0,087	0,175	0,26	0,5	0,61	0,78	0,87	1,22	1,39	1,57
D_D=10 mm	1,64	1,95	3,07		3,73	3,82	3,57				
D_D=14 mm	1,76	2,22		2,88		3,75		3,89	4,41	5,21	5,82

These values show that a rising heat transfer occur with rising inclination angle. In figure 1 the Nu-number is plotted against the Re-number for an inclination of X = 0. The diagram contains the measuring points of two measuring series with different nozzle distances and some results calculated with the code BETON 3.6. The comparison of the experimental and theoretical results yields, that the calculated values are minimal higher. In figure 2 the temperature fields in the near of a gas liberating wall at X = 0 and X = 0,5 are shown.

To 3.2 Figure 3 shows a typical result of the calculations. The progress of penetration of the melting front into concrete is plotted stepwise as a function of time. Rotational symmetry of the pool is assumed. The parameters describing the state of the melt in time are shown in the diagram above. These are temperature, velocity of the melting front, rate of gas release from the concrete, decay heat, heat flux to the concrete and volume of the pool. The calculations were done by considering a bubble-heat-transfer-model, a homogeneous melt and the variation of the material constants which correspond to the ratio of core material and molten concrete. The calculations are performed with data given by the report RS 154 of KWU for concrete and molten material. The results yield, that the uncertainty in the determination of the melting enthalpy is not very important. But for considering the radiation at the surface, the calculations give a strong dependency for the volume melted within 48 hours as shown in figure 4. The variation of the radiation number ε simulates the not completely known behaviour of the melt at the top. This state may vary from liquid at the top, assuming a very good mixing, to establishing at crust, which drops the radiation number. In the last case the heat transfer to the bottom is correspondingly high.

To 3.3 With the code BETON 3.6 the bubble formations from a horizontal layer with large density differences were calculated. Figure 5 shows a typical example of rising bubbles with a density difference of 1:100. The heat flux to the bottom between the bubbles is increased to Nusselt-numbers of 10 - 12 compared to a linear temperature profile (Nu = 1). It depends strongly on the Laplace-number with constant density difference. (The Laplace-number describes the possible shape of the bubbles, relating the forces of buo ancy and surface tension.) The dependency is smaller with decreasing density difference while the shape of the bubbles is nearly constant by setting the corresponding surface tension. This effect is probably caused by the different mechanisms of material detachment from the bottom layer. This detaching is very quick with increasing density difference, whereas with lower density difference the connection to the bottom is kept for a long time. In this way the influence of the bubbles to the fluid movement at the bottom is longer, which may compensate the lower buoyancy force.

To 3.4 An example of the calculations done with the code BETON 3.5 is shown in figure 6. In this case only in the layer at the bottom internal heat sources occur, while the lighter phase is unheated. In each layer separate temperature and convection patterns can be observed, which are coupled at the phase border. Additional the typical temperature profil is given. To verify the results of the code a heat transfer model for a 2-layer-system was developed and also some experiments were performed. In figure 7 two equations for the heat transport in a 2-layer-system are given. They were developed from the heat transfer model for the case of equal temperatures at the walls. The variable η indicates the heat flux at the lower boundary related to the heat, generated in the lower layer. A comparison of the code and the heat transfer model is shown in figure 8. For the same initial and boundary conditions as in figure 5 the calculations yield, that there is a small variation of η even with a change of the mixed dimensionless number $Ra_2^+ \cdot V_x^{1/0,305}$ up to 10^2. The agreement of the code prediction (BETON 3.5) with the empirical formula (Fig. 7) is good.

To 3.5 The results of the experimental investigations yield, that already by a rather small gas injection rate the movement of the surface prevents a complete crust formation at the top of the molten metal. The melt is cooled down rapidly after onset of the bubble boiling as clearly demonstrated in figure 10 by the curves of the heat flux from the melt to the water. At higher gas injection rates, a violent reaction between water and molten metal accompanied by pressure pulses occur and a strong fragmentation of the metal was observed. In figure 10 a microscopic picture of a fragmentation part can be seen, in which hollow spheres are formed.

7. Next steps

Finish of the final report and continuation of the work in a new RS project.

8. Relation with other projects

RS 154 Investigation of the interaction between molten core material and concrete; KWU-Erlangen 01.02.1975 - 30.09.1976

RS 183 Energy balance after a hypothetical reactor vessel failure under consideration of the concrete destruction; KWU-Erlangen 01.09.1975 - 31.05.1977

9. Reference documents

Quarterly reports in the series: GRS Forschungsberichte (German)

Report period:	Jan. 1978 - March 1978	GRS
	Apr. 1978 - June 1978	GRS
	July 1978 - Sept. 1978	GRS
	Oct. 1978 - Dec. 1978	GRS

Annular report 1977 (German)

10. Degree of availibility

The reports are available at the GRS, Cologne

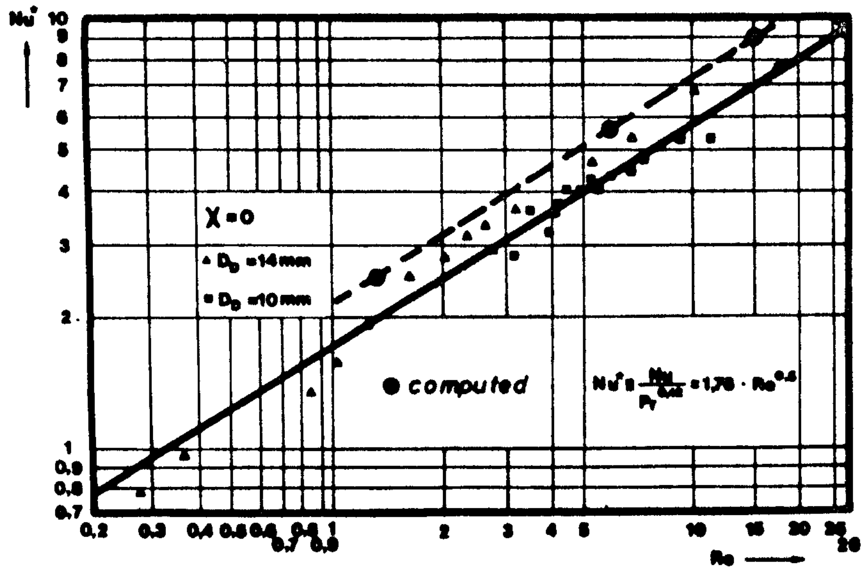

Fig. 1: Mean Nu-number at a gas liberating wall for x = 0

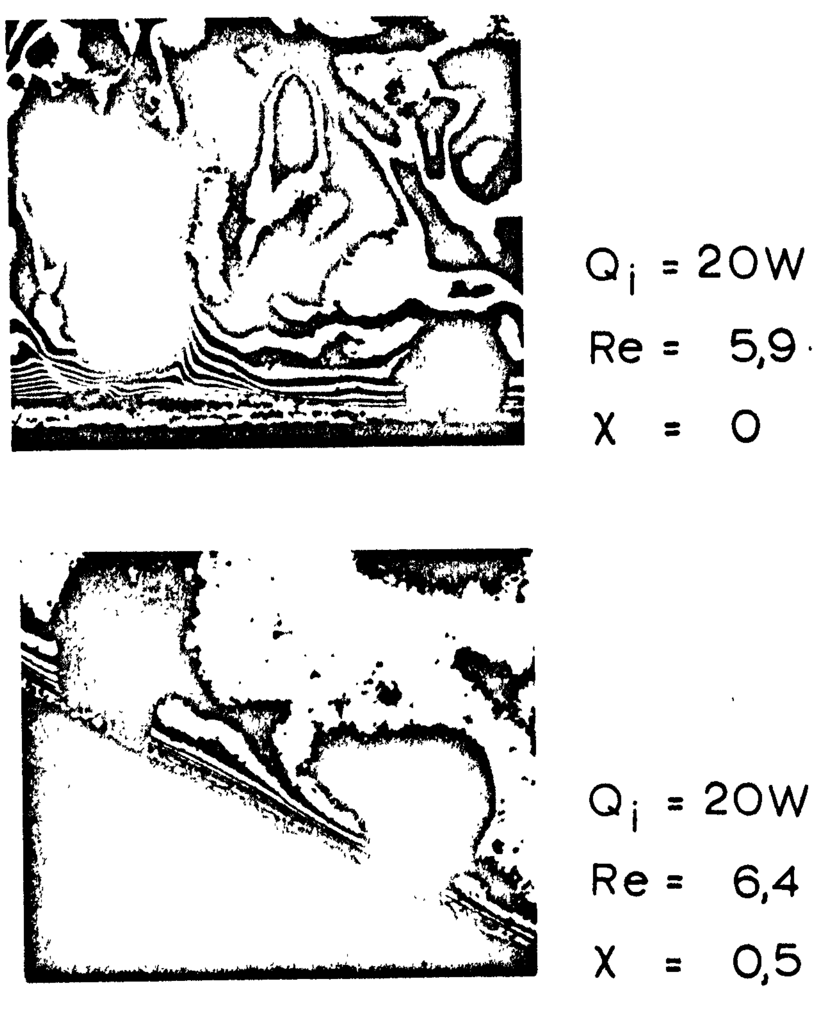

Fig. 2: Temperature field at a gas liberating wall for different inclination angles

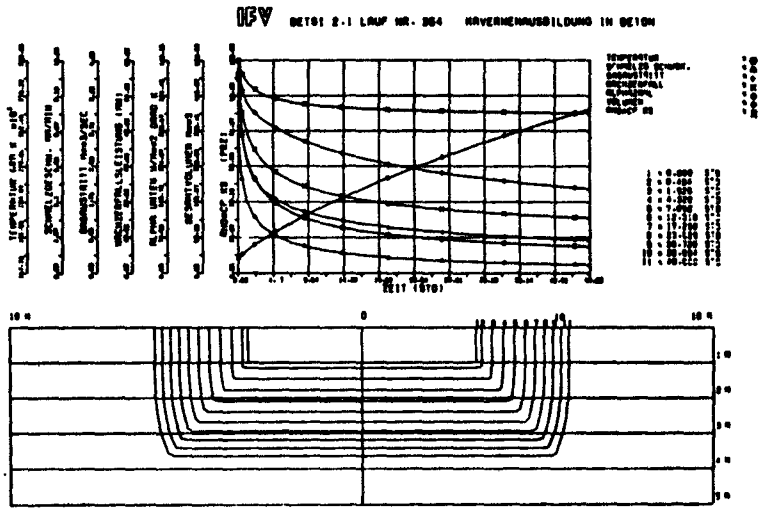

Fig. 3: Penetration of melting front into concrete with time calculated with code BETSI

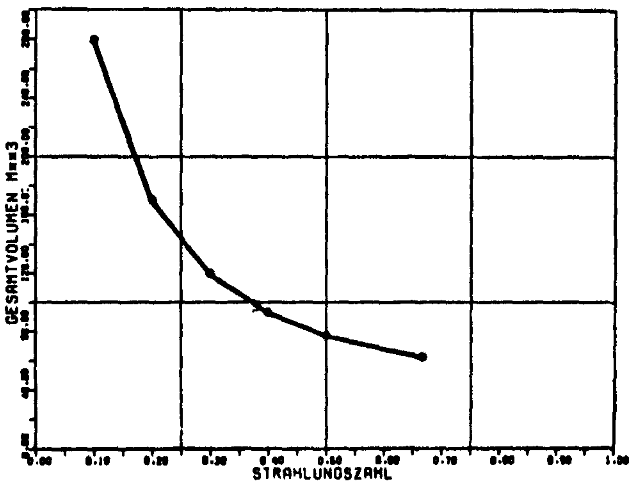

Fig. 4: Volume of the melting pool as function of the radiation coefficient at the top

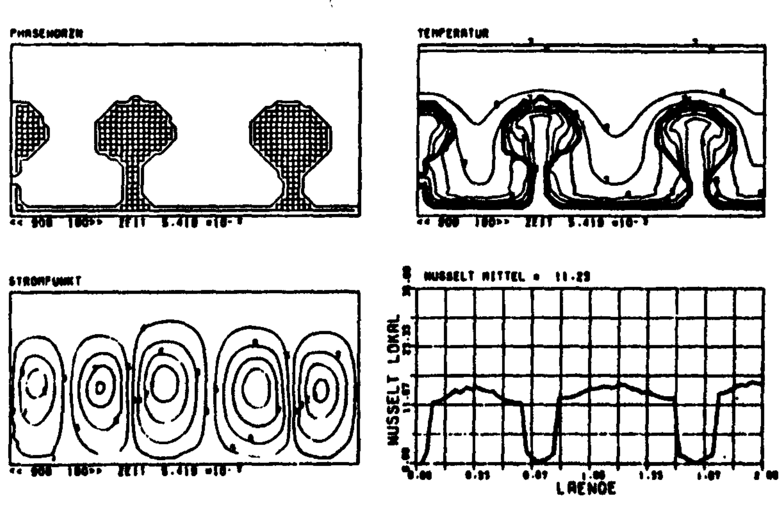

Fig. 5: Isotherms, flow pattern and Nu-number at rising bubbles calculated with code BETON 3.6

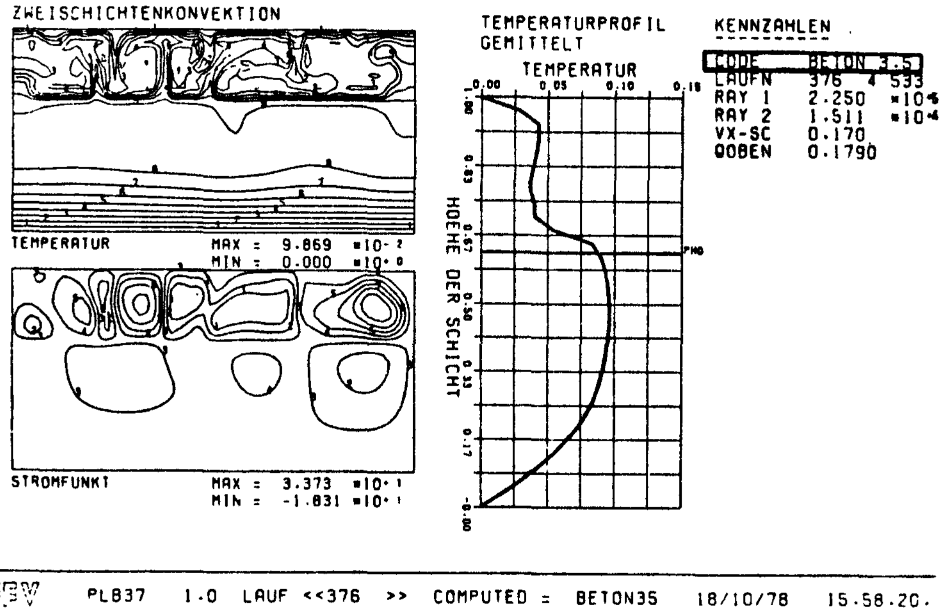

Fig. 6: Isotherms, flow pattern and temperature profile in a 2-layer-system calculated with code BETON 3.5; upper layer is unheated ($Ra'_{lower} = 2{,}25 \cdot 10^5$)

HEAT TRANSFER MODEL FOR 2-LAYER-SYSTEMS
EQUATIONS FOR $\eta = f(Ra'_1, Ra'_2, Ra^*_2, \theta^*, V_x)$

1) T_o — ① Bénard — ② q_{i_1} — T_u

$T_o = T_u \qquad q_{i_2} = 0$

$$(1-\eta) = 0{,}104 \cdot Ra_2^{*\,0{,}305} \left[0{,}5\,\eta^2 - \frac{(1-\eta)^{0{,}87}}{0{,}345 \cdot Ra_1'^{\,0{,}226}} \right]^{1{,}305} \cdot V_x$$

2) T_o — ② q_{i_2} — ① q_{i_1} — T_u

$T_o = T_u$

$$(1-\eta) = 0{,}104 \cdot Ra_2^{*\,0{,}305} \left[0{,}5\,\eta^2 - \frac{(1-\eta)^{0{,}87}}{0{,}345 \cdot Ra_1'^{\,0{,}226}} - \theta^* \right]^{1{,}305} \cdot V_x$$

HEAT TRANSFER MODEL FOR 2-LAYER-SYSTEM
USED DEFINITIONS AND DIMENSIONLESS NUMBERS

$$Ra'_1 = \frac{g \cdot \beta_1 \cdot q_{i_1} \cdot L_1^5}{\nu_1 \cdot a_1 \cdot \lambda_1}$$

$$Ra'_2 = \frac{g \cdot \beta_2 \cdot q_{i_2} \cdot L_2^5}{\nu_2 \cdot a_2 \cdot \lambda_2}$$

$$Ra^*_2 = \frac{g \cdot \beta_2 \cdot L_2^3 \cdot L_1^2 \cdot q_{i_1}}{\nu_2 \cdot a_2 \cdot \lambda_1}$$

$$\eta = 1 - \varepsilon = \frac{q_u}{q_{i_1} \cdot L_1}$$

$$\theta^* = \frac{q_{i_2} \cdot L_2^2 \cdot \lambda_1}{\lambda_2 \cdot q_{i_1} \cdot L_1^2 \cdot 0{,}345 \cdot Ra_2'^{\,0{,}226}}$$

T_o, q_o — ② L_2, T_z, q_{i_2} — ① L_1, q_{i_1} — T_u, q_u

$$V_x = \frac{\lambda_2 \cdot L_1}{\lambda_1 \cdot L_2}$$

Fig. 7: Equations for the heat transfer in a 2-layer-system determined with a heat transfer model

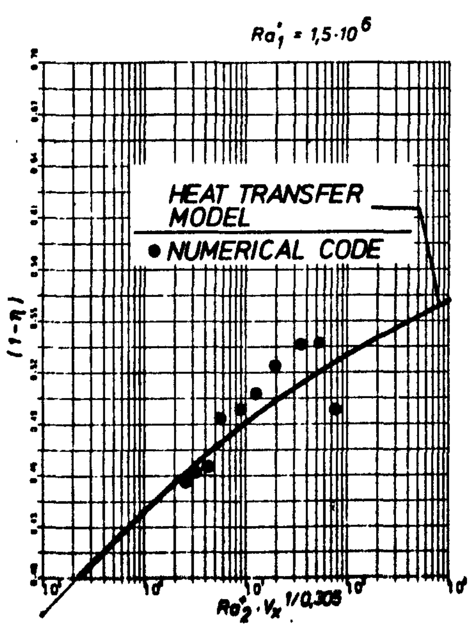

Fig. 8: Comparison of results calculated with the code BETON 3.5 and the heat transfer model, 2-layer-system, upper layer unheated

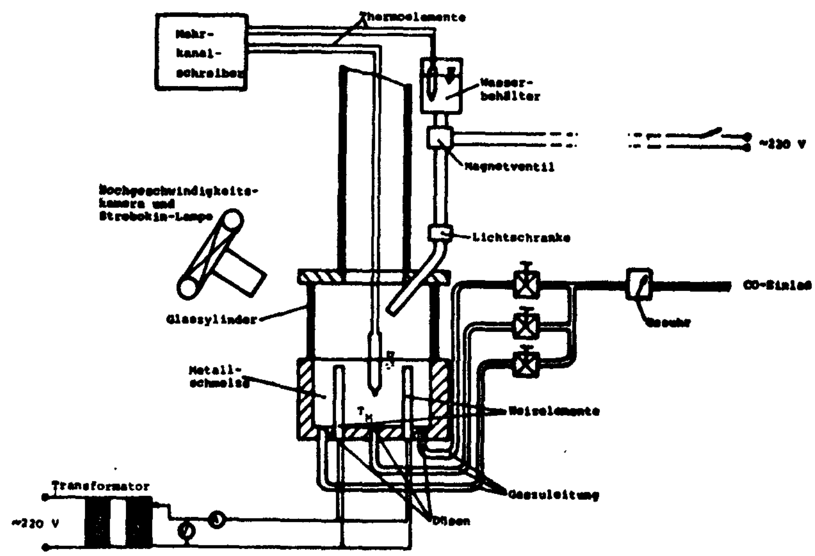

Fig. 9: Principle design of the test chamber used for molten metal-water interaction experiments

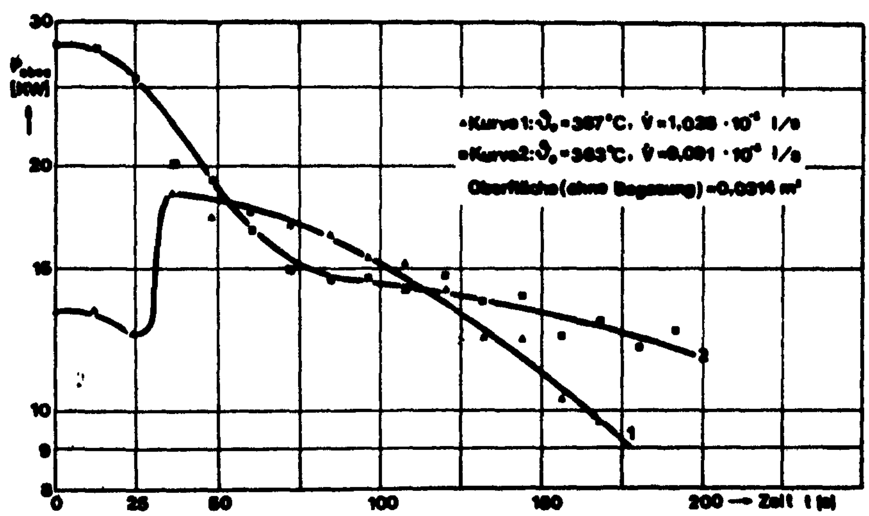

Fig. 10: Heat flux between molten metal surface and water as function of time and gas injection rate

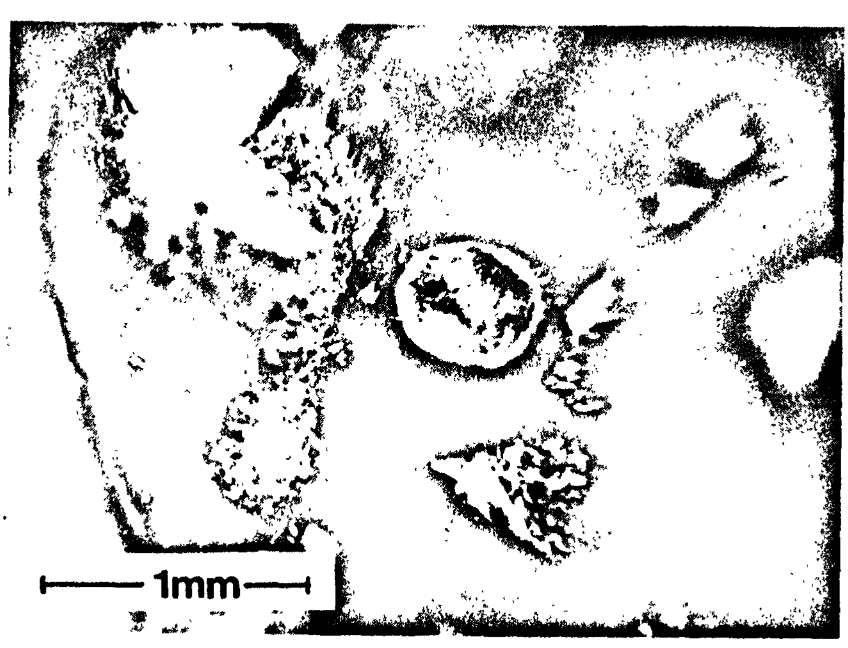

Fig. 11: Microscopic picture of fragmentated metal

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	2.1	RS 214/A

Vorhaben/Project Title	
Performance of Viscosity Measurements on Oxidic Corium-Concrete Melts Durchführung von Viskositätsmessungen an oxidischen Corium-Beton-Schmelzen	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Battelle-Institut e.V. Frankfurt am Main Abt. Werkstofftechnik

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.12.1977	31.3.1979	R. Skoutajan
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

In the last two phases of the hypothetical core melt accident of a light-water reactor, i.e. "RPV heating" and "concrete penetration", the course of the accident depends mainly on the thermohydraulic behavior of the core melt and its interaction with the adjacent structural elements. The viscosity of the melt is an important characteristic for the advance calculation ot the further course of the accident. It should be noted, however, that the viscosity of the melt varies continuously, as a result of both the oxidation of its metallic constituents and the enrichment with concrete components in the accident phase "concrete penetration".

The theoretical estimates of the viscosity known to date in general are satisfactorily accurate only for pure substances of simple structure. It is not yet possible at present to use them for forecasting the viscosity of mixtures of complex composition. Therefore, it is necessary to measure the viscosity of core melts of different compositions.

2. Particular Objectives

Measurement of the viscosity of melts of mixtures of corium (A+R)3 and idealized concrete with the mixing ratios corium : concrete = 10 : 90, 30 : 70 and 50 : 50 (all mixing ratios indicated in wt.-%).

3. <u>Research Program</u>

3.1 Construction of the viscometer measuring systems

3.2 Preparation of the corium-concrete samples

3.3 Measurement of the viscosity of corium-concrete melts

3.4 Metallographic and chemical analysis of the solidified melts

4. <u>Experimental Facilities, Computer Codes</u>

Ball mills for grinding and homogenizing the powdered corium-concrete mixtures.

Hydraulic press for the manufacture of pellets from the powder mixtures.

Sintering furnace with controllable protective atmosphere for temperatures above 1100°C, for carrying out solid-state reactions.

High-temperature rotation viscometer, suitable for operation in vacuum and under protective gas. The viscometer can be operated as Searle or Couette type. The apparatus was developed and constructed under the research projects RS 71 and RS 214.

Metallurgical microscope.
Scanning electron microscope with EDAX attachment.

5. Progress to Date

Re 3.1 The viscometer measuring systems in tungsten, consisting of rotary crucible, shear body and driver were ordered and received. The shear bodies were centered for true running by means of axial centering pieces.

The finished measuring systems were tested by calibration measurements using standardized calibrating oils.

Re 3.2 The FeO which is not commercially available was prepared and examined for phase purity by means of chemical and X-ray diffraction analysis.

The commercial non-stoichiometric UO_2 was reduced and analyzed.

For viscosity measurement, test specimens were prepared from the above described corium-concrete mixtures by solid-state reaction. To this end, the oxides UO_2, ZrO_2, FeO, NiO, Cr_2O_3; SiO, Al_2O_3 and CaO, in proportions corresponding to the desired mixture, were ground for 20 h pressed into pellets and subsequently sintered in a protective gas atmosphere for 20 h at 1000°C and then for 2 h at 1100°C.

Random samples were examined for homogeneous distribution of the components using a microprobe.

Re 3.3 A total of 30 viscosity measurements were made on all the above-described corium-concrete mixtures.

Re 3.4 Microsections were made of the solidified melts; these were examined for changes of the melt due to the experimental conditions(e.g. physical and chemical interaction with the crucible material) by means of light microscopy and SEM.

Chemical analyses of the solidified melts are to provide information about possible changes of the melts during the measurements.

6. Results

Re 3.1 The calibration measurements showed that over-charging of the measuring system and variation of the distance from the lower edge of the shear body to the crucible bottom does not have a measurable influence on the measuring results.

Re 3.2 The test specimens were produced in the desired quality. The examination of the test specimens by means of the microprobe showed a uniform distribution of the constituents after the solid-state reaction.

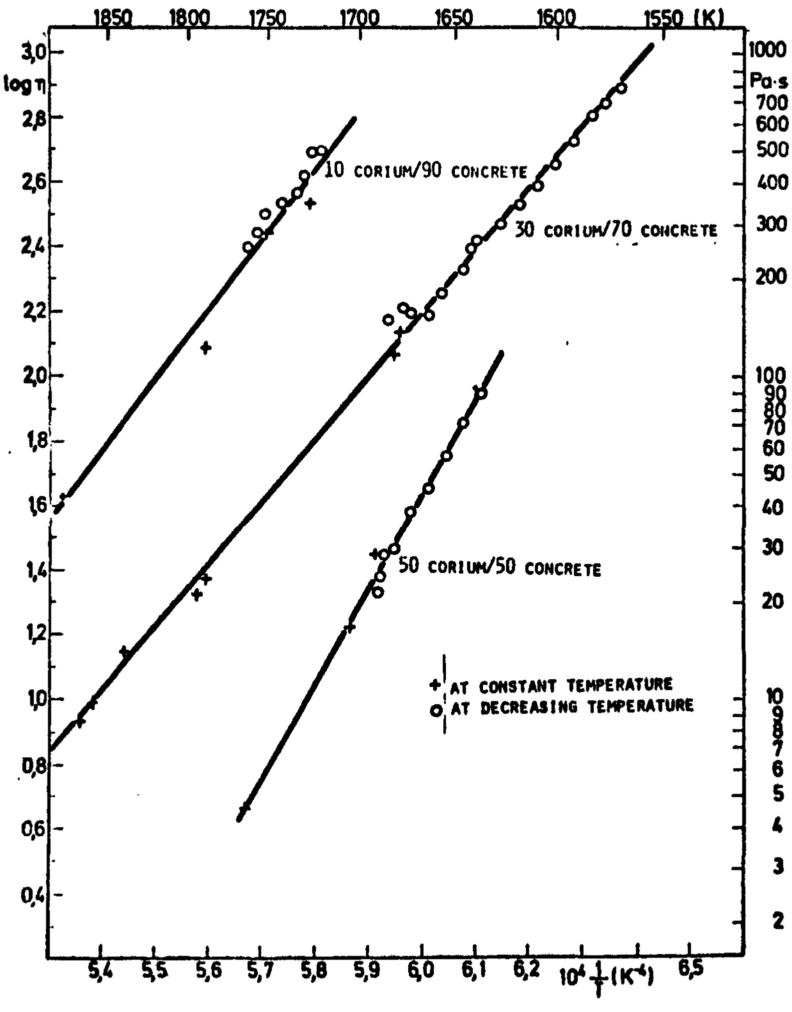

Fig. 1: Viscosity of corium(A+R)3-concrete melts; mixing ratio 30 : 70 wt. %: final result; mixing ratios 10 : 90 wt. % and 50 : 50 wt. %: preliminary results.

Re 3.3 The preliminary results of the viscosity measurements is plotted in Fig. 1.

Re 3.4 The evaluation of the microsections of the solidified melts is not yet fully completed. In the case of the corium : concrete = 10 : 90 and 30 : 70 mixture, no metal precipitation has taken place.
The chemical analysis showed the same composition as had been weighed in for the sample preparation.

7. Next Steps
The investigations will be continued according to schedule.

8. Relation to Other Projects
-

9. References
-

10 Degree of Availability of the Reports
-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	2.1	RS 310

Vorhaben/Project Title	
Emergency Core Cooling Analysis within the Core Meltdown Research Program Kernnotkühlanalyse im Rahmen des Forschungsprogramms Kernschmelzen	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Babcock-Brown Boveri Reaktor GmbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.11.77	31.1.79	Dr. G. Haury
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

'he objectives of the Core Meltdown Project within the Reactor Safety Research Program of the Federal Ministry of Research and Technology so far encompassed the experimental and theoretical investigation of core meltdown by means of a hypothetical accident (under the boundary conditions of the postulated complete failure of all emergency cooling systems in the event of a double-ended rupture of the reactor coolant line of a pressurized water reactor with given remaining water levels in the reactor vessel).

The results of research within the scope of the Core Meltdown Project show that the meltdown of a fuel rod need only be expected when a cladding temperature of ca. 1 850°C is exceeded. However, the design of the emergency core cooling systems according to a RSK guideline shall ensure that the maximum cladding temperature after a loss-of-coolant accident will not rise beyond 1 200°C under any circumstances.

Thus an interesting region whose safety-engineering margins have not been investigated sufficiently to date lies between the region of safe emergency cooling of the reactor core in the event of an accident as defined by the RSK guidelines and the core meltdown which will occur in case of a hypothetical total and simultaneous failure of the multiple redundant emergency core cooling systems.

The region may be described such that it will only be reached in case of a given leakage spectrum in the primary system of the pressurized water reactor if an additional system failure is postulated

within the last available design redundancies.
The starting conditions for hypothetical core meltdown accidents are to be assessed by means of emergency cooling analyses of a primary system characterized by these system failures and be realistic initial conditions.

2. Particular Objectives

The system behavior for two rupture magnitudes (2 A rupture and rupture of approx. NW 100) is to be analized if, starting with the remaining redundancies stated in the licensing procedure, the various emergency cooling systems (core flooding tanks, high pressure and low pressure injection systems) are systematically further reduced or activated with a time lag.

This analysis is to be performed for the BBR pressurized water reactor (1 300 MWe) whose main feature is a two-loop system (2 hot legs, 2 steam generators, 4 cold legs with one pump to each).

3. Research Program

3.1 Preparation of "best estimate" input and code adaptations
3.2 Analysis of a 2 A rupture, reduced number of emergency cooling systems
3.3 Analysis of a rupture NW 100, reduced number of emergency cooling systems
3.4 Analysis of a 2 A rupture, delayed injection from emergency cooling systems
3.5 Analysis of a NW 100 rupture, delayed injection from emergency cooling systems.

4. Computer Codes

The codes below are being used for the analyses:

CRAFT for the thermohydraulic behavior of the reactor cooling system during the blowdown phase

REFLOOD for the thermohydraulic behavior of the reactor cooling system during the refill and reflood phase

THETA 1 B for calculating the transient temperature distribution in a hot channel fuel rod in the course of an accident

CONTEMPT for calculating pressure and temperature in the containment atmosphere.

5. Progress to Date

All projected analyses acc. section 3 have been largely completed.

6. Results

Within the scope of this project is being examined by means of "Best Estimate" emergency cooling analyses to what extent primary side emergency cooling systems may fail or be activated late after a piping break without the core entering a temperature range that will lead to a meltdown of major core regions. In accordance with the agreements reached by the expert group "Core Meltdown" the system behavior after a double-ended break on the pump discharge side in the cold leg and the rupture of a line of nominal width ca. 100 mm was examined.

Starting in each case with the emergency cooling system combinations which for the licensing procedure are assumed as being always available individual systems were further reduced or activated late. The analyses were terminated when either a maximum hot spot temperature of 2200°C was reached or when a long-term steady condition had occured, i. e. when long-term energy removal from and cooling of the core were assured. A case is thus described as being under control in the sense of these investigations if the maximum hot spot temperature in the core does not exceed 2200°C until this long-term steady state is reached.

The analyses yielded the following system availability limit:

- for coping with a double-ended rupture no core flooding tank (pressure reservoir) is needed if the vent valves open. 1 high pressure and 1 low pressure system will suffice in that case.
- Additional delays in the activation time of the remaining LP-system will lead after a few hundred seconds only to core conditions which can no longer be coped with even if a pressure reservoir

is available. Countermeasures for upgrading the systems are not feasible during these times.
- For coping with a leak of NW 100 one high pressure system and one pressure reservoir are sufficient.
- For the last remaining high pressure system the maximum allowable delay is approx. 600 sec.

This proves clearly that system failures exceeding the licensing requirements by far may occur without causing a core meltdown.

7. Next Steps

Is is planned to continue with "best-estimate-analyses" for secondary system failures after a small primary side piping break and after loss of off-site power.

8. Relation with Other Projects

The paramount objectives of the project are identical with those of parallel projects of GRS-K (RS 311) and KWU (RS 306). Starting conditions, leak sizes and scope of work have been coordinated accordingly. The intermediate results of the work part of which was carried out in parallel have been compared by the Core Meltdown Committee of Experts. Further work will be coordinated.

9. References

10. Accessibility of Reports

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	2.1	RS 306

Vorhaben/Project Title	
Best-Estimate-Notkühlrechnungen für hypothetische Ausfallkombinationen der Notkühlsysteme bei DWR's Best Estimate Emergency Cooling Calculation for Hypothetical Failure Combinations of the Emergency Core Cooling Systems in Pressurized Water Reactors	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 11, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 1. 78		Winkler
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 78	

1. General Aim

In the Core Meltdown Research Project investigations of the accident sequence are to be extended from those hypothetical cases where all emergency core cooling systems break down completely and at the same time to partial failures.

2. Particular Objectives

Besides working out details of the theoretical models for a description of the sequence of events in an accident involving total loss of the emergency core cooling systems investigations are to be made as to which partial failures of the emergency core cooling system lead to unacceptable high fuel rod temperatures and/or partial or total core meltdown. With best estimate assumptions it is to be calculated in what cases of partial failure a core geometry results from which residual heat can no longer be removed, thus causing core meltdown in the long run. It is the purpose of those investigations to ascertain whether, under such boundary conditions, core melt paths may arise, the further development and outcome of which differs from any previously investigated core meltdown accidents.

3. Research Program

3.1 Analysis of 2F rupture

3.1.1 Choice of system combinations and sorting them to form groups

3.1.2 Program modification WAK

3.1.3 Undelayed start of injection of ECC systems

3.1.4 Delayed start of injection of ECC systems

3.2 Analysis of a small leak with 100 mm diameter

3.2.1 Choice of system combinations and sorting them to form groups

3.2.2 Undelayed start of injection of ECC systems

3.2.3 Delayed start of injection of ECC systems

3.3 Documentation

4. Experimental Facilities

To perform the calculations of the "core meltdown" project a set of programs is applied at KWU which consists of

- the LECK 4 blowdown computer program
- the WAK2Z refill and reflood program
- the BETHY core heat-up program
- the Small Leak program

5. Progress to Date

To 3.1 Reference case

As a reference case a failure involving a guillotine break of the main coolant pipe between reactor pressure vessel and reactor coolant was examined in which all ECC systems were fully set in operation.

Case 1

Case 1 is marked by the hypothetical assumption that all emergency core cooling systems are lost. This assumption changes the process of depressurization only slightly.

Case 2

Delayed injection into all systems at the moment when 1900 °C has been reached in the core.

Case 3

Delayed injection into all systems in analogy to case 2 at 1540 °C.

Case 4

Failure of all accumulators, 2 low pressure pumps effective without delay.

Case 5

To 3.2 Loss of all low-pressure pumps, all accumulators effective. Leak sizes of 12.5 to 50 cm^2 were investigated.

6. Results

To 3.1 Reference case

On the assumptions made (all systems start functioning, F_Q = 2.0) a figure of less than 680 °C after about 15 sec. is computed as the maximum clad tube temperature.

On account of the use of all reactor safety systems the water level reaches the lower edge of the core after 35 sec. and the axial hot spot in the core centre after 40 sec.

Case 1

The water present in the lower plenum at the end of the blowdown phase is sufficient to initiate zirconium/water reaction in the core if the clad tube temperature rises above approx. 900 °C.

After about 300 sec., the melting temperature is reached in the clad tube at the hot spot in the core (F_Q = 2.0) At these temperatures the energy release by the exothermic zirconium/water reaction already amounts to a multiple (approx. a factor of 4) of the decay power in the fuel. The rod under consideration with the hot spot in the core has burst after approx. 94 sec.

Case 2

Approx. 10 sec. after operation of all systems, core meltdown arises.

Case 3

The maximum clad tube temperature is 1760 °C after about 180 sec.

Case 4

In the core, a maximum of 860 °C is attained after approx. 95 sec.

Case 5

The core meltdown temperature is attained after about 2700 sec.

To 3.2 With leak sizes smaller than 50 cm^2, the core remains covered even without any effective safety injection pumps. With a 50 cm^2 leak, one safety injection pump is sufficient to keep the core flooded. It was assumed in all investigations that shutdown on the secondary side proceeds at a rate of 100 K/h.

7. Next Steps

To 3.1 The following case is to be investigated: Loss of all accumulators, one low-pressure pump effective. The strain behaviour of the fuel rod cladding on reflood is to be allowed for.

8. Relation with Other Projects

RS 310, RS 311

9. References

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10. Degree of Availability

	CLASSIFICATION: 2.1
TITLE (ORIGINAL LANGUAGE): MOLTEN CORE DEBRIS STUDIES - Fuel Fragmentation	**COUNTRY:** UK
	SPONSOR: SRD
TITLE (ENGLISH LANGUAGE):	**ORGANISATION:** UKAEA Culham Laboratory
	PROJECT LEADER: Dr.R.S.Peckover
INITIATED: 1972 / **COMPLETED:** -	**SCIENTISTS:** Mr.T.A.Dullforce
STATUS: Continuing / **LAST UPDATING:** May'79	

General Aim

To understand the mechanism of molten fuel fragmentation in the presence of coolant after core meltdown.

Particular Objectives

To develop satisfactory models of small-scale steam explosions in order to assess

(i) their potential for fuel redistribution outside the core matrix
(ii) their effects on the integrity of core debris containment barriers.

Experimental Facilities and Programme

Small-scale laboratory experiments using simulant materials. The effects of the proximity of the vessel base on the spontaneous triggering of vapour explosions.

Progress to Date

The temperature interaction zone (TIZ) has been mapped out for a number of simulant materials. Necessary conditions for spontaneously triggered steam explosions appear to be that the interface temperature T_I shall (i) exceed the melting point of the fuel (ii) exceed the spontaneous nucleation temperature of the coolant. For the more vigorous interactions a significant dwell time is required. The spontaneous triggering of small vapour explosions appears to be inhibited if the fuel lands on a shaped base prior to the expiry of the normal dwell time.

Relation to Other Projects

This is a part of a program in the UKAEA to understand the consequences of a postulated core meltdown. For other work at Culham Laboratory see entries in section 7.4.1 and 7.4.2.

Reference Documents

1. Buchanan, D.J. and Dullforce, T.A. Fuel-coolant interactions: small-scale experiments and theory. 2nd Specialist Meeting on Sodium-Fuel Interaction in Fast Reactors, Ispra, Italy, November 1973. (Culham Preprint CLM-P362).

2. Buchanan, D.J. and Dullforce, T.A. Mechanism for vapour explosions. Nature, 245, p32 (1973).

3. Dullforce, T.A., Buchanan, D.J. and Peckover, R.S. Self-triggering of small-scale fuel-coolant interactions I; experiments. J.Phys.D: Appl. Phys., 9, p.1295, (1976). (Culham Preprint CLM-P424).

4. Reynolds, J.A., Dullforce, T.A., Peckover, R.S. and Vaughan, G.J. Fuel-coolant interactions - some basic studies at the UKAEA Culham Laboratory. 3rd Specialist Meeting on Sodium-Fuel Interaction in Fast Reactors, Tokyo, Japan, March 1976. (Culham Report CLM-RR/S2/7).

5. Dullforce, T.A. and Rimmer, W. Thermal interactions between Cerrobend and water. Culham Laboratory Report CLM-RR/S2/18, (1976).

6. Dullforce, T.A., Jelphs, A.N. and Rimmer, W. Thermal interactions between Cerrotru and water. Culham Laboratory Report CLM-RR/S2/17, (1976).

7. Dullforce, T.A., Reynolds, J.A. and Peckover, R.S. Interface temperature criteria and the spontaneous triggering of small-scale fuel-coolant interactions. Deutsches Atomform e.V. Reactor Meeting, Hannover, W.Germany, April 1978. (Culham Preprint CLM-P517).

8. Dullforce, T.A. The influence of solid boundaries in inhibiting spontaneously triggered, small-scale FCIs. Fourth Specialist Meeting on Fuel-Coolant Interaction in Nuclear Reactor Safety, Bournemouth, U.K., April 1979.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	2.2	RS 206

Vorhaben/Project Title	
Theorie zur Dampfexplosion in Tankgeometrie, Entwicklung von Fragmentationsmodellen (RS 206 -I.1.5, Jahresbericht A 77), experimentelle Untersuchung stark transienter Siedezustände Theoretical Investigation of Vapor Explosions in Pool-Type Geometry, Development of Fragmentation Models, Experiments Related to Highly Transient Boiling	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Institut für Kernenergetik u. Energiesysteme der Universität Stuttgart

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.5.76	30.6.79	Prof.H.Unger/DI R.Benz
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

Within the frame of the research project investigating core meltdown problems of light water reactors experimental and theoretical calculations on hypothetical vapor explosions in light water reactors are performed. Both activities initiated are intended to lead to a broadening of the knowledge on conditions, course and extend of vapor explosions possibly occuring during a hypothetical core meltdown.

2. Particular Objectives

- Estimation of the upper limits of the energy release and the pressure buildup during a reaction between molten materials and water in reactor geometry under reactor-relevant conditions with respect to specific courses of the accident after the beginning of the meltdown
- Clarification of the conditions and the course of the fragmentation processes which might lead to vapor explosions
- Development of calculational models in order to describe the fragmentation based on relevant physical mechanisms
- Development of a calculational model in order to describe vapor explosions in pool-type geometry
- Theoretical calculations of the experiments performed in pooltype geometry at the EURATOM-research-center Ispra
- Experimental research on highly transient boiling, particularly with respect to the fragmentation of melts
- Experimental research on the trigger conditions for a coherent fragmentation by means of entrapment.

3. Research Program

3.1 Performance of an Investigation Using Engineering Methods

Estimates on upper limits for energy release and vapor pressure buildup during hypothetically postulated vapor explosions in reactor geometry.

3.2 Development of Fragmentation Models

Theoretical investigation of fragmentation models for various materials and reactor conditions. Selection of mechanisms which may occur during core meltdown and development of calculational models in order to describe course and extend of the reaction (e.g. bubble collapse model, shock wave model).

3.3 Development of a Model for Pool-Type-Geometry and Calculation of Experimental Results

Coupling of the fragmentation models within a computer code describing the course of a vapor explosion. Collection of data obtained from the experiments performed in pool-type-geometry, calculation of partial results from experiments, e.g. surface increase of the melt, fragmentation time, pressure distribution.

3.4 Experimental Research on Highly Transient Boiling, Particularly with Respect to Fragmentation

Experiments to the highly transient boiling with large temperature differences will be performed in order to get data to the heat transfer and a picture of the phenomena under these conditions. Included are measurements of the direct contact during and after the stable film boiling.

3.5 Experimental research of trigger conditions for a coherent fragmentation by means of entrapment

Trigger conditions for a coherent fragmentation will be verified by means of entrapment of water in different melts. Temperatures, layer thickness and material of the melt will be varied as well as the temperature and the mass of the water entrapped.

4. Experimental Facilities, Computer Codes

The facilities for subcooled boiling measurements around spheres and for entrapment experiments have been completed. Computer codes to estimate the transient pressure distribution (TRANS), the quasi-static pressure-buildup in reactor geometry (STADR), the fragmentation of materials with low melting point due to vapor bubble collapse (DABKO) and the fragmentation of melts by boundary layer stripping within shock waves (STRIP).

5. Progress to Data

Development of a computer code describing interactions in the Ispra-tank-facility and of a vapor bubble collapse and a boundary layer stripping model. Continuation of measurements of boiling data. Description of heat transfer on subcooled film boiling around spheres. The test facility for entrapment experiments has also been completed.

6. Results

The process of boundary layer stripping alone seems to be too slow to be an important fragmentation mechanism by itself. The computer code TANDEM has been completed. The calculated pressure histories agree well with those obtained by the ISPRA tank experiments.

7. Next Steps

Experiments to transient boiling at free surfaces and to the entrapment as a trigger mechanism for vapor explosions will be continued.

8. Relation with Other Projects

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9. References

R.Benz et al.: Ingenieurmäßige Abschätzung der Energiefreisetzung und des Druckaufbaus bei Dampfexplosionen in Reaktorgeometrie; 1. Techn. Fachbericht BMFT-RS 206, Dez. 1976

R.Benz et al.: Theoretische Arbeiten zur Dampfexplosion 2. Techn. Fachbericht, BMFT-RS 206, Okt. 1978

10. Degree of Availability of the Reports

Available at the GRS, Cologne

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	2.2	RS 296

Vorhaben/Project Title	Land/Country
Experimentelle Untersuchungen des Verhaltens einer im Verlauf eines hypothetischen Kernschmelzunfalles mit dem Sumpfwasser in Kontakt kommenden Kernschmelze Experimental Investigations of the Interaction between Core Melt and Sump Water Spreading on the Surface of the Melt during a Hypothetical Core Meltdown Accident	FRG **Fordernde Institution/Sponsor** BMFT **Auftragnehmer/Contractor** KRAFTWERK UNION AG Reaktortechnik B 222, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 1. 78	30. 9. 79	Dr. Peehs
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 78	

1. General Aim

In the scope of the Core Meltdown Research Program sponsored by the BMFT (Bundesministerium für Forschung und Technologie = Federal Ministry for Research and Technology) it was for the first time that the sequence of events was analysed in the hypothetical case of the cooling of a reactor not being sufficient to remove the entire residual heat output generated in the core. The time dependent distribution of pressure and temperature in a reactor containment was calculated taking into account the results of all the current experimental and theoretical research projects in the Federal Republic of Germany on the subject of core meltdown. According to the present state of perfection and of the knowledge acquired, four characteristic phases are distinguished to describe what exactly happens in the case of an accident.

The first phase includes core heat-up until the supporting structure fails. This phase of the accident begins with the attainment of a specified residual water level in the reactor pressure vessel after blowdown, and it ends with the breakdown of the support grid. The evaporation of the residual water after the supporting structure has failed represents the second phase. After drying-out of the reactor pressure vessel this phase lasts until a melt pool has formed. The third phase comprises the reactor pressure vessel heat-up after a melt pool has formed. When the reactor pressure vessel has melted through

-2-

the melt will get into direct contact with the foundation concrete of the containment. So the destruction of the concrete after the reactor pressure vessel failure is considered to be the fourth phase of the accident.

2. Particular Objectives

The question of whether an over-pressure failure of the containment will occur due to the formation of steam when the core melt gets into contact with the sump water, is closely connected with the knowledge of the above-mentioned processes. As knowledge of the relevant processes can only be improved by experiments, experimental investigations are made into the sump contact of the core melt. The object of such investigations is to make basic data, supported by experiments, available for a better theoretical treatment of the question whether or not there may be an over-pressure failure of the containment due to sump water contact of the core melt.

3. Research Program

3.1 Investigations into the corium-water interaction in cases of direct surface contact (nature of problem: flooding of the core melt by sump water).

3.2 Investigations into the corium-water interaction in cases of lateral contact below the melt surface (nature of problem: water contact with core melt through cooling or ventilation slots in the reactor foundation).

3.3 Investigations into the corium-water interaction in case the melt surface is sprayed (nature of problem: water contact of core melt when spray cooling system is used).

4. Experimental Facilities

The experimental work is largely done in the test installation set up already in the scope of the RS 74 a and RS 154 research projects.

5./6. Progress to Date / Results

To 3.1 - 3.3 The design work was concluded with the design of the medium-frequency coil, the crucibles, the holding devices and various supplementary devices. These parts have been manufactured and are now mostly being assembled.

7. Next Steps

To 3.1 - 3.3 The assembly of the test set-up will be completed. Some initial preliminary check experiments will be started prior to investigate in detail the interaction between core melt and sump water.

8. Relation with Other Projects

Theoretical work: RS 183

9. References

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10. Degree of Availability

Berichtszeitraum/Period 1. 1. 78 - 31. 12. 78	Klassifikation/Classification 2.2	Kennzeichen/Project Number RS 288
Vorhaben/Project Title Dampfentwicklung nach Fluten der Kernschmelze Steam Evolution after Core Melt Flooding		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik RZR 2, Erlangen
Arbeitsbeginn/Initiated 1. 10. 77	Arbeitsende/Completed 31. 12. 78	Leiter des Vorhabens/Project Leader Dr. K. Hassmann
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating 31. 12. 78	Bewilligte Mittel/Funds

1. General Aim

In the R+D project Core Melting, KWU prepares a study on "energy balances after hypothetical RPV failure". In addition to the development of an overall model in co-operation with the TU-Hannover and based on experimental work, a detailed analysis of the penetration of the concrete base interacting with core melt after RPV failure will be carried out. This project completes the work which is being done in RS 183 by a few subject recently found.

2. Particular Objectives

Current results have shown that, following reactor pressure vessel failure, the bottom part of the concrete shield in the reactor cavity, which separates the melt from the sump, will be penetrated after approx. 4 to 5 hours. This is the earliest possible moment the core melt surface can be flooded by the sump water.
This subject will be investigated by TU-Hannover and KWU. TU-Hannover will consider the molten corium pool, i.e. the thermohydraulics. The heat up of the solid structures the integration of the codes to be developed in the overall computer system, and the computer runs will be carried out by KWU.

The following description refers only to the KWU work.

3. Research Program

3.1 The following questions are to be clarified:
Will the innermost concrete shield glide down because of its weight preventing massive evaporation of the sump water? The fact has to be considered that the concrete shield at its outer surface is enclosed by concrete ribs.

3.2 Will the quenching of the melt from above by the sump water result in the formation of a stable solid crust?

3.3 What fraction of the overall heat generated in the melt (decay heat, heat due to chemical reactions and stored heat) will be removed from the upper surface by evaporation of the sump water? What fraction will melt the concrete foundation?

3.4 In what time interval the containment failure pressure will be reached? Will a depressurization of the containment occur before penetration of the concrete foundation?

4. Test Facilities

No test equipment is required for this project.

5./6. Progress to Date, Results

To 3.1 The subroutine RUTSCH, calculating the heating-up and gliding down of the concrete shield, was integrated in KAVERN II and debugged. Simple models have been developed, concerning the thickness of the resudual wall which is affected by a changing distribution of forces.
To establish the resistance of the cylinder wall to subsequent gliding down has shown that the concrete shield fails because of its own weight at a relatively small residual wall thickness. The investigation also has indicated that the concrete shield doesn't jam during the gliding down period. The subroutine OXYD considering the chemical reactions of the metals in the melt with the steam produced by heating up of the concrete was integrated in KAVERN.

To 3.2 Computer runs with KAVERN were made with and without OXYD. The shape of the cavity as well as the moment of flooding the melt by the sump water have been determined. Considering the subsequent gliding down of the concrete shield, massive flooding of the melt upper surface will occur after 111.000 seconds. Since KAVERN requires the containment pressure and the sump water temperature for the calculation of the evapóration rates during the sump water evaporation, the containment code CONZU was integrated in KAVERN II.

To 3.3 Computer runs have shown that immediately after sump water ingression with film boiling conditions the temperature of the melt quickly decreases below its liquidus temperature. Since heat conduction alone is not sufficient to remove the energy released in the melt, the melt will not cool down below its solidus temperature. Therefore, for this low temperature region a model was developed and programmed. Gliding down and melting of the concrete shield before and after flooding can be analysed by RUTSCH. A model for the analysis of sump water evaporation was also developed (subroutine SVERD) and integrated in KAVERN II. This became necessary, since compared with the above mentioned KAVERN II/CONZU code the KAVERN II/SVERD code consumed much less computer time.

To 3.4 Model experiments have shown that, as long as convection in the melt is the predominant mechanism for heat removal, and the temperature of the melt is above its liquidus temperature film boiling can be assumed between the upper surface of the flooded core melt and the sump water. With this assumption a containment pressure of 8 to 10 bar will be reached after about 2 1/2 days at the earliest. The results of computer runs with COCO have shown that, with regard to the subsequent gliding down of the concrete shield, the pressure in the containment increases to 10 bars after about 4 days. With the assumption that the melt is immediately flooded after melting of the concrete shield in its lower region, a containment pressure of 10 bars after about 2 1/2 days was calculated. This time period increases if it is assumed that the sump water and the atmosphere are in thermodynamic equilibrium. The melt has penetrated 5 meters into the concrete in the vertical direction after more than 10 days.

7. Next Steps

The project is completed.

8. Relation with Other Projects

Theoretical Work: RS 183, RS 166

Experimental Work: RS 154, PNS 4331/4332/4314

9. References

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10. Degree of Availability

141.1.09		2-2
TITRE CONTRAINTES RESIDUELLES DANS LES SOUDURES.		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) RESIDUAL STRESS IN WELDS.		Organisme exécuteur CEA/D.TECH Responsable
Date de démarrage 1/79	Etat actuel	Scientifiques
Date d'achèvement 12/80	Dernière mise à jour 1/79	

1. OBJECTIF GENERAL :

Le programme a pour but d'étudier l'importance des contraintes résiduelles dans les soudures de composants importants tels que les cuves de réacteurs à eau pressurisée.

2. OBJECTIFS PARTICULIERS :

Le programme comporte les phases suivantes :

- Mise au point de la méthode de mesure sur le matériau de base et sur le matériau de soudage totalement détensionné.
- Mesure sur petites éprouvettes du matériau de base sous contrainte connue.
- Mesure sur petites éprouvettes contenant des soudures non détensionnées.
- Mesure sur grosses éprouvettes au moyen d'un équipement transportable.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES :

4. ETAT D'AVANCEMENT :

Une étude préliminaire de l'influence des contraintes résiduelles sur la vitesse de propagation des fissures, a montré que des contraintes résiduelles d'amplitude maximum égale à la moitié de la limite élastique du matériau de base pouvaient multiplier par trois la vitesse de propagation de fissure, par rapport à une soudure totalement détensionnée.

141-1-08/4111-03		2-2
TITRE DOMMAGES CONSECUTIFS AUX DEFAUTS DES STRUCTURES ET A L'EVOLUTION DU CHARGEMENT.		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) MECHANICAL FACTORS AFFECTING RELIABILITY OF STRUCTURE COMPONENTS.		Organisme exécuteur CEA/DEMT Responsable
Date de démarrage 1/01/77	Etat actuel en cours	Scientifiques
Date d'achèvement 1/01/80	Dernière mise à jour 1/01/79	

1. OBJECTIF GENERAL

Il s'agit de completer les éléments permettant de s'assurer que les analyses de sûreté des structures mécaniques sont validées, ou tout du moins pessimistes. On se préocupe donc de la naissance, du développement et des conséquences de la présence de fissures. L'effet des conditions PWR (eau chaude sous pression) est examiné sur les différents métaux utilisés. Les résultats déja obtenus sont de nature à préciser les phénomènes et à corriger les valeurs usuellement admises. En outre des essais sur l'effet de la biaxialité sur la fatigue sont prévus , ainsi que des tests globaux sur réservoirs de grande dimension.

2. OBJECTIFS PARTICULIERS.

- Fatigue en milieu PWR.
- Fissuration en milieu PWR.
- Fatigue biaxiale.
- essais sur recipients

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME.

Boucle PRIMEAU (CEN-Saclay)

.../...

4. ETAT DE L'ETUDE :

Les résultats déjà publiés en 1977, ont été complétés en 1978, en particulier sur l'inconel (1) et les aciers austénitiques..

5. PROCHAINES ETAPES :

- Complément sur les aciers austénitiques.
- Influence de la pollution du milieu.
- Vitesse de fissuration milieu PWR.

6. RELATIONS AVEC D'AUTRES ETUDES :

Sans objet.

7. DOCUMENTS DE REFERENCE :

(1) Experimental tests on low cycle fatigue of metals in hot water.

C. GARNIER, G. KOWALCZUK, B. BARRACHIN, R. ROCHE
Specialist Meeting AIEA - INNSBRUCK, AUSTRIA (20-21 novembre 1978).

8. DEGRE DE DISPONIBILITE :

Disponible.

	Classification 2.2
Title 1 FUEL-COOLANT INTERACTIONS (2)	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION CULHAM LABORATORY
Title 2	Project Leader DR T DULLFORCE
Initiated 1972 Completed : Status : Last updating 1976	Scientists:

Description:

1. General Aim

To predict and thus contain the mechanical effects following core melt down.

2. Particular Objectives

To identify and quantify the various phenomena when particular hot and cold liquids are brought into contact.

3. Experimental Facilities and Programme

The work uses gram quantities. Heat transfer regimes and dispersion mechanisms are studies. High-speed cine films (500 frames per sec) have been made and studied. Initially the system molten tin/distilled water has been studied; other materials are planned.

Reference Documents

D Buchanan, T A Dullforce, Nature 245, September 1973. Mechanism for Vapour Explosions.

	CLASSIFICATION: 2.2
TITLE (ORIGINAL LANGUAGE): FUEL-COOLANT INTERACTIONS - DEBRIS SIZE	COUNTRY: UK SPONSOR: SRD
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD UKAEA PROJECT LEADER: G J VAUGHAN
INITIATED: JUNE 1977 / COMPLETED:	SCIENTISTS:
STATUS: CONTINUING / LAST UPDATING: MAY 1979	

GENERAL AIM

Estimate size of particles produced by an FCI so that use can be made of predictions in FCI consequence codes, PAHR studies, aerosol studies etc.

PARTICULAR OBJECTIVES

Theoretical treatment of processes involved in production of debris.

PROGRESS TO DATE

Sample calculations are encouraging giving sizes comparable with experiment.

NEXT STEPS

Encorporation of theory into computer program of FCI mechanism.

RELATION WITH OTHER PROJECTS

See above.

REFERENCES

Some theoretical consideration concerning molten fuel coolant interaction debris size - G J Vaughan
Paper presented at 4th CSNI specialist meeting on fuel coolant interaction in nuclear reactor safety, Bournemouth April 1979

	CLASSIFICATION: 2.2
TITLE (ORIGINAL LANGUAGE): FUEL-COOLANT INTERACTIONS - YIELD CALCULATIONS	COUNTRY: UK
	SPONSOR: SRD
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD UKAEA
	PROJECT LEADER: G J VAUGHAN
INITIATED: OCTOBER 1976 / COMPLETED:	SCIENTISTS: F BRISCOE
STATUS: CONTINUING / LAST UPDATING: MAY 1979	

GENERAL AIM

Obtain theoretical limits for fuel-coolant interactions in various situations and to measure the efficiency of experiment and accidents against a known limit calculation.

PARTICULAR OBJECTIVES

Ability to predict the outcome of fuel-coolant interactions and compare explosion in different systems.

EXPERIMENTAL FACILITIES AND PROGRAMME

Computer code to calculate Hicks-Menzies energy release (HIME).

PROGRESS TO DATE

Code written and some reports. Analytic approximation to Hicks-Menzies calculation carried out.

NEXT STEPS

Continue to analyse experiments and accident situations.

ATION WITH OTHER PROJECTS

This work is related to other parts of the fuel-coolant interaction field aid to containment generally.

REFERENCES

The definition of efficiency for fuel coolant interactions - G J Vaughan and F Briscoe, SRD R99

	CLASSIFICATION: 2.2
TITLE (ORIGINAL LANGUAGE): FUEL-COOLANT INTERACTIONS - EXPERIMENTAL	COUNTRY: UK
	SPONSOR: SRD
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/UKAEA
	PROJECT LEADER: G J VAUGHAN
INITIATED: 1975 / COMPLETED:	SCIENTISTS:
STATUS: Continuing / LAST UPDATING: May 1979	

GENERAL AIM

Investigate possibility of FCIs between thermite produced molten UO_2 and water or sodium.

PARTICULAT OBJECTIVES

Find conditions for FCIs to occur and measure efficiency of interaction.

EXPERIMENTAL FACILITIES AND PROGRAMME

EXPTAL: Experiments in THERMITE rigs A & B at Winfrith.

PROGRESS TO DATE

Experiments have investigated effects of supressing expansion of two phase UO_2 bubble by increasing cover gas pressure or decreasing cover gas volume. It has been shown that a moderate constraint leads to an FCI but that at higher cover gas pressures interactions are inhibited.

NEXT STEPS

...ntinue experiments in water and sodium using larger charges and conditions deemed more likely to produce interactions.

REFERENCE

Fuel Coolant Interaction Studies with Water and Thermite Generated Molten Uranium Dioxide by M J Bird and R A Millington. Paper presented at 4th CSNI Specialist Meeting on Fuel Coolant Interaction in Nuclear Reactor Safety - Bournemouth, April 1979

	CLASSIFICATION: 2.2
TITLE (ORIGINAL LANGUAGE): REVIEW OF MOLTEN FUEL/COOLANT INTERACTION (MFCI) RISKS/CONSEQUENCES FOR PWR (GSD 3.1)	COUNTRY: UK SPONSOR: UKAEA
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/UKAEA PROJECT LEADER: RNH McMILLAN

INITIATED: DECEMBER 1978	COMPLETED:	SCIENTISTS/ENGINEER: GJ VAUGHAN
STATUS:	LAST UPDATING: MAY 1979	

BACKGROUND

Under certain accident conditions - the occurrence of which on present evidence appear to be of a very low order of probability - the fuel/clad structure may achieve melting point temperatures. Subsequent contact by coolant can give rise to an energetic fuel/coolant interaction which can generate explosive pressures which may cause severe structural damage to the adjacent fuel, the core support structure or even failure of the pressure boundary itself.

Recent tests in the US Power Burst Facility (Ref 1) seem to indicate the exsistence of MFCI with UO_2 fuel in pressurised water Experiments at AEEW releasing thermite-produced molten UO_2 beneath pressurised water have demonstrated MFCIs at low ambient pressures (ref 2). Extensive analytical/experimental studies on metal and corium interactions with water are being undertaken in the USA and elsewhere. It remains to be demonstrated that the conditions under which these interactions took place cannot foreseeably occur in the PWR. Conditions under which local failure to cool may occur may be achieved by coolant channel blockage due to debris or clad ballooning.

A modest programme of work is required to investigate the likelihood of such an event. If this yields unacceptable probabilities of occurrence, then analysis and/or experimentation would be required in order to establish consequences.

Previous and current SRD work in connection with the Fast Reactor would be of considerable value for the PWR programme.

OBJECTIVES

1. To review available analytical and experimental evidence in order to determine whether MFCI constitutes a foreseeable risk for PWR.

2. If appropriate to devise and apply methods of evaluation of MFCI consequences

REFERENCE DOCUMENTS

1. USNRC Sixth Water Reactor Safety Research Information Meeting. November 1978. Workshop: Recent Results in Vapour Explosion Research.

2. Bird M J and Millington R A Fuel-Coolant Interaction Studies with Water and Thermite Generated Molten Uranium Dioxide. Paper at 4th CSNI Specialist Meeting on Fuel Coolant Interactions, Bournemouth, April 1979.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	2.3	06.01.11/02A (PNS 4314)
Vorhaben/Project Title Konstitution und Reaktionsverhalten von LWR-Materialien beim Coreschmelzen Constitution and Reaction Behaviour of LWR Materials at Core Melting Conditions		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe GmbH. (KfK) Projekt Nukleare Sicherheit (PNS) - IMF I
Arbeitsbeginn/Initiated 1.1.1974	Arbeitsende/Completed 31.12.1979	Leiter des Vorhabens/Project Leader H.Holleck, A.Skokan
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General aim

Theoretical and experimental investigations on chemical reactions between coremelt, fission products and concrete.

2. Particular objectives

Experimental examination of the chemical interactions in a complex system containing corium+fission products+concrete (chemical reactions and vaporization behaviour) and characterization of those reactions in the sequence of the accident phases with regard to their effects on the course and consequences of a hypothetical core meltdown accident.

3. Research program

Melting experiments upon homogeneous powder samples (corium+fission products + concrete) differing in their composition and in their degree of oxidation, and vaporization tests upon molten samples containing corium+fission products+concrete.

4. Experimental facilities

Laboratory high temperature furnaces (tungsten resistance furnace, induction furnace, electric arc furnace), metallography, ceramography, X-ray diffraction, microprobe analysis, chemical analysis, differential thermal analysis.

5. Progress to date

- The oxidation behaviour of the fission products (esp. Mo and Ru) in preoxidized corium melts interacting with basaltic and limestone concrete was experimentally investigated.

- The solidification behaviour of the metallic and oxide fractions of corium + (basaltic and limestone) concrete melts was experimentally investigated as a function of the degree of oxidation and of the amount of concrete solved in the corium melt.
- Thermoanalytical investigations were performed with concrete of different types ranging from pure basaltic to pure limestone aggregates.
- Studies were carried out on the different phases of a core meltdown accident with regard to the chemical reactions and their significance on the course and consequences of the accident.

6. Results

- The oxidation of Mo (this will be also valid for Tc) was shown to start not before a considerable amount of the Fe present was oxidized to FeO. As it was expected, Ru (as well as Rh and Pd) is not oxidized at the prevailing oxygen potential ($\approx$ -272 kJ/mole O_2) at 2000 K /1/.
- The solidification temperature of the metallic fraction of the corium melt decreases considerably in the course of the oxidation of Zr; from this time onwards it does not change significantly in the course of the oxidation of Cr and Fe.

  The solidification temperature of the oxide fraction of the corium+concrete melt decreases continuously with increasing FeO concentration and with increasing admixture of basaltic concrete. Admixture of limestone concrete results in a reincrease above $\sim$60 w/o concrete. The core melt is therefore expected to solidify sooner with the increasing amount of limestone aggregates in the concrete /1/.
- An exothermic reaction between CaO and SiO_2 was shown to take place just below the melting temperature when concrete with a mixture of basaltic and limestone aggregates was heated up. According to the DTA curve the thermal effect was insufficient to compensate the endothermic effect of the dissociation of $CaCO_3$ /2/.
- The chemical reactions taking place in the consecutive accident phases and their mayor effects on the course and consequences of a LWR core meltdown have been compiled /1,3/.

7. Next steps

- Investigations on the vaporization behaviour of corium melts of different types and in different degrees of oxidation interacting with concrete.

- Melting experiments upon corium constituents with concrete at oxidizing atmosphere and at temperatures between 1775 K and 2275 K.

8. Relation with other projects

This project is connected with the other research projects of the German reactor safety program dealing with core meltdown accidents.

9. References

/1/ A.Skokan, H.Holleck
Chemical Reactions between Core Melt and Concrete and their Effects on the Course and Consequences of a Hypothetical LWR Core Meltdown Accident
ENS/ANS Intern.Topical Meeting on Nuclear Reactor Safety, Oct.16 - 19, 1978, Brussels, Belgium

/2/ M.Peehs, A.Skokan, M.Reimann
Investigations in Germany of the Barrier Effect of Reactor Concrete against Propagating Molten Corium in the Case of a Hypothetical Core Meltdown Accident of a LWR
ENS/ANS Intern.Topical Meeting on Nuclear Reactor Safety, Oct. 16 - 19, 1978, Brussels, Belgium

/3/ H.Holleck, A.Skokan
On the Significance of Chemical Reactions with regard to the Course and Consequences of a LWR Core Meltdown Accident (in German)
Reaktortagung, Deutsches Atomforum, Hannover, 1978, Compacts p. 264

10. Degree of availability of the reports

Unlimited availability.

Berichtszeitraum/Period 1.1. - 31.12.1978	Klassifikation/Classification 2.3	Kennzeichen/Project Number 06.01.12/03A (PNS 4323)
Vorhaben/Project Title Experimente zur Wechselwirkung zwischen Stahlschmelzen und Beton Experiments on the Interacton of Steel Melts and Concrete		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor KfK Projekt Nukleare Sicherheit, Abt.Ingenieurtechnik
Arbeitsbeginn/Initiated 1976	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader D. Perinić
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

1.1 Clear quantification of the physical and chronological development of a core meltdown accident in the fourth phase.

1.2 Clarification under which circumstances a representative melt can be retained in the reactor concrete or in the dry concrete bed, in sand, etc.

1.3 Clarification at which time and under which circumstances serious impacts on the environment have to be anticipated from the core meltdown accident.

1.4 Verification of relevant computer codes and extrapolation to reactor dimensions.

2. Particular Objectives

2.1 Investigations into the concrete (melting bed) destruction influenced by

- the thermohydraulics of the melt,
- the thermal power induced,
- the release of vapor and gas,
- the oxidation of the metallic melt, the formation of ceramic melts, the solubility of core melts and melting bed components,
- the material condition of the melting bed,
- the crust and stratum formation,
- the molten pool depth.

2.2 Verification of computer codes, models and theories by combination of significant parameters.

2.3 Study of the long-term behavior of large simulated core melts.

2.4 If appropriate, investigation into the fission product release influenced by

- the combination of thermohydraulics (inclusive of gas bubbles) and the chemical reactions taking place within the melt,
- the amount of melt,
- the ratio of surface to volume of the melt.

2.5 If appropriate, investigations into the release, transport and behavior of aerosols.

3. Research Program

3.1 Studies on the choice of simulation materials. Comparison of the interaction of corium and thermite melts with concrete.

3.2 Development, construction and operation of the facility for experiments on the interaction between molten steel and concrete, design stage 1 (BETA 1), including measuring systems and building construction.

3.2.1 Concrete crucible thermite experiments on the development of the measuring systems and concrete crucibles.

3.2.2 Development and procurement of the melting facility including the power supply system.

3.2.3 Development and procurement of the measuring system.

3.2.4 Implementation of building construction measures including the licensing procedure.

3.2.5 Trial operation of the BETA 1 facility.

5. Progress to Date

Early in the year the final report on the preliminary project entitled "Experiments on the Simulation of Large Core Melts Interacting with the Concrete of the Reactor" was compiled and submitted to the Core Meltdown Expert Group of the Federal Ministry for Research and Technology (BMFT). Execution of the work proposed was advocated. Development of the BETA 1 test facility was started right away.

Ad 3.2.2 Work on design and planning of the BETA 1 facility was performed.

Ad 3.2.3 Work on the development of the measuring systems was started.

Ad 3.2.4 Planning of the building construction measures was begun.

6. Results

Ad 3.2.2 After discussions with companies experienced in the field a concept for the BETA 1 test facility and the respective measuring systems was elaborated. After harmonization with the groups involved the tendering procedure for the melting facility and the power supply system was carried out. The facility is to produce molten thermite masses of 300 kg of iron and pour them into a concrete crucible. Subsequent induction heating is to maintain the temperatures of the molten pool between 1500 and 2000^{o}C over a period of 15 minutes up to a maximum of 5 hours. The maximum power input to the molten pool is 1.7 MW.

Ad 3.2.3 Drafts of the following measuring systems were completed:

- Immersion probe for remote measurement of the temperature of the molten pool.
- Remote periscope with TV- and cine-cameras for observation and recording, respectively, of the melting chamber.

Ad 3.2.4 In cooperation with the Civil Engineering Department of KfK and with external bureaus preliminary drafts of the experimental hall for the BETA 1 facility have been made and cost analyses carried out. A safety assessment has been compiled.

7. Next Steps

Ad 3.2.1 Preparation of the melting experiments with thermite melts in concrete crucibles.

Ad 3.2.2 Planning and procurement of the melting system including the power supply;

Ad 3.2.3 Planning and procurement of the measuring system.

Ad 3.2.4 Initiation of the licensing procedure for the building construction measures.

8. Relation with Other Projects

06.01.13

9. References

KfK 2600, pp. 463 ... 494

KfK 2700, pp. 4300 - 49 ... 58.

10. Degree of Availability of the Reports

KfK/LA

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	2.3	06.01.13/01A (PNS 4331)

Vorhaben/Project Title	Land/Country
Hydrodynamische und thermische Modelle zur Wechselwirkung einer Kernschmelze mit Beton Hydrodynamical and Thermal Models for the Interaction of a Core Melt with Concrete	FRG
	Fördernde Institution/Sponsor: BMFT
	Auftragnehmer/Contractor: Kernforschungszentrum Karlsruhe Projekt Nukleare Sicherheit IRB

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.1978		Dr. M. Reimann
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

- Quantification of the chronological development of a core melt down accident in the fourth phase.
- Clarification at which time and under which circumstances serious impacts on the environment have to be anticipated from the core melt down accident.

2. Particular Objectives

- Calculations of the cavity shape and the erosion velocity of a core melt penetrating into the reactor basement.
- Calculations of the mass flux and the composition of the released gases and of the resulting pressure increase in the containment.
- Assessment of the transferability of the BETA simulation experiments (06.01.12) on a core melt down accident.

3. Research Program

- Model experiments for separate effects.
- Development of theoretical models describing the separate effects.
- Integration of the models in the WECHSL code describing the fourth phase of a core melt down accident.
- Verification of the WECHSL code by comparison of the calculated results with the BETA simulation experiments.
- Definition of the general aims of the BETA program with regard to code verification.

4. Experimental Facilities, Computer Codes

- WECHSL code (description in 6)
- BEZENT code for the concrete decomposition enthalpy
- FILM code containing the gas film model for parametric studies

5. Progress to Date

- Contributions to the preliminary project compiled in the report "PNS 4323: Experiments on the Simulation of Large Core Melts Interacting with the Concrete of a Reactor".
- Extension of the gas film model to inclined and vertical walls.
- Development of a boundary layer model for calculating the temperature drops in the liquid adjacent to the gas film.
- Development of a bubble rise model.
- Model experiments for checking the theoretical models.
- Implementation of the models in the WECHSL code.
- Development of a geometrical model for the melt front propagation.
- Documentation of the WECHSL code.

6. Results

The WECHSL code includes to date the following physical phenomena and models:

- Quasi stationary concrete decomposition model,
- Extended gas film model with heat transfer by conduction and radiation through the gas film from the pool to the concrete,
- Separated metallic and oxidic layers in the melt,
- Heat transfer model between the layers of the melt and radiation from the top of the melt,
- Bubble rise model and resulting different void fractions in the layers of the melt,
- Chemical reactions of the released gases with the metallic phase,
- Temperature and composition dependent properties of the melt layers,
- Freezing temperatures of the metallic and the oxidic phase depending on composition.

The time dependent pool temperatures as well as the masses and the composition of the two melten layers are determined by mass- and energy balances. The liberated gases percolate through the melt stirring it vigorously. This results in a nearly uniform pool temperature in each layer.

However, as shown by model experiments with dry ice in glycerine in accordance with the results of the boundary layer model, a considerable temperature drop takes place from the pool temperature to the melt surface adjacent to the gas film at liquids with high Prandtl-numbers. This results in a considerable reduction of the total heat transfer. Therefore, at least in the oxidic layer, the liquid boundary layer must be taken into consideration. For liquids with low Prandtl number (i.e. the metallic layer), the influence of the liquid boundary layer on the total heat transfer is low.

The boundary layer model is evaluated at present, it will be prepared for use in the WECHSL code.

Some preliminary thermite simulation experiments were recently carried out at KfK and computed with the WECHSL code. Within the frame of the test conditions (short time tests with strong transient effects), good agreement with respect to cavity shape, temperature drop in the molten layers and concrete decomposition velocity was found.

7. Next Steps

- Further improvements of the physical models in WECHSL,
- Programming of WECHSL in separate moduls,
- Computation of tests within the BETA program (06.01.12)

8. Relations with other Projects

06.01.12 (PNS 4323)

9. References

/1/ Alsmeyer, H., Murfin, W.B., Reimann, M.:
PNS-Halbjahresbericht 1977/2, KfK 2600, Mai 1978, S. 495-513

/2/ Alsmeyer, H. Murfin, W.B., Reimann, M.:
PNS-Halbjahresbericht 1978/1, KfK 2700, Sept. 1978

/3/ Reimann, M., Murfin, W.B.: "Calculations for the Decomposition of Concrete Structures by a Molten Pool", PAHR Information Exchange Meeting, Ispra, Italy, Oct. 10 - 12, 1978

/4/ Peehs, M., Skokan, A., Reimann, M.: "Investigations in Germany of the Barrier Effect of Reactor Concrete Against Propagating Molten Corium in the Case of a Hypothetical Core Melt down Accident", Nuclear Power Reactor Safety Topical Meeting, Brussels, Belgium, Oct. 16 - 19, 1978

/5/ Reimann, M., Murfin, W.B., Alsmeyer, H.: "On the Penetration of Hot Melts into Concrete Structures", European Nuclear Conference, Hamburg, May 6 - 11, 1979

10. Degree of the Availability of the Reports

Unlimited.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.7. - 31.12.78	2.3	0.6.01.13/02A (PNS 4332)
Vorhaben/Project Title Modellexperimente zum Kernschmelzen Simulation experiments concerning core melt down		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KFK PNS/INR
Arbeitsbeginn/Initiated 1.1.78	Arbeitsende/Completed 31.12.79	Leiter des Vorhabens/Project Leader Dr. H. Werle
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Within the framework of the research project core melt down, codes for the analytical description of core melt down accidents are developed. This requires, besides other knowledge, also relations, which describe the heat and mass transfer (including fission products) between the two phases (oxid, metal) of a core melt and from the surface to the atmosphere. These processes are strongly influenced by the gas release from the melt/concrete interaction. With simulation experiments the influence of a gas stream and of other parameters on these processes shall be investigated.

2. Particular objectives

- Description of the heat and mass transport between two liquid layers and from the surface of a liquid to the atmosphere as a function of an induced gas stream and of other governing parameters (thermophysical properties, bubble diameter, local distribution of bubble centers)
- Understanding of the transport processes of dissolved materials (fission products) between the interface of two liquids and at the surface caused by a gas stream.

3. Research program

3.1 Experimental investigation of heat and mass transfer caused by a gas stream a) at the interface between two layers b) at the surface of a layer as a function of
- superficial gas velocity
- material properties (different combination of liquids)
- bubble diameter and distribution of bubble centers.

3.2 Experimental investigation of the transport of dissolved materials caused by a gas stream a) at the interface b) at the surface

- Identification of relevant transport processes
- Identification of governing parameters

4. Experimental facilities

For 3.1

Test chambers have been developed to study heat and mass transfer between two layers in dependance of a gas stream and other parameters.

For 3.2

Methods for mass transfer measurements are studied.

5. Progress to date

For two systems (silicon oil / water and silicon oil / wood metal) the influence of a gas stream on the interfacial heat flux was studied. In the experiments the lower layer was heated, the upper cooled. In this way the initial (oxid heavier and higher power density than metal) respectively the final phase (oxid less dense and lower power density than metal) of the melt/ concrete interaction are simulated.

6. Results

For both systems investigated a gas stream causes a remarkable increase in the interfacial heat flux as compared to the case of pure thermal convection. For a superficial gas velocity of 0.63 cm/s the heat fluxes are increased by factors of 630 and 8 for the silicon oil / water and silicon oil / wood metal systems, respectively. Above a superficial gas velocity of about 0.4 cm/s a strong interfacial mass transfer is observed for the silicon oil / water system. Because in core melt / concrete interaction even higher superficial gas velocities are encountered a remarkable increase of the heat flux between the two layers of a core melt due to released gases has to be expected.

7. Next steps

The investigations concerning the influence of a gas stream on the interfacial heat transfer will be terminated after some further experiments with the system silicon oil / wood metal.

8. Relation with other projects

This work belongs to the subprogram 4300 of the PNS. This subprogram is part of the German research program on reactor safety.

9. References

/1/ H.Werle, PNS-Halbjahresbericht 1978/I

10. Degree of availability of the reports

Unrestricted

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	2.3	RS 183
Vorhaben/Project Title Energiebilanzen nach hypothetischem RDB-Versagen unter Berücksichtigung der Betonzerstörung Energy Balances after Hypothetical RPV-Failure under Consideration of Concrete Decomposition		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik RZR 2, Erlangen
Arbeitsbeginn/Initiated 1. 9. 75	Arbeitsende/Completed 31. 12. 79	Leiter des Vorhabens/Project Leader Dr. K. Hassmann
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 78	Bewilligte Mittel/Funds

1. General Aim

In continuation of the theoretical investigations of the energy balances within the pressure vessel and within the containment (RS 72 a,b), the progression of the melt in the concrete foundation 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. In addition based on the data management system RSYST an overall computer code will be developed. This computer code will contain all single computer codes developed up to now on the project core melt down.

3. Research Program

3.1 Problem related theoretical investigations:
- Analyzation of the available destruction models
- Definition and formulation of the heat transport model.

3.2 Energy balances for the RPV surrounding:
- Analyzation of the region in which contact with the molten core can occur after hypothetical RPV-failure
- 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.

3.5 Integration of the single computer codes developed in the project are melt down into an overall code-system based on the data management system RSYST.

4. Experimental Facilities

No experimental facility necessary.

5. Progress to Date

To 3.2 The geometry-model in KAVERN I was completed, especially the simulation of the corner between bottom and side wall of the melt. The routine for determination of the time steps in KAVERN I with which computer time can be minimized, was developed and debugged.

Contrary to KAVERN I, KAVERN II treats heat conduction into the concrete when calculating the melt-through of the concrete foundation during the core melt/concrete interaction. Test calculations were performed with KAVERN II.

The OXYD program, which calculates the chemical reactions in the melt due to oxidation of the metallic constituents in the melt was modified considering the recently obtained experimental results due to the sequence of metal oxidation. Further test runs were performed with KAVERN II to establish the optimum zone thickness with regard to the heat being conducted into the undestroyed concrete. The results of the heat conduction calculation in KAVERN II were verified by an analytical solution for a plane plate.

To 3.3 The COCO computer code calculates pressure and temperature in the containment. A restart option was incorporated into COCO. A detailed input description was also prepared for this code.

To 3.5 KAVERN I and the modification of COCO were completed. The integration of KAVERN I and BOIL (core heat-up) into RSYST was started. KAUHZ (core heat-up), CONZU pressure and temperature in the containment) and WAVER (residual water evaporation), already existing as moduli in RSYST were submitted to TU-Stuttgart (technical university of Stuttgart) for the development of a time step routine.

The RAUHZ code (which calculates the RPV heat-up after formation of a melt pool) was modified. Computer runs have been performed. Another subroutine was also established by which melting of the RPV and its internals can be plotted as a function of time.

6. Results

To 3.2 KAVERN II test runs with an initial zone thickness of 5 cm chosen for the heat conduction calculation in KAVERN II, resulted in a temperature profile with a thickness of 3 m until sump water ingression after 110000 s. With smaller zones the temperature profile changed drastically. The optimum zone thickness was 2 cm fixed by many computer test runs. Good agreement with the temperature profile determined with an analytical solution for plane plates was reached.

Results obtained by KAVERN II on the concrete penetration during the hypothetical core melt accident (PWR standard plant) have shown that the heat conduction in the undestroyed concrete is negligible until the melt will be flooded by sump water (at the latest 110.000 s from the beginning of the hypothetical melt down accident). The time step interval depends on the heat flow and the zone (thickness between 25 and 450 s for the above mentioned calculations). Also, KAVERN II was modified. In the new version the calculation of the temperature profile in the concrete can be calculated in up to 200 zones. 2 cm zone thickness corresponds to 4 m concrete layer.

To 3.5 RSYST was converted for the new operating system of the CDC/CYBER 176. A new established energy balance considering the physical effects in more detail was integrated in KAVERN I.

Test calculations on data exchange between the computer code sequences KAUHZ/CONZU and WAVER/CONZU were completed and debugged. The input description as well as the print-out of the results of KAUHZ, WAVER and CONZU were modified. The user will be able to have four different data quantities printed out depending on an input parameter.

7. Next Steps

To 3.2 The work on KAVERN I and KAVERN II will be completed during the next quarter. A detailed model- and result description will be given in a technical report. In particular, it is planned to perform comparison calculations with KAVERN I and KAVERN II, the long-term influence of heat conduction on the calculated results.

To 3.5 Modification of BOIL, KAVERN I and RAUHZ in RSYST is to be completed. Furthermore, it is planned to start the integration of COCO and KAVERN II.

8. Relation with Other Projects

RS 154, RS 166, RS 288, RS 237, PNS 4331/4331/4314

9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.10.1978 - 31.12.1978	2.3	RS293
Vorhaben/Project Title Theoretical Analysis of the Heat Effects in the Core Melt-Concrete Interaction Wärmeeffekte bei Kernschmelze Beton - Wechselwirkung		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Battelle-Institut
Arbeitsbeginn/Initiated 1.11.1977	Arbeitsende/Completed 31.1.1979	Leiter des Vorhabens/Project Leader Poeschel
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Theoretical determination of the heat effects occurring as a result of the core melt-concrete interaction in the fourth phase of the hypothetical core melt accident.

2. Particular Objectives

Determination of the heat effects as "integral melt enthalpy" by breaking down the overall reaction into single reaction steps and theoretical analysis of the contribution of each step.

3. Research Program

3.1. Breaking down the overall reaction "core melt-concrete interaction" into single reaction steps

3.2. Theoretical determination of the Δ H value of each individual reaction step

3.3. Investigation into the oxidation-reduction behaviour in the core melt-concrete system

4. Experimental Facilities, Computer Codes

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5. Progress to Date

Re 3.1. The literature study on the state of the art in the cement industry relevant to the problem under consideration was completed.

Those reaction steps were defined whose caloric effects have to be taken into account.

Re 3.2. The caloric effects of the reaction steps defined in 3.1 were calculated.

Re 3.3. The calculation of free reaction enthalpies and of the equilibrium gas composition for the reactions of the "corium" metals with the water vapor originating from the concrete was completed.
Compilation of the results in a final report was started.

6. Results

Re 3.1. The reaction steps to be considered were described already in the last quarterly report (July 1 through September 30, 1978).
The result given in that report for the solid-state reaction between free CaO and SiO_2 has to be corrected in so far as it does not yield calcium orthosilicate, but wollastonite, which is due to an excesss of SiO_2.

Re 3.2. All the reaction steps considered in 3.1 are endothermic with respect to their caloric effect, with the exception of the steps of wollastonite formation and oxidation of the reinforcing steel, provided that the final product is Fe_3O_4.

Re 3.3. The chronological order of the reactions (provided that the stability of the oxides and thus the free reaction enthalpies are taken as criterion, as agreed) was found to be as follows: the zirconium metal reacts first, followed by chromium, then iron, and finally nickel.
It should be noted that the caloric effects involved in the zirconium and chromium oxidation are strongly exothermic; the oxidation of iron (by means of water vapor) is an endothermic reaction as long as FeO is formed rather than Fe_3O_4. The same applies to the oxidation of nickel with formation of NiO.

7. Next Steps

The results will be compiled in a final report.

8. **Relation to Other Projects**

The investigations are being coordinated with project RS 154.

9. **References**

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10. **Degree of Availability of the Reports**

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Berichtszeitraum/Period 1. 1. 78 - 31. 12. 78	Klassifikation/Classification 2.3	Kennzeichen/Project Number RS 295
Vorhaben/Project Title Wechselwirkung der Kernschmelze mit erweitertem Fundamentbereich Interaction of Core Melt with Extended Foundation Region		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik B 222, Erlangen
Arbeitsbeginn/Initiated 1. 12. 77	Arbeitsende/Completed 30. 9. 79	Leiter des Vorhabens/Project Leader Dr. Peehs
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 78	Bewilligte Mittel/Funds

1. General Aim

To provide a risk analysis for LWR-accidents of very low probability, the research project "Kernschmelzen" of the BMFT was started in the FRG in 1971. The objective of the related R and D-work is to study the consequences of a hypothetical core melting accident. This may occur if we hypothesize that the emergency core cooling system fails completely after a loss-of-coolant accident. Performing this analysis, one can define the chronological order in the sequence of a core melting accident:

1) the heating up of the core until the core structure may fail; which starts at a certain water level in the core and ends with the failure of the grid plate,
2) the second phase is characterized by the evaporation of the water left in the lower plenum and it lasts after the dryout of the pressure vessel until a molten core debris is formed,
3) the third phase is concerned with the heating up of the pressure vessel after the melt was formed and
4) in the fourth phase after the pressure vessel failure, the molten Corium will interact with the concrete structure beneath the pressure vessel,
5) finally after the penetration of the concrete foundation of the reactor the molten Corium plus the molten Concrete starts to interact with the materials just beneath the concrete structure.

2. Particular Objectives

Subsequent to penetration of the bottom concrete layer the hot core melt will directly contact the geological strata found underneath the concrete reactor foundation. However, it is not realistic to assume that, when the melt reaches these strata, an instantaneous release of the core melt fission product inventory occurs. It should rather be asumed that the extended foundation similar to the concrete, will be heated up by the core melt, causing a series of melting processes, which again leads to a crust formation between the melt and the geological strata.

This crust should now prevent in some extent the propagation on fission product into the geological strata beneath the concrete foundation and into the ground water.

It is the objective of the R+D-work to give an experimental base for theoretical modelling of the 5th phase of a hypothetical core melt accident.

3. Research Program

3.1 Definition of the status of molten corium after penetration of the reactor foundation.

3.2 Compilation of the status of the extended foundation region interacting with the molten corium.

3.3 Literature study on the performance of typical natural aggregate material and geological strata.

3.4 Determination of experimental boundary conditions.

3.5 Testing of the available experimental techniques to cover the experimental requirements.

3.6 Test of the pure aggregate material

3.7 Investigation of core melt/mineral aggregate interaction

3.8 Test evaluation in order to prepare a core melt/extended foundation interaction model

3.9 Data verification by means of RS 183 computer program in order to provide an estimate of the extended foundation penetration.

4. Experimental Facilities

Experimental work will be performed with the equipment already installed for RS 74 a and RS 154, i.e. melt quantities of 1 to 2 kgs will mainly be used. The max. melt quantity, however, will be increased up to 5 kgs, if possible.

5. Progress to Date

To 3.5 Melt test details were completed. For evaluation of the test technique, only UO_2-free melts could be used. Three test series were performed as follows:

- Corium - single components/sand - dry
- Corium - single components/ sand + concrete portion-dry
- Corium - single components/ sand + concrete portion -moist

To 3.6 The reaction experiments with moist sand were performed with different sand/water ratios (10/1 to 10/4).

To 3.7 The reaction tests were performed - as is shown in the attached table - in the following test series:

- Corium	- single	components/gravel	+ concrete	shell	-dry
- "	- quadruple	components/gravel	+ "	"	-dry
- "	- "	" /clay	+ "	"	-dry
- "	- "	" /limestone	+ "	"	-dry
- "	- single	" /sand - moist			
- "	- quadruple	" /clay - moist			
- "	- A + R/sand - dry				
- "	- " "/gravel - dry				
- "	- " "/clay - dry				
- "	- " "/sand - moist				

6. Results

To 3.5 The chosen test technique was successfully applied (arc heating, concrete container for reception of the samples, off-gas via an ice condenser with subsequent H_2 oxidation).

To 3.6 A first check of the tests performed showed that the melt propagation rate is distinctly retarded in moist sand.

To 3.7 The current tests revealed tightly adhering crusts of approx. 5 to 20 mm thickness around the metallic Corium single components.
A strongly foaming oxide layer showing in part very large gas bubbles formed on the melting bath surface.

7. Next Steps

To 3.7 The test series are being continued

To 3.8 The tests already performed will be evaluated in detail.

8. Relation with Other Projects

RS 154, RS 183

9. References

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10. Degree of Availability

146-2-03		2-3
TITRE ETUDE DE L'INTERACTION CORIUM-BETON. (REACTEURS A EAU ORDINAIRE SOUS PRESSION)		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) CONCRETE-CORIUM INTERACTION (PWR)		Organisme exécuteur CEA/DTCE Responsable
Date de démarrage 7/78	Etat actuel	Scientifiques
Date d'achèvement 12/80	Dernière mise à jour 1/79	

1. OBJECTIF GENERAL :

Les accidents hors dimensionnement des réacteurs à eau provoquant la fusion du coeur conduisent vraisemblablement à la traversée de la cuve par le coeur en fusion, et au déversement dans le puits de cuve d'un mélange fondu combustible, gaine et matériaux de structure. Ce corium interagit avec le béton provoquant sa décomposition chimique, entraînant un dégagement de CO_2 et de vapeur d'eau.

2. OBJECTIFS PARTICULIERS :

Le STT se propose d'étudier la cinétique de décomposition du béton et donc sa vitesse de pénétration. Une étude bibliographique a été entreprise dans le deuxième semestre 78. Cette étude sera poursuivie en 79 et sera complétée par une étude portant sur la caractérisation des bétons utilisés par EDF dans ses différentes centrales et portera sur les domaines suivants :

- Mécanisme et cinétique du dégagement des produits de fission.

- Influence des produits de l'interaction corium-béton (CO_2, H_2O) sur l'entraînement des produits de fission.

- Vitesses de pénétration, horizontales et verticales du corium dans le béton.

141-3-01/4111-10		2-3
TITRE Programme Européen de contrôle de tôles fortes par ultrasons.		Pays FRANCE Organisme Directeur CEA/DgCS/DSN
TITLE (Anglais) European Program of Ultrasonic testing of heavy section steel plates		Organisme exécuteur CEA/DTECH/STA/SCND Responsable
Date de démarrage 01/03/77	Etat actuel en cours	Scientifiques
Date d'achèvement ✱ 15/01/79	Dernière mise à jour 12/78	

1. OBJECTIF GENERAL

Participation à un programme Européen de contrôle de tôles de forte épaisseur comportant des soudures avec défauts volontairement introduits. Action entreprise dans le cadre des communautés Européennes avec des tôles mises à disposition des laboratoires européens par les USA dans le cadre du programme HSST.

2. OBJECTIFS PARTICULIERS

1) Réalisation d'une cuve d'immersion capable de recevoir les tôles à contrôler Cette cuve est équipée d'un dispositif d'enregistrement cartographique "en tout ou rien" disponible au laboratoire de la STA et susceptible d'effectuer des cartographies "C-SCAN" à l'échelle 1.

2) Réalisation des traducteurs focalisés.
La participation du CEA a été dès l'origine limitée à l'utilisation des traducteurs focalisés développés à la STA.
Au total 11 traducteurs ont été utilisés.
à 2 MHz : 4 en ondes transversales / 3 en ondes longitudinales) Ø tache focale 5mm environ
à 4 MHz : 2 en ondes transversales / 2 en ondes longitudinales) Ø tache focale 2,5mm environ

plusieurs autres traducteurs avec correction de courbure ont été réalisés pour le contrôle de la soudure du piquage.

✱ *Date modifiée pour tenir compte des retards*

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

L'installation comporte la cuve réalisée conformément au § précédent, l'ensemble des traducteurs et l'équipement ultrasonore du laboratoire de la STA. Le programme comporte la détection des défauts dans la soudure de trois tôles dont une soudure de piquage, leur localisation et leur caractérisation, notamment leur dimensionnement. Pour ce faire la méthode mise au point au CEA en parallèle avec le développement des traducteurs focalisés est utilisée. Elle permet le dimensionnement avec une précision égale au diamètre ultrasonore au foyer.
A l'issue des essais non destructifs réalisés tant à Saclay que dans les autres laboratoires français et européens, le programme prévoit la découpe des tôles et l'examen macro ou micrographique des défauts identifiés, afin de déterminer le degré de corrélation
- dans la détection des divers types de défauts
- dans leur localisation
- dans leur caractérisation, notamment leur dimensionnement

Il est à noter que le programme standard comporte normalement un contrôle en manuel suivant une procédure voisine de celle du Code ASME section XI et proposée par le PISC (Plate Inspection Steering Committee) organisme mis sur pied dans le cadre des Communautés Européennes et responsable du programme européen.

4. ETAT DE L'ETUDE

En ce qui concerne la participation CEA des problèmes sont apparus. En effet les tôles fournies dans le cadre du programme HSST pour ce programme européen se révèlent être de très mauvaise qualité et le nombre de défauts présents dans les soudures et hors des soudures et très important. Ceci est vrai pour les deux tôles planes repérées 50-52 et 51-53. Pour la tôle 204 qui comporte un piquage le nombre de défauts relevés est relativement faible et, de ce fait le dépouillement a été fait.

En ce qui concerne les deux autres tôles, la méthode de dimensionnement mise au point par le CEA conduit à un dépouillement extrêmement complexe et long. (défauts vus suivant 5 directions différentes, parfois compte tenu de leurs dimensions par 3 traducteurs simultanément pour chaque direction).
Il a donc été admis que la participation du CEA se bornerait au dimensionnement d'un défaut susceptible d'intéresser les autres participants.

5. PROCHAINES ETAPES

- Transmission des résultats au PISC (Plate Inspection Steering Committee)
- Participation aux réunions relatives au programme d'essais destructifs de corrélation.
- Eventuellement participation à l'élaboration du rapport au niveau du PISC.

6. RELATIONS AVEC D'AUTRES ETUDES

Ce programme entre dans le cadre des études de justification de la méthode utilisant les traducteurs focalisés et de leur aptitude au dimensionnement des défauts décelés.

7. DOCUMENTS DE REFERENCE

PISC/UK (76) P1
Fiche réf. /H5 / GR SR / 100 12 13

"Procedure for ultrasonic examination of P.V.R.C. welded test blocks"

Révision 2 Novembre 1976.

141 - 3 - 02/4111-10		2-3
TITRE Détectabilité par ultrasons des défauts en compression		Pays FRANCE Organisme Directeur CEA/DgCS/DSN
TITLE (Anglais) Ultrasonic detectability of flaws under compression		Organisme exécuteur CEA/DTECH/STA Responsable
Date de démarrage 01/07/77	Etat actuel en cours	Scientifiques
Date d'achèvement ✗ 30/06/79	Dernière mise à jour 12/78	

1. OBJECTIF GENERAL

Etude de l'influence du champ de contraintes sur la détectabilité des défauts lors d'un contrôle par ultrasons.

2. OBJECTIFS PARTICULIERS

1) Réalisation des éprouvettes
Elles sont réalisées dans le cadre d'une autre étude.
Il s'agit d'éprouvettes de fatigue en traction-compression, comportant en leur milieu une soudure réalisée volontairement avec un défaut de type déterminé. Les défauts sont décelés au stade de la fabrication et dimensionnés tant par radiographie que par ultrasons.

2) Réalisation des traducteurs focalisés
Les 4 traducteurs focalisés (Ø tache focale = 2mm) ont été réalisés : 2 à 45° en ondes T ; 1 à 60° en ondes L. Les défauts doivent se situer à mi-épaisseur soit 80mm.

3) Le personnel du STCAN (Service Technique des Constructions et Armes Navales) qui doit se charger des contrôles en cours d'opération a subi un stage de formation au laboratoire de Saclay pour s'initier à l'utilisation des traducteurs focalisés.

3. INSTALLATION EXPERIMENTALE :

Voir à ce sujet la fiche "Propagation par fatigue des défauts naturels des soudures"

✗ Date modifiée pour tenir compte des retards

4. ETAT DE L'ETUDE

Les contrôles en usine des éprouvettes sont maintenant terminés et ont donné lieu à un compte rendu de la part de Creusot-Loire.
Un second contrôle a été réalisé par la Section Contrôles non destructifs du Service des Technique Avancées de Saclay. Il fait également l'objet d'un compte rendu (ces essais ont été réalisés par immersion avec l'équipement standard STA/SCND).

Il était prévu que les dimensionnements des défauts en cours d'évolution seraient effectués grâce à un dispositif manuel (pantographe) mis au point par STA/SCND. Les premiers essais ont montré , compte tenu des caractéristiques de réponse de l'équipement utilisé, que la reproductibilité n'était pas suffisante et ne permettait pas d'assurer la précision recherchée.

L'équipement STCAN (machine de traction-compression) a donc été modifié pour permettre l'introduction d'un dispositif de contrôle par immersion. Cette modification est terminée et l'ensemble est en cours de mise au point.

5. PROCHAINES ETAPES

Il s'agit essentiellement des essais de fatigue dans les laboratoires de la marine :
Chaque période sera précédée et suivie d'un dimensionnement des défauts.
La partie de ce travail intéressant cette étude sera la comparaison du dimensionnement en l'absence et en présence d'un champ de contraintes de compression.

6. RELATIONS AVEC D'AUTRES ETUDES

L'essentiel des essais de fatigue sera effectué dans le cadre d'une autre étude.

7. DOCUMENTS DE REFERENCE

- Contrat avec la Société Creusot-Loire SA -6416- "Réalisation et Contrôle d'éprouvettes comportant des défauts de soudures".

- Contrat avec le STCAN -SA - 6417- "Essais de fatigue sur grosses éprouvettes comportant des défauts de soudures".

	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.

contd.....

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

	CLASSIFICATION: 3.1
TITLE (ORIGINAL LANGUAGE): Rete di rilevamento sismico	COUNTRY: ITALY
	SPONSOR: ENEL
TITLE (ENGLISH LANGUAGE): Seismic monitoring network	ORGANISATION: ENEL
	PROJECT LEADER: F. CAPOZZA
INITIATED: 1973 / COMPLETED:	SCIENTISTS: A. BERENZI
STATUS: In progress / LAST UPDATING: June 1979	

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 consists of 168 monitoring points distributed in the whole Italian territory with the exception of Sardinia.
Each monitoring point is 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 "CNEN-ENEL Commission on Problems Associated with the Installation of Nuclear Plants".

5. Next steps

Further recording will be necessary to obtain the definition of reference earthquake.

142-1-08 121-1-01/4113-01		3-1
TITRE Signaux sismiques synthétiques.		**Pays** FRANCE
		Organisme Directeur CEA/DgCS
TITLE (Anglais) Synthetic seismic signal studies.		**Organisme exécuteur** CEA/DEMT
		Responsable A. SOKOLOVSKY - DSN M. LIVOLANT - DEMT
Date de démarrage 1.1.1975	**Etat actuel** en cours	**Scientifiques** F. JEANPIERRE - DEMT
Date d'achèvement 31.12.1978	**Dernière mise à jour** 1.12.1977	DSN/M.SOKOLOVSKY

1. OBJECTIF GENERAL

Constructions de signaux temporels synthétiques vraisemblables, avec des spectres réguliers qui serviraient de standard de vérification de la tenue des structures aux séismes.

2. OBJECTIFS PARTICULIERS

1. Analyse détaillée d'enregistrements sismiques disponibles.
2. Mise au point de spectres réguliers.
3. Calculs de validation.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

4. ETAT DE L'ETUDE

1. Avancement à ce jour :

Une analyse détailée des accélérogrammes de séismes réels a été faite ; le but de cette étude est de dégager les paramètres caractéristiques de ces accélérogrammes (durée, évolution du contenu spectral au cours du temps, etc) de façon à construire des séismes synthétiques vraisemblables.

2. Résultats essentiels :

L'étude est terminée pour le cas du séisme de SAN FRANCISCO.

Un séisme synthétique de caractéristiques semblables mais avec des spectres réguliers a été mis au point. Une série de spectres-types a été mise au point à partir de quelques séismes américains.

121-1-02/4172-10

3-1

TITRE	Pays
EXPLOITATION DES DONNEES SISMOTECTONIQUES DE LA FRANCE	France
	Organisme Directeur CEA/DSN
TITLE (Anglais) APPLICATION OF SEISMOTECTONIC DATA OF FRANCE	**Organismes exécuteurs** BRGM CEA/LDG
	Responsables J. FAURE A. LEVRET

Date de démarrage	Etat actuel	Scientifiques
01:01.1979	Démarrage	MM. GODEFROY BRGM VOGT BRGM MASSINON LDG
Date d'achèvement 31.12.1982	**Dernière mise à jour**	

1. OBJECTIF GENERAL

Préciser le cadre sismotectonique des sites pour conduire à une évaluation réaliste du niveau de sismicité:

- définition des données géologiques (provinces, ensembles tectoniques, accidents),
- détermination des paramètres physiques des séismes,
- prise en compte des phénomènes naturels induits par les séismes.

2. OBJECTIFS PARTICULIERS

Collecte des données (homogénéisation des données de base pour réduire les marges d'incertitude).

- Données géologiques (pour mémoire: l'essentiel a été rassemblé pour la carte sismotectonique); des compléments sont à réaliser.
- Données de la sismicité historique:
  . Réalisation d'un fichier par le BRGM.
- Caractéristiques au foyer: magnitudes, profondeurs.
  . Mise au point d'un système de rassemblement des données.
  . Dépouillement et interprétation.
  . Intégration des données du LDG.

.../...

Synthèse et corrélation.

- Définition d'ensembles tectoniques homogènes.
- Etudes de lois d'atténuations locales.
- Définitions de provinces et d'ensembles sismotectoniques.

Prise en compte des phénomènes induits par les séismes.

- Recensement des risques (tsunamis, seiches, éboulements, liquéfaction ...).
- Définition d'un programme d'étude.

3. INSTALLATIONS EXPERIMENTALES et PROGRAMMES

Réseau du LDG et des différents Observatoires. Stations de Cadarache et de Pierrelatte.

Un programme expérimental pourrait être défini au cours de l'étude des phénomènes induits par les séismes.

4. ETAT de l'ETUDE

Cette fiche, qui relaie la réalisation de la carte sismotectonique de la France, est en cours de démarrage.

5. RELATIONS avec d'AUTRES ETUDES

La fiche 121-1-03/4172-10 (Mise au point d'une méthodologie de calcul des spectres des séismes de référence des sites, à partir des paramètres physiques) est en aval du processus de détermination du risque dont elle constitue le stade suivant.

121-1-03/4172-10		3-1
TITRE METHODOLOGIE pour le CALCUL des SPECTRES des SEISMES de REFERENCE à PARTIR des PARAMETRES PHYSIQUES		Pays France Organisme Directeur CEA/DSN
TITLE (Anglais) METHODOLOGY for the CALCULATION of REFERENCE EARTHQUAKE SPECTRA BASED UPON PHYSICAL PARAMETERS		Organisme exécuteur CEA/DSN Responsable B. MOHAMMADIOUN
Date de démarrage 01.01.1976	Etat actuel en cours	Scientifiques H. FERRIEUX A. LEVRET G. MOHAMMADIOUN
Date d'achèvement 31.12.1979	Dernière mise à jour 20 /12.1978	

1. OBJECTIF GENERAL

Détermination des mouvements sismiques de référence adaptés aux sites des installations nucléaires. Ces mouvements de référence sont utilisés :

. soit directement dans l'analyse du dimensionnement des ouvrages et matériels,

. soit pour le contrôle d'un mouvement de référence standard.

2. OBJECTIFS PARTICULIERS

Elaboration des méthodes de calcul des spectres adaptés aux sites à partir de la magnitude, de la distance focale et de l'intensité du séisme de référence à l'aide d'une analyse statistique des données obtenues dans le monde et sélectionnées selon les différents critères suivants:

a- régional (Californie, Europe, etc..),

b- géologie similaire du site (roche dure, alluvions)

Etude probabiliste du risque sismique. Cette étude a pour but de définir une approche pour la détermination du mouvement de référence adapté au site associé à une valeur de probabilité annuelle. Une étude de la faisabilité d'une telle approche est en cours.

3. INSTALLATIONS EXPERIMENTALES et PROGRAMMES

Voir Fiches 121-1-05/4172-10 et 121-1-04/4172-10

.../...

4 - ETAT de l'ETUDE

Avancement à ce jour

Etablissement d'une relation liant la magnitude à la distance focale et à l'intensité au lieu d'enregistrement pour les séismes américains.

Calcul de corrélations liant les spectres à la magnitude et à la distance focale à partir des données américaines pour les intensités V, VI et VII. Les coefficients de corrélations ont été obtenus pour 46 fréquences et 5 amortissements.

Proposition, à la lumière de ces résultats, d'une méthodologie de calcul des spectres adaptés aux sites.

Résultats essentiels

Une nouvelle méthodologie a été conçue pour calculer des spectres adaptés aux sites qui accorde une importance prépondérante aux données généralement les mieux connues dans les cas des différents sites, à savoir l'intensité et la magnitude du SMHV. Cette méthodologie préconise le calcul d'un spectre SMHV au moyen des coefficients de corrélation liant le spectre à la magnitude, et à la distance focale obtenus à partir d'un ensemble d'enregistrements de même intensité, en l'occurrence celle du SMHV. Un couple magnitude/distance toujours compatible avec l'intensité caractéristique du site est défini selon une corrélation magnitude/intensité/distance focale réalisée sur un ensemble de données comprenant celles qui servent au calcul du spectre et homogènes entre elles. Par convention, le spectre SMS est défini comme étant deux fois celui du SMHV.

A l'heure actuelle, cette méthode permet de traiter valablement les cas des sites dont l'intensité du SMHV est inférieure à VIII et la distance focale égale ou supérieure à 10 km qui constituent la grande majorité des sites français. Toute nouvelle donnée acquise contribuera à améliorer la précision des coefficients de corrélation et à lever les limitations d'application.

Prochaines étapes

Amélioration des coefficients de corrélation par l'introduction de nouvelles données (pour les intensités V, VI et VII) et surtout à courtes distances ($R < 10$).

Etude des séismes alpins et établissement des corrélations analogues.

Possibilité, grâce à un nombre accru de données, de faire une sélection à l'intérieur d'un ensemble de spectres de même intensité selon des critères géologiques ou autres.

5 - RELATIONS avec d'AUTRES ETUDES

La fiche "Collecte de mesures sur les mouvements en zone épicentrale et d'informations sur les dégâts correspondants" apporte des informations essentielles à l'étude des corrélations entre les différents paramètres et au calcul des spectres de référence.

121-1-05/4172-10		3-1
TITRE COLLECTE de MESURES sur les MOUVEMENTS en ZONE EPICENTRALE et d'INFORMATIONS sur les DEGATS CORRESPONDANTS		Pays France Organisme Directeur CEA/DSN
TITLE (Anglais) COMPILATION of RECORDINGS of NEAR FIELD MOTION and of INFORMATION on the CORRESPONDING DAMAGES		Organisme exécuteur CEA/DSN Responsable B. MOHAMMADIOUN
Date de démarrage 01.01.1976	Etat actuel en cours	Scientifiques H. FERRIEUX A. LEVRET G. MOHAMMADIOUN
Date d'achèvement 31.12.1981	Dernière mise à jour 20.12.1978	

1. OBJECTIF GENERAL

Connaissance des caractéristiques du mouvement fort en zone épicentrale et de leurs effets en vue de la protection des installations nucléaires contre les séismes.

2. OBJECTIFS PARTICULIERS

Rassemblement des enregistrements des mouvements forts obtenus dans le monde et établissement d'une sismothèque.

Rassemblement des caractéristiques correspondant à ces enregistrements (intensité au lieu d'enregistrement, condition géologique du site, instrument enregistreur, etc..).

Enregistrements des répliques en zone épicentrale d'un séisme important à l'aide des dispositifs DSN.

Enregistrement de signaux sismiques d'origine artificielle dans le cadre d'études ponctuelles des sites.

Classement et analyse de ces données en fonction des différents critères:

. régional,

. intensité au lieu d'enregistrement,

. condition du site (roche dure, alluvions).

.../...

3. INSTALLATIONS EXPERIMENTALES et PROGRAMMES

Stations sismologiques légères destinées à l'enregistrement des séismes (répliques) dans la zone épicentrale après un séisme important.

Stations simologiques temporaires dans des zones à forte sismicité.

Réalisation des expérimentations sur un site à partir de sources sismiques artificielles afin d'étudier la loi de transmission des ondes sismiques dans la région immédiate du site.

4. ETAT de l'ETUDE

Avancement à ce jour

Enregistrements des nombreuses répliques dans la région de Frioul avec la détermination de leur intensité: création d'un fichier magnétique format CALTECH destiné à comporter tous les spectres de réponse obtenus; y figurent déjà les séismes importants des 11 et 15 septembre 1976.

Enregistrements des séismes dans la vallée de l'Ubaye (calcul des spectres en cours).

Enregistrements en plusieurs points des nombreuses répliques dans la région de Tubingen après le séisme du 3 septembre 1978, dans cette région; (dépouillement de ces enregistrements et calcul des spectres correspondants, en cours).

Acquisition des enregistrements du séisme de Frioul réalisés par les réseaux italien et yougoslave.

Acquisition, dans le cadre de l'établissement d'un fichier semblable, pour les séismes enregistrés en Europe et au Moyen Orient (zone alpine) des enregistrements de la séquence d'Ancona, vus et corrigés par le Pr Ambraseys (Imperial College de Londres). Une action est engagée auprès de celui-ci afin de préciser et d'homogénéiser les caractéristiques de ces enregistrements.

Acquisition, dans le cadre de l'accord CEA/NRC, d'une bande réalisée à la demande du NRC qui rassemble les caractéristiques des enregistrements du mouvement fort obtenus dans le monde.

Obtention de documents détaillés traitant des caractéristiques des sites d'enregistrements des mouvements forts réalisés aux Etats-Unis.

Mise au point à la CISI d'un programme d'accès aux fichiers magnétiques de format standard CALTECH permettant d'extraire, de façon commode, accélérogrammes ou spectres de réponse

Résultats essentiels

La sismothèque comporte actuellement sous forme standard la plupart des mouvements forts obtenus aux USA et les enregistrements des séismes importants du Frioul réalisés par le DSN/BERSSIN. L'accès facile et sélectif à ces données est assuré. Les caractéristiques relatives à ces séismes ont été précisées. Des démarches ont été entreprises pour compléter la sismothèque aussi bien sur le plan européen que mondial.

Prochaines étapes

Etablissement d'un fichier semblable pour les séismes européens à mouvement fort.

Etude des caractéristiques des séismes enregistrés en Europe (magnitude, intensité, distance focale).

Apport à la sismothèque de données étrangères (complément d'enregistrements américains, données japonaises, etc..).

Poursuite des enregistrements des séismes (vallée de l'Ubaye, Pyrénées) en France et dans les pays limitrophes.

.../...

5. **RELATIONS avec d'AUTRES ETUDES**

Les résultats de cette étude sont indispensables à l'établissement des méthodologies de calcul des mouvements de référence adaptés aux sites (Fiche 121-1-03/4172-10).

6. DOCUMENTS de REFERENCE

Etude des répliques du séisme du 6 mai 1976 au Frioul
A.BARBREAU, B.MOHAMMADIOUN, H.FERRIEUX, G.MOHAMMADIOUN
OCDE-Rome, 11/13 octobre 1977

121-1-06 / 4172-10 et 4113-01		3-1
TITRE MECANIQUE DU SOL AU VOISINAGE DES INSTALLATIONS ET INTERACTION SOL-FONDATION DANS LA SURETE SISMIQUE		**Pays** FRANCE **Organisme Directeur** CEA/DSN
TITLE (Anglais) SOIL MECHANICS (VICINITY OF INSTALLATIONS) AND SOIL-STRUCTURE INTERACTION IN SISMIC SAFETY		**Organisme exécuteur** CEA/DSN **Responsable** M.MOHAMMADIOÚN (DSN) M.LIVOLANT (DEMT)
Date de démarrage 01.01?78	**Etat actuel** en cours	**Scientifiques** H. FERRIEUX E. FAVIER
Date d'achèvement 31.12.82	**Dernière mise à jour** 31.12.1978	

1. OBJECTIF GENERAL

L'objectif général de cette étude est la recherche des données expérimentales pour la validation des méthodes de calcul qui permettent d'établir une relation entre le signal sismique en champ libre et les mouvements de niveau de fondation, et d'une façon générale, au voisinage des structures .

2. OBJECTIFS PARTICULIERS

2.1. Acquisition de données expérimentales sur l"interaction, in situ

Comme il n'existe que quelques exemples de séismes enregistrés simultanément en champ libre et dans les constructions, l'idéal serait d'enregistrer dans le futur, des répliques après un séisme important en champ libre et au niveau de la fondation (radier) .

2.2. Mesures des paramètres élastiques du sol (qui interviennent dans le calcul de l'interaction sol-fondation par une méthode de mesure de vitesse de propagation des ondes sismiques) .

Une solution de rechange et complémentaire par rapport à l'approche précédente consiste à faire appel à des vibrations artificielles, engendrées par un dispositif mécanique ou par explosion, comme source d'excitation d'une maquette de fondation à l'échelle réduite .

2.3. Validation du modèle de sol à rouleaux

Si le calcul permet d'étudier le domaine de comportement linéaire du sol, le domaine non linéaire et la propagation d'ondes nécessitent des études expérimentales .

Un modèle de sol bi-dimensionnel à rouleaux permettant la validation des méthodes de calcul pour les comportements non linéaires doit être utilisé pour faire une étude sous sollicitation dynamique . Un certain nombre d'essais sont nécessaires pour déterminer le comportement du modèle sous divers types de sollicitations (mesures d'accélération de déplacement et de contrainte) . Les résultats de ces essais doivent être comparés aux modèle de calcul .

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

3.1. En collaboration avec EdF et DEMT, une étude sur des maquettes de l'interaction sol-fondation est en cours de réalisation sur l'aéroport de Nice . Des excitations sismiques seront les vibrations engendrées par un dispositif de compactage (chute d'une masse de 150 T d'une hauteur de 23m) Le DSN effectuera:

- des mesures du mouvement en champ libre en fonction de la distance et de la hauteur de la chute,
- des mesures expérimentales des modules élastiques intervenant dans l'étude de l'interaction sol-fondation par une méthode spécifique dite "cross-holes" .

3.2. Table vibrante VESUVE du DEMT (CEN-Saclay): essais de comportement du modèle à rouleaux .

4. ETAT DE L'ETUDE

Avancement à ce jour

Obj. 2-1 - La première campagne de mesure a été effectuée par le DSN/SESRS/BERSSIN sur l'aéroport de Nice en Juin 1978, pour déterminer le mouvement sismique en champ libre, induit par la machine compacteuse MEYNARD .

Résultats essentiels

Obj. 2-1 - Cette étude a montré que les mouvements correspondants sont assez proches des enregistrements de certains séismes et peuvent donc être utilisés pour simuler un ébranlement sismique .

5. PROCHAINES ETAPES

Obj. 2-2 et 2-3 -

6. RELATION AVEC D'AUTRES ETUDES

Les résultats de cette étude sont indispensables à l'analyse parasismique des installations nucléaires (Fiche 142-1-02 / 4113-01) .

142-1 - 01/4113-01		3.1
TITRE ANALYSE PARASISMIQUE D'UNE CENTRALE NUCLEAIRE A EAU ORDINAIRE Méthode de Calcul		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) Seismic analysis of a nuclear power plant. PWR reactor building.		Organisme exécuteur CEA/DEMT Responsable
Date de démarrage 1.1.75	Etat actuel en cours	Scientifiques
Date d'achèvement 31.12.79	Dernière mise à jour 1.2.79	

1. OBJECTIF GENERAL

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 modifications de réalisation.

2. OBJECTIFS PARTICULIERS.

1) Modélisation des structures.
2) Analyse modale des constituants.
3) Méthodologie de prise en compte des couplages.
4) Evaluation des spectres de planchers.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME.

4. ETAT DE L'ETUDE

1) Avancement à ce jour :

ETUDE D'UNE CENTRALE PWR 900. L'étude de comportement et de la tenue au séisme d'une enceinte de confinement du type contrat programme a été effectuée. Une modélisation optimisée des structures internes (génie civil) est recherchée.

2) Résulats 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)

Parallèlement une méthode de calcul des spectres de plancher par la transformée de Fourier a été dévelopé. Une analyse mathématique de la réponse d'équipements à points de fixation multiples a été faite.

5. PROCHAINES ETAPES

6. RELATIONS AVEC D'AUTRES ETUDES

Calculs interconnectés avec l'étude de la tenue aux séismes de portiques (essentiellement financée par EdF) .

7. DOCUMENTS DE REFERENCE

Note EMT 77.212

Note EMT 78/199

8/ DEGRE DE DISPONIBILITE

152-1-02 / 142-1-02 / 4113-01		3-1
TITRE ANALYSE PARASISMIQUE D'UNE CENTRALE NUCLEAIRE, INTERACTION SOL-FONDATION		Pays FRANCE Organisme Directeur CEA-DgCS , EdF
TITLE (Anglais) SEISMIC ANALYSIS OF A NUCLEAR PLANT. SOIL-STRUCTURE INTERACTION		Organisme exécuteur CEA/DEMT Responsable
Date de démarrage 1978	Etat actuel en cours	Scientifiques
Date d'achèvement 1983	Dernière mise à jour 15.12.1978	

1. OBJECTIF GENERAL

Le but de cette étude est la mise au point d'une méthode pour établir la relation entre le signal sismique en champ libre et les mouvements imprimés aux fondations .

Deux méthodes de calcul pour la prise en compte de l'interaction sol-structure sont actuellement utilisées . La première méthode utilise les ressorts et les amortisseurs de sol équivalents, mais un certain nombre de configurations (fondation en profondeur, sol en couches de propriétés mécaniques différentes ...) ne peuvent être prises en compte . La seconde méthode utilise le calcul par éléments finis à deux dimensions mais l'effet de la troisième dimension comme celui des conditions aux limites sont très mal pris en compte . Devant ces considérations, il est nécessaire de développer et améliorer les méthodes de calcul pour pouvoir traiter correctement les points laissés de côté par les méthodes actuelles .

2. OBJECTIFS PARTICULIERS

2.1 .Etude analytique de l' impédance d'un radier circulaire sur un demi espace infini .

2.2 .Extension au cas de radiers de forme quelconque .

2.3 .Développement des modèles par éléments finis avec frontières absorbantes

2.4 .Qualification des codes de calcul utilisés dans le cas de bâtiments sur radier par des essais in situ .

2.5 .Codes de calcul pour taiter des fondations sur pieux à partir d'essais in situ .

2.6 .Etude théorique accompagnée d'essais sur table vibrante, pour établir un modèle cohérent de calcul .

2.7 .Traitement de bâtiments voisins sur fondations séparées .

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

3.1 Essais in situ

.Pour une complète validation expérimentale des modèles de calcul, des expériences sur sol réel sont nécessaires . On peut ainsi étudier la réponse de maquettes de construction (bâtiment réacteur par exemple) liées au sol par divers types de fondation (radier unique, radiers multiples, pieux ancrés, pieux flottants ...) à une excitation sismique artificielle . Un procédé d'excitation existant actuellement sur le chantier e l'aéroport de Nice consiste à faire tomber sur le sol, à une certaine distance de la maquette à tester, une masse de 150 tonnes d'une hauteur variable . Par ailleurs, les impédances équivalentes peuvent être déterminées par excitation harmonique de la maquette avec balayage defréquence .

3.2 .Table vibrante du DEMT

(CEN.Saclay) Programme non parvenu

4. ETAT DE L'ETUDE

. Avancement à ce jour

Objectifs 2-1, 2-2 réalisés

Objectif 2-7 réalisé partiellement: la méthode de calcul de ressorts de sol élaborée prend en compte l'effet de couplage de plusieurs radiers .

. Principaux résultats
Voir ci-dessus

5. PROCHAINES ETAPES

(programme 1979)

. Réalisation et exploitation des expériences sur l'aéroport de Nice (Obj. 2-4 et 2-5)

. Mise au point du modèle cohérent de calcul (Obj. 2-3 et 2-6)

6. RELATIONS AVEC D'AUTRES ETUDES

- ensemble des études définissant le signal sismique incident (rubrique 121-1) et notamment la mécanique du sol et les mouvements au voisinage de l'interface sol-fondations voir fiche 121-1-06 .

- ensemble des autres études de génie parasismique (rubrique 142-1)

7. DOCUMENTS DE REFERENCE

J. GAUVAIN - note technique EMT 78/23

"Interaction sol-fondation - Relation entre les mouvements de plusieurs zones à la surface d'un milieu élastique semi-infini" .

8. DEGRE DE DISPONIBILITE

Disponible

142-1 - 03 / 152-1 - 03/4113-01		3-1
TITRE Tenue de structures - types sous excitation sismique. Essais en laboratoire.		**Pays** FRANCE
		Organisme Directeur CEA/DgCS
TITLE (Anglais) Behaviour of typical structures under seismic excitation. Shake table tests.		**Organisme exécuteur** CEA/DEMT - Saclay
		Responsable
Date de démarrage 1/75	**Etat actuel** en cours	**Scientifiques**
Date d'achèvement 1980	**Dernière mise à jour** 12/78	

1. OBJECTIF GENERAL

L'objet principal de cette étude est d'approfondir la connaissance des limites de résistance et du processus de ruine en régime dynamique des éléments de structures utilisées dans la construction des centrales nucléaires. Subsidiairement, les essais sur des éléments représentatifs de construction traditionnelle devront permettre de fonder une meilleure corrélation entre les intensités macrosismiques et les paramètres mécaniques utilisés en ingénierie.

2. OBJECTIFS PARTICULIERS

1) Essais de structures traditionnelles.
2) Essais d'éléments en béton armé ordinaire - Flexion : poteaux.
3) Essais d'éléments en béton armé ordinaire - Cisaillement : voiles.
4) Essais d'éléments en béton armé composites (en L, etc.)
5) Essais d'éléments de béton précontraint.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Table vibrante VESUVE (DEMT).

Laboratoires du CEB TP

4. ETAT DE L'ETUDE

1) Avancement à ce jour :

1 - Campagne d'essais exécutée sur murs non contreventés, chargés.

a) Briques et mortier classés "anciens".
b) Briques et mortier classés "modernes".

2 - Campagne d'essais sur poteaux de béton armé

3 - Calculs préliminaires relatifs aux voiles de béton armé

2) Résultats essentiels :

1 - Murs.

Connaissance des fréquences propres et de leur variation au cours de la dégradation du mur. Amortissements en essais de lâcher et variation des amortissements.

2 - Poteaux en béton armé.

Comportement dynamique des poteaux en flexion. Connaissance de la longueur effective (encastrements) et comparaison avec l'approche par sections équivalentes du béton fissuré. Etablissement d'une loi de comportement du béton armé en flexion jusqu'à la ruine, compte tenu de la connaissance des raideurs et des amortissements en fonction de la déflexion.

5. PROCHAINES ETAPES

a) Voiles en béton armé travaillant dans leur plan principal.
Comportement dynamique en cisaillement.
Calculs préliminaires : estimation des efforts nécessaires pour des effets significatifs. Etude de l'effet du rapport hauteur-longueur.
Essais statiques.
Essais dynamiques.

b) Poteaux en béton précontraint. Programme avec contrat DGRST

6. RELATIONS AVEC D'AUTRES

7. DOCUMENTS DE REFERENCE

- Comportement sismique d'éléments de bâtiments (étude de murs anciens et modernes). Note technique EMT 76/228.
- Comportement statique. Essais dynamiques des poteaux en flexion. Rapport d'études CEBTP mai 1977. Dossier 952-6-033.
- Communication à la conférence ENS/ANS. Bruxelles Octobre 1978
- Note technique EMT 78/177.

8. DEGRE DE DISPONIBILITE

	CLASSIFICATION: 3.1
TITLE (ORIGINAL LANGUAGE): ANALISI DI RISCHIO SISMICO PER POSSIBILI SITI NUCLEARI	COUNTRY: Italy
	SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): SEISMIC HAZARD ANALYSIS FOR CANDIDATE NUCLEAR SITES	ORGANISATION: CNEN
	PROJECT LEADER: M. Basili, N. Pacilio

INITIATED:	COMPLETED:	SCIENTISTS:
1976		A. Basili, M. Basili, V. Cagnetti, V. Gorelli, O. Iacurto, N. Pacilio, G. Tinelli.
STATUS: in progress	LAST UPDATING: July 1979	

1 GENERAL AIM: Assessment of the maximum expected acceleration level on a given nuclear site via a probabilistic approach for the definition of the contributing factors of seismic risk mapping.

2 PARTICULAR OBJECTIVES: (1) Stochastic analysis of seismic occurrences
(2) Identification of theoretical models for interpreting the distribution of seismic occurrences
(3) Effects of a probabilistic approach on the determination of maximum expected acceleration.

3 PROJECT STATUS: Progress to date
- Theoretical models for the time distribution of historical data
- Stochastic analysis of the 1976 Friuli earthquake shocks
- Statistical analysis of historic catalogues of seismic data
- Stochastic model of the Gutenberg-Richter relationship.

Essential results
- Analytical solutions in terms of the expected variance-to-mean ratio of the number of shocks
- Labeling of a seismic zone via stochastic and geophysical indicators
- Development of computer codes SEISMOS, QUAKOS, and REGLIN
- Statistical analysis of a historic catalogue of seismic occurrences in the Central part of Italy

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
SESMIC HAZARD ANALYSIS FOR CANDIDATE NUCLEAR SITES	3.1

4 NEXT STEPS:
- -Statistical analysis of a historic catalogue of seismic occurrences in the Northeastern part of Italy.
- -Seismic risk analysis of the Northeastern part of Italy
- -Statistical analysis of seismic occurrences via non-linear least-square method computer codes.

5 RELATION TO OTHER PROJECTS : 3.1 (other CNEN projects)

6 REFERENCE DOCUMENTS:

A. BASILI, M. BASILI, A. COLOMBINO, O. IACURTO, N. PACILIO, G. SENA - Analisi dei dati di Karnik: indicatori sismici stocastici come funzioni della magnitudo - CNEN RT/AMB(78)1
A. BASILI, M. BASILI, A. COLOMBINO, O. IACURTO, N. PACILIO, G. SENA - SEISMOS: una routine per il trattamento di dati sismici storici su IBM-370 - CNEN RT/AMB(78)2
A. BASILI, M. BASILI, A. COLOMBINO, O. IACURTO, N. PACILIO, G. SENA - QUAKOS: una routine per il trattamento di sequenze sismiche su IBM-370 - CNEN RT/FIMA(78)3
A. BASILI, M. BASILI, A. COLOMBINO, O. IACURTO, N. PACILIO, G. PRATICO', L. ROMANELLI, G. SENA - REGLIN: una routine per analisi di regressione lineare su IBM-370 - CNEN RT/AMB(79)2
A. BASILI, M. BASILI, A. COLOMBINO, O. IACURTO, N. PACILIO, G. SENA - Il ruolo della analisi stocastica nei problemi di rischio sismico - CNEN RT/AMB(79)
A. BASILI, M. BASILI, A. COLOMBINO, O. IACURTO, N. PACILIO - SEISMIC RISK: (1) analysis of the number of earthquakes in magnitude domain - CNEN RT/AMB(79)
A. BASILI, M. BASILI, A. COLOMBINO, O. IACURTO, N. PACILIO - SEISMIC RISK: (2) fitting the Gutenberg-Richter relation to earthquake data - CNEN RT/AMB(79)

7 DEGREE OF AVAILABILITY: free.
Contact persons:
M. Basili, N. Pacilio
CNEN, CSN Casaccia, CP 2400, I-00100 ROMA

	CLASSIFICATION: 3.1 - 3.5
TITLE (ORIGINAL LANGUAGE): Studi di ingegneria del sito	COUNTRY: ITALY
	SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Studies of site engineering	ORGANISATION: CNEN
	PROJECT LEADER: Laboratorio Ingegneria Sito

INITIATED:	COMPLETED:	SCIENTISTS:
Nov. 1974 (present phase)		M. Basili, V. Cagnetti, A. Fontanive
STATUS: In progress	LAST UPDATING: July 1979	

1) General Aim
Studies on parameters occurring in the evaluation of sites for nuclear plants.

2) Particular Objectives
The program is organized into the following tasks:
- analysis of earthquake strong-motion and seismometric records
- analysis of soil dynamic characteristics
- analysis of soil-structure interactions
- statistical analysis and studies of exceptional meteorological events
- stochastic analysis of seismic data.

3) Project Status
1) Procedures for macroseismic and microseismic records processing have been developed.
2) Studies on seismicity of Friuli (Northern Italy) have been carried out (focal mechanism, earthquake spectra, seismic moment, linear dimension of fault, dislocation and stress drop, etc.).
3) Codes and experimental techniques have been developed.

4) Next Steps
Besides development of above items: research on correlations between seismic parameters and sands liquefaction.

5) Relation to other Projects
3.1 CNEN, ENEL programs.

6) Reference Documents
1. Reports on the 1976 Friuli earthquake published by the CNEN-ENEL Commission on Seismic Problems Associated with the Installation of Nuclear Plants.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Studies of site engineering	3.1 - 3.5

a) "Strong-motion Earthquake Accelerograms-Digitized and Plotted Data - Uncorrected Accelerograms-Part 1, 2, 3, 4," Roma, 1976 - 1978

b) "Contribution to the Study of Friuli Earthquake of May 1976," Roma, November 1976.

2. Various reports on the 1976 Friuli earthquake, presented at the OECD Specialist Meeting on the 1976 Friuli Earthquake and the Antiseismic Design of Nuclear Installations, Roma, October 1977.
3. Other reports on specific matters are available.

7) Degree of Availability

Open.

For any information: Laboratorio Ingegneria Sito, CNEN, CSN Casaccia, C.P. 2400, I-00100 Roma.

8) 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.

	CLASSIFICATION: 3.1
TITLE (ORIGINAL LANGUAGE): Sviluppo di strumentazione e misure sismiche per la valutazione dei siti	COUNTRY: Italy SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Seismic instrumentation development and seismic measurements for site evaluation	ORGANISATION: CNEN PROJECT LEADER: R. Cervellati

INITIATED:	COMPLETED:	SCIENTISTS:
May 1974 (present phase)		
STATUS: In progress	LAST UPDATING: June 1979	

General aim:

Seismic instruments development and seismic measurements, in order to study sites from the seismological point of view.

Particular objectives:

Operation of accelerometers; setting up and operation of seismometric equipments; data logging; analysis of the response of seismometric instrumentation.

Experimental facilities:

A live network of accelerometers. Operating seismometers. An electronic shop for maintenance and calibration.

Project status:

1. In the frame of a collaboration CNEN-ENEL a network of accelerometers has been set up all over Italy with the aim of recording the accelerations during strong earthquakes. The records will be employed in the characterization of the design earthquake.
   In five accelerometers a broadcast time code is introduced in the accelerometer records.
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.).
3. A seismic network has been set up. It is presently in operation in Irpinia (Southern Italy). It consists of 6 radiolinked seismometric stations.

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TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Seismic instrumentation development and seismic measurements for site evaluation	3.1

Next steps:

- Cable telemetry.
- Direct transfer of seismic data into a scientific computer.
- Assembling of a second seismometric network.
- Documentation on present instrumental techniques related to physical forerunners of earthquakes.

Relation to other projects:

CNEN, ENEL programs (3.1).

Reference documents:

- Reports on the 1976 Friuli earthquake.
- Studies on performance of accelerometers of the network; studies on performance of seismometers.

Degree of availability: Open

Contact person: R. Cervellati, CNEN, CSN Casaccia, CP 2400, I-00100 Roma.

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.

	CLASSIFICATION: 3.1
TITLE (ORIGINAL LANGUAGE): Ricerche strutturali e sismotettoniche ai fini della sicurezza degli impianti nucleari	COUNTRY: ITALY
	SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Structural and seismotectonic research for the safety of nuclear plants	ORGANISATION: CNEN
	PROJECT LEADER: G. MAGRI
INITIATED: Jan. 1975 / COMPLETED:	SCIENTISTS:
STATUS: in progress / LAST UPDATING: July 1979	

1) General Aim
Development of methods and research criteria in an interdisciplinary framework for nuclear plant sites evaluation.

2) Particular Objectives
a) Elaboration of criteria to define the seismotectonic behaviour of areas with and without historical seismicity.
b) Elaboration of criteria to identify seismotectonical provinces.
c) Elaboration of criteria to define the maximum expectable earthquake associated to a tectonic structure.

3) Experimental Facilities and Programme
Seismometers with recorders and geochemical facilities.

4) Project Status
Progress to Date:
1) Paleogeographical data as well as Quaternary geological data have been collected.
2) Studies on Friuli seismotectonic features have been performed.

5) Next Steps
Collection and study of records of seismical events in the Southern Apennine Region.

6) Relation to other Projects: 3.1 (other CNEN, ENEL programs).

7) Degree of Availability: Free; CNEN Lab. Geologia Ambientale, C.P. 2400, I-00100 Roma, Italy.

	CLASSIFICATION: 3.1
TITLE (ORIGINAL LANGUAGE): Rete di rilevamento sismico	COUNTRY: ITALY
	SPONSOR: ENEL
TITLE (ENGLISH LANGUAGE): Seismic monitoring network	ORGANISATION: ENEL
	PROJECT LEADER: F. CAPOZZA
INITIATED: 1973 / COMPLETED:	SCIENTISTS: A. BERENZI
STATUS: In progress / LAST UPDATING: June 1979	

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 consists of 168 monitoring points distributed in the whole Italian territory with the exception of Sardinia.
Each monitoring point is 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 "CNEN-ENEL Commission on Problems Associated with the Installation of Nuclear Plants".

5. Next steps

Further recording will be necessary to obtain the definition of reference earthquake.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Seismic monitoring network	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 published by the "CNEN-ENEL Commission on Seismic Problems Associated with the Installation of Nuclear Plants":

a - "Contribution to the Study of Friuli Earthquake of 1976", Rome, Nov.1976
b - "Strong Motion Earthquake Accelerograms - Digitized and Plotted Data - Uncorrected Accelerograms", Part 1, Rome, July 1976
c - Id., Part 2, Rome, January 1977
d - Id., Part 3, Rome, Nov.1977
e - Id., Part 4, Rome, July 1978

	CLASSIFICATION: 3.1
TITLE (ORIGINAL LANGUAGE): Studio sulla possibilità di previsione di terremoti con metodi idrogeochimici	COUNTRY: ITALY
	SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Study on the possibility of predicting earthquakes by hydrogeochemical methods	ORGANISATION: CNEN
	PROJECT LEADER: M. Dall'Aglio
INITIATED: January 1975 / COMPLETED:	SCIENTISTS: M. Brondi, E. Ghiara, C. Mignuzzi
STATUS: in progress / LAST UPDATING: July 1979	

1. General Aim

Earthquake prediction.

2. Particular Objectives

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.

3. Experimental Facilities and Programme

Some hydrothermal Italian systems are regularly checked in order to study the variation of water composition in relation to seismic activity.
Major constituents dissolved in spring waters, and some elements successfully used as premonitory indicators of earthquakes, are analyzed; particular emphasis is given to the He and Rn measurements in liquid and gaseous phases.
Classical analytical instruments are used in the laboratory. Electric conductance, pH, temperature, HCO_3 content measurements are carried out on the field by miniaturized analytical kits specially designed.

4. Project Status

Basic measurements on the field.

5. Next steps

Analysis of water and gas samples will be carried out by a mass spectrometer.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Study on the possibility of predicting earthquakes by hydrogeochemical methods	3.1

6. Reference Documents

6.1. Dall'Aglio M. "Geochimica e gestione dell'ambiente". Notiziario CNEN, Anno 20, n. 7, July 1974.

6.2. Dall'Aglio M. "Geochemistry of Stream and Ground Waters from Western Sicily. The Changes in Spring Water Chemistry after the 1968 Earthquake". Atti Conv. Intern. Acque Sotterranee, Palermo, December 1974.

6.3. Dall'Aglio M. "Earthquake Prediction by Hydrogeochemical Methods". IAGL Symposium on the Geochemistry of Natural Waters, Burlington, 1975. Soc. Italiana di Mineralogia e Petrologia, Rendiconti, vol. XXXII (1), pag. 421-436, 1976.

7. Degree of Availability

Free available by M. Dall'Aglio, CNEN, CSN Casaccia, C.P. 2400, I-00100 Roma.

	CLASSIFICATION: 3.1
TITLE (ORIGINAL LANGUAGE): ASSESSMENT OF SEISMIC HAZARD WITHIN THE UNITED KINGDOM	COUNTRY: UK
	SPONSOR: UKAEA
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/UKAEA
	PROJECT LEADER: MAGH ALDERSON
INITIATED: JANUARY 1978 / COMPLETED:	SCIENTISTS: DR P WINTER
STATUS: IN PROGRESS / LAST UPDATING: MAY 1979	

1. GENERAL AIM: To review the level of hazard to nuclear power plants due to the possibility of seismic disturbance.

2. PARTICULAR OBJECTIVES: To review the seismicity of the United Kingdom and to combine the probability of earthquake occurrence with the probability of structural damage given an earthquake occurring.

3. EXPERIMENTAL FACILITIES AND PROGRAMME:

4. PROJECT STATUS:

   1. Progress to Date:- Report in preparation

   2. Essential results:-

5. NEXT STEPS: To extend method to multiple plant sites.

6. RELATION TO OTHER PROJECTS AND CODES:-

7. REFERENCE DOCUMENTS:-

8. DEGREE OF AVAILABILITY:-

9. ADDITIONAL INFORMATION:-

	CLASSIFICATION: 3.1
TITLE (ORIGINAL LANGUAGE): SEISMIC ANALYSIS (GSDR.1) WITH APPLICATION TO PWRS	COUNTRY: UK
	SPONSOR: UKAEA
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/UKAEA
	PROJECT LEADER: RNH McMILLAN
INITIATED: DECEMBER 1978 / COMPLETED:	SCIENTISTS/ENGINEER: MAHG ALDERSON
STATUS: / LAST UPDATING: MAY 1979	

BACKGROUND

Safety Assessment Principles require that two levels of free field ground motions designated the Safe Shutdown Earthquake (SSE) and the Operating Basis Earthquake (OBE) shall be determined for the reactor site. It is a further requirement that the design should be such as to ensure that in the event of an SSE the reactors can be shut down safely and all safety-related structures and plant can be maintained in a safe condition. Additionally the reactors, fuel storage and active waste storage facilities shall be demonstrated to be safe in the event of repeated ground motions equivalent to the OBE level.

Thus there are two separate aspects of the safety justification which need to be addressed. First the definition of suitable time motion histories requires considerable subjective judgement based on geological/tectonic evaluations coupled with a treatment of rare-event probability. Secondly the evaluation of structural response of the plant to the postulated ground motions. The first aspect (free field ground motions) is not specific to PWRs, nevertheless it is important that the analysis of ground motions is relevant to typical PWR buildings and construction methods.

The following objectives cover both ground motion studies and structural response.

OBJECTIVES

1. To further develop probabilistic/statistical methodology for the determination of free field ground motions under SSE and OBE conditions.
2. To complete development of the SARSPEC code for the generation of Response Spectra from known or postulated earthquake time-motion histories.
3. To review available methods for soil/structure interaction analysis relevant to typical PWR structures and to recommend suitable methods eg the FLUSH code.
4. To maintain SAP IV for the evaluation of PWR structural response of systems and components to SSE/OBE input motions. Carry out typical calculations.
5. To extend SAP IV for the inclusion of fluid elements in order to evaluate free liquid surface responses in PWR fuel storage ponds. Carry out typical calculations.
6. To review the requirements or otherwise for installing instrumentation for the detection of design earthquake level exceedance for PWRs. Administrative action following exceedance will also be reviewed.

FACILITIES

(a) Analytical Codes   SAPSPEC   SAP IV   NONSAP   FLUSH

(b) Experimental   No UK facilities identified.

OVERSEAS COLLABORATION

France: CEA/EDF/AEA/NPC/HSE information exchange

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	3.2	RS 149

Vorhaben/Project Title	
Untersuchungen der Widerstandsfähigkeit von Betonstrukturen gegen Flugzeugabsturz Investigation of the Resistance of Concrete Structures to Crashing Aircrafts	Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Bundesamt für Wehrtechnik und Beschaffung KG IV 7

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.10.74	30.6.1981	Weymar
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	December 1978	

The report covering RS 149 is contained in the RS 165 report.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	3.2	RS 165
Vorhaben/Project Title Grenztragfähigkeit von Stahlbetonplatten bei hohen Belastungsgeschwindigkeiten (z.B. Flugzeugabsturz) Ultimate Bearing Capacity of Reinforced Concrete Slabs Under Time-Dependent Loads (e.g. Aircraft Crash)		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Hochtief
Arbeitsbeginn/Initiated 1.4.75	Arbeitsende/Completed 30.9.1981	Leiter des Vorhabens/Project Leader Riech (coordination)
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

The combined research programs RS 165 and 149 are dedicated to the theoretical and experimental investigation of the two essential problems encountered with loading case "aircraft crash":

1.1 Determination of impact load/time functions under the impact of strongly deformable missiles;

1.2 Determination of the kinetic ultimate bearing capacity of reinforced concrete slabs.

2. Particular Objectives

The preceding and the attendant theoretical studies within the scope of RS 165 aim at contributing to the understanding of the following items:

- the impact of deformable missiles
- the physical nonlinear material behaviour of reinforced concrete structures subjected to time-dependent loading
- the influence of finite deformation (geometrically nonlinear theory)
- the stress distribution via the thickness of the slab at the area of loading by means of the three-dimensional continuous theory.

Research program RS 149 comprises procurement, setup and test operation of the acceleration plant, the construction of the abutment for the targets, the manufacture of approx. 24 missiles and approx. 24 reinforced concrete test slabs as well as the handling of the measuring equipment.

3. Research Program

3.1 Impact load/time characteristics during the impact of deformable missiles onto quasi-rigid reinforced concrete structures (see 1.1)

3.1.1 Variation of the impact velocity

3.1.2 Variation of the distribution of the rigidity in longitudinal direction of the missile

3.2 Kinetic ultimate bearing capacity of reinforced concrete slabs subjected to the impact of deformable missiles (see 1.2)

3.2.1 Investigation of the influence of several design parameters on the local bearing capacity (punching strength):
Variation of the concrete strength, shear reinforcement, thickness of the slab, bending reinforcement

3.2.2 Influence of the load/time function:
Variation of the impact period, time of load increase, load amplitude

3.2.3 A limited number of tests is scheduled to check the calculation models for the global behaviour of structural members (ultimate bending bearing capacity) of reinforced concrete slabs.

4. Experimental Facilities

4.1 In order to improve the experimental methods for the determination of the impact load/time characteristics of the tests of item 3.1 it is checked whether the application of a force measuring system specially designed to this purpose leads to reliable results.

Two expert firms submitted and offered two different design drafts for this force measuring system.
The qualification tests for both drafts have not yet been finally concluded.

4.2 During the numerical preparation of the measurement data of the first four tests of measuring series II (see 3.2) relevant divergences appeared between the displacement/time characteristics determined by way of integration from the acceleration measured at the rear front of the slab and the deflection of the slab directly measured by means of inductive displacement transducers. As the displacement values gained from the acceleration values by way of integration reached physically absurd sizes and as the mistake causing this error cannot be eliminated, this measuring procedure will no more be applied in future tests. In order to control the measuring results delivered by the inductive displacement transducers specially constructed mechanical displacement transducers shall be additionally applied.

5. Progress to Date

5.1 Tests of measuring series II (see 3.2.1)

Based on the findings of the first four tests in autumn 1977 (see Annual Report 1977 for RS 165), five additional firing tests were performed with reinforced concrete plates and deformable missiles.

The purpose of this series was to obtain information concerning the effect of the shear reinforcement and the thickness of the slab on the punching strength of the reinforced concrete slabs. The following variants of the test slabs were set up:

- Test II/5: The difference between this test slab and slab II/4 tested on Dec. 20th, 1977 lies in the fact that slab II/5 does not contain any shear reinforcement;
  missile type 11; recorded impact velocity 235 m/s (requested velocity = 223 m/s); state of destruction of the slab: perforation;
- Test II/6: The test slab fully corresponds to that used for II/4;
  missile type 11; impact velocity = 257,6 m/s; state of destruction of the slab: wide-spread scabbing of the concrete cover at the back of the slab, reinforcement not destructed, no perforation, penetration depth at the front of the slab approx. 15 cm;
- Test II/7: Test slab corresponds to that of II/4, the shear reinforcement, however, being reduced to 50 % (20 Ø 14 per m^2);
  missile type 11; impact velocity = 225,3 m/s; state of de-

struction of the slab: penetration at the front of the slab approx. 8 cm, reinforcement not destructed, minor spalling of the concrete cover at the back of the slab, maximum permanent deflection approx. 9 cm;

- Test II/10: The reinforcement of the slab corresponds to that of II/5, the thickness of the slab, however, amounting to 90 cm (II/5: 70 cm);
  missile type 11; impact velocity = 245,6 m/s; state of destruction of the slab: penetration depth at the front of the slab approx. 4 cm (concrete spallings extending to the front reinforcement layer), back of the slab: hardly no cracks (width of cracks 0,5 mm).

5.2 The documentation and evaluation of previous tests at Meppen was continued. Presently the emphasis of evaluation activities lies with the tests of measuring series I (see 3.1), only part of these measuring data, however, being useable.

5.3 In the course of the attendant theoretical investigations additional emphasis was attributed to the revision of the theory concerning the impact procedure of deformable missiles under consideration of the coupling of missile and local behaviour of the structural member. To this purpose a phenomenological description was set up and the scope of applied parameters was restricted by means of empirical formulas.

6. Results

6.1 A comparison of the presently available test results of the tests performed at Meppen with regard to item 3.2.1 at least qualitatively demonstrates that the shear reinforcement essentially influences the kinetic ultimate bearing capacity at the immediate impact area of the reinforced concrete slabs.

7. Next Steps

7.1 The tests concerning item 3.2.1 will be continued.

7.2 For June/July 1978, four additional tests of measuring series I (3.1) had been originally scheduled. This test series is expected to be postponed until September 1979. Up to this date it will be checked again whether the direct measuring of the impact force between missile and target by means of a force measuring system specially constructed to this purpose will deliver useable results at reasonable expenses. Similar force transducers are used at MPA-Stuttgart and at pilot plants of the car industry. Of course, this force measuring system at the impact area can be applied for the tests of measuring series I only (see item 3.1), since the size of the force transducers would entail a falsification of the state of destruction for the test slabs of measuring series II (see item 3.2). In order to enable a comparison of the tests with and without direct impact force measuring, the essential measuring systems for the determination of the impact force used during previous tests (high-speed camera, telemetry, reaction force measuring at the back of the target) must also be provided for all future tests.

7.3 The documentation and evaluation of the available measurement data of the tests appertaining to 3.1 and 3.2.1 will be continued.

7.4 Completion of the Technical Reports I (objective, working program, description of the test plant, basic equations of the calculation models) and II (documentation of the abutment design, measuring program).

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	3.2	RS 121

Vorhaben/Project Title	Land/Country
Das Tragverhalten quergestoßener Stahlbetonbauteile bei geregeltem Stoßkraft-Zeit-Verlauf Experimental Studies Concerning Energy Absorption of Reinforced Concrete Members Subjected to Impact Load (Closed loop testing).	FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Bundesanstalt für Materialprüfung (BAM)

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.10.1977	31.12.1980 (31.3.1982)	Brandes, Struck
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31.12.1978	

1. General Aim

The safety analysis of nuclear power plants includes the consideration of "external events" like an aircraft crash to ensure a consistent reliability level to the different types of risks.

To close some gaps related to the knowledge of characteristic mechanical data and failure mechanism mainly experimental investigations have to be performed to evaluate the effects and data and - beside this - to verify the different computer codes applied in the designing of containment structures.

2. Particular Objectives

The purpose of the experimental investigation, which shall be performed during the project RS 121, is to describe the mechanical behaviour of reinforced concrete structural members, impacted by deformable missiles, in a more sophisticated manner than it could be done up to now. For this reason on the one hand large-scaled specimens are needed to model the "internal structure" of reinforced concrete members in a realistic way and on the other hand different types of specimens have to be investigated to include the different appearing effects as e.g. punching shear and strain rate effects of the value of the yield moments of beams.

The dynamic (kinetic) ultimate load bearing capacity is influenced by several factors which must be varied also in the experimental investigation: Thickness of slab or beam; type, location and percentage of bending reinforcement; stirrups; impact load-time-curve.

3. Research Program

The investigation is directed to experiments performed on reinforced concrete beams and slabs and to the evaluation of characteristic value of different mechanical models starting from the tests results.

3.1 Tests

Reinforced concrete specimens (beams or slabs) are deformed by the piston rod of the hydraulic cylinder in such a way that the ultimate deflection is nearly reached within a certain impact time T_s (40...200 milliseconds). The tests are performed using either an impact load-time-curve or a deflection-time-curve.

3.1.1 Values of parameters of the specimens

Reinforced concrete beams (span width l = 272...576 cm):

thickness $h = 12...60$ cm
ratio of reinforcement $\rho = 0.4...1.5$ %

Reinforced concrete slabs (square, supported at the 4 corners in a distance of 256 cm):

thickness $h = 16...22$cm
ratio of reinforcement $\rho_x = \rho_y = \rho_x' = \rho_y' = 0.4...1.5$ %

Shear reinforcement (stirrups) are varied additionally.

3.1.2 Impact load

The impact load is described as a function of time, characterized by the impact time and certain shapes of this function, one of them is shown in fig. 3.

3.2 Evaluation

The evaluation is directed on the one hand to the filtering and smoothing of the measured values of the different physical quantities of the tests, and on the other hand to the determination of the characteristic values of different mechanical models, which are used to describe theoretically the behaviour of the reinforced concrete specimens.

4. Experimental Facilities; Computer Codes

4.1 Testing setup

The tests are performed with a servohydraulic cylinder, fastened to a rigid frame (see fig. 1), operating under closed loop control. The hydraulic capacity of the cylinder is shown in fig. 2.

4.2 Evaluation

During a single test the measured values are stored in a computer working under real time processing. The evaluation of the characteristic values can be done after each test by using different computer codes.

5. Progress to Date

The testing setup was installed and tested by performing some tests on beams and slabs.
A slab without stirrups failed by punching shear.

6. Essential Results

The results of the first experiments on large scale specimens (see fig. 5) using the just installed testing setup verified the determinations evaluated in a pilot project stated in ref. [1]: The failure-deflection is obviously influenced by the deflection-velocity.

7. Next Steps

In the near future the work will be related to the evaluation of two effects:

- influence of parameter variation (cross section, ratio of reinforcement, stirrups, strength of concrete, deflection-velocity, load-time-curve) on the behaviour of reinforced concrete beams,
- influence of stirrups and impact load-time-curve upon the punching shear failure of slabs.

To verify the different hypothesis in this field a large number of tests is required.

8. Relations to Other Projects

The work is closely related to the projects

RS 165/149 Ultimate Bearing Capacity of Reinforced Concrete Plates under Time-Dependent Loads (e.g. Aircraft Crash) and: Investigation of the Resistance of Concrete Structures to Crashing Aircraft (Hochtief AG, Frankfurt; BWB Meppen).

RS 337 Grenztragfähigkeit von Stahlbetonbalken bei großer Belastungsgeschwindigkeit (Institut für Beton und Stahlbeton, Universität Karlsruhe).

9. Reference Documents

[1] Limberger, E.; Brandes, K. und Herter, J.: Grenztragfähigkeit von Stahlbetonbauteilen unter Stoßbelastung. Amts- und Mitteilungsblatt der Bundesanstalt für Materialprüfung (BAM) 7, Nr. 3 (1977), S. 149-152

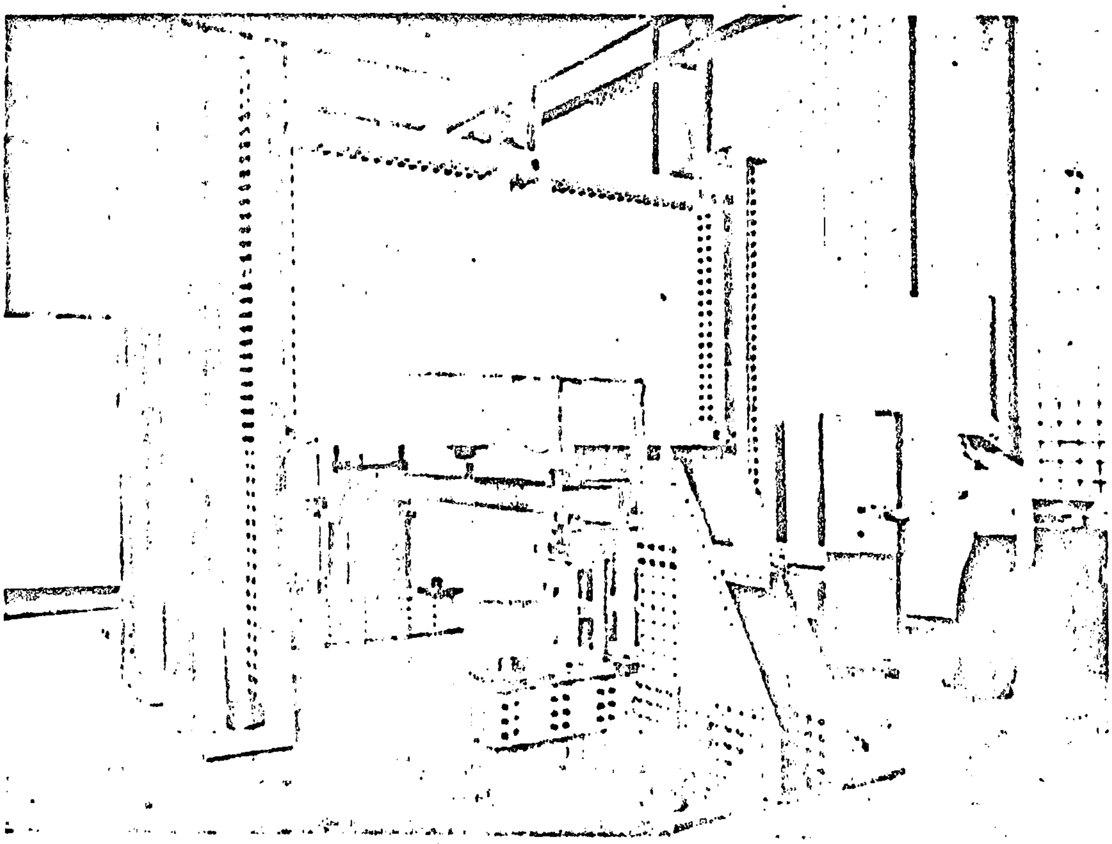


Fig. 1: Testing machine for large scale specimens

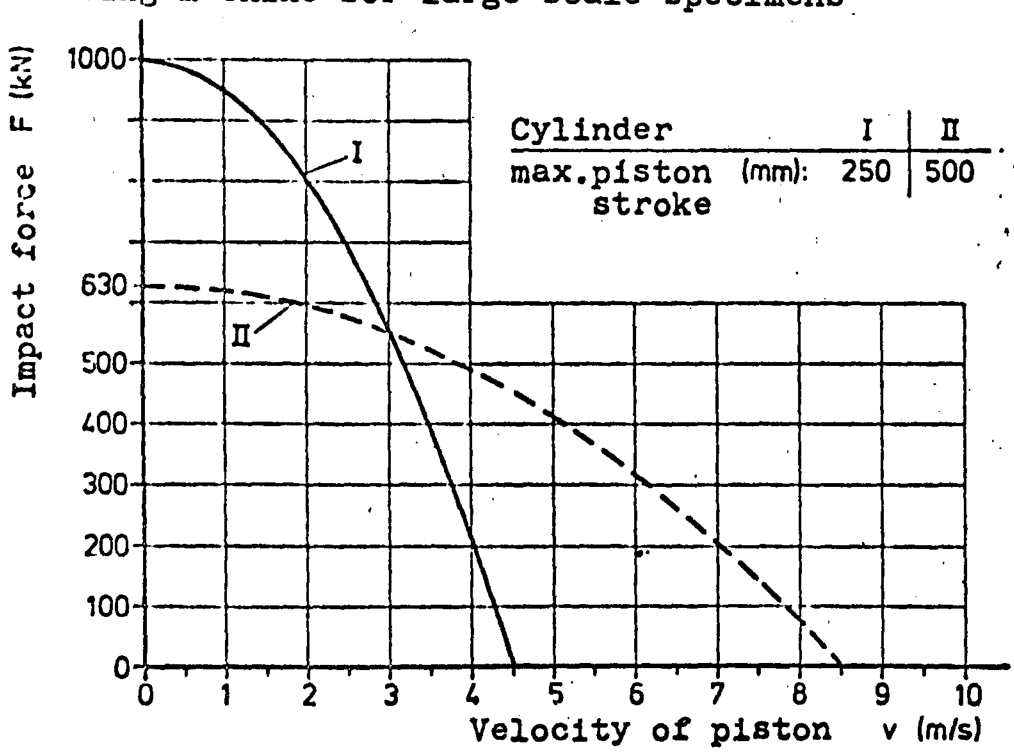


Fig. 2: Hydraulic capacity of the cylinder of the testing machine (steady state).

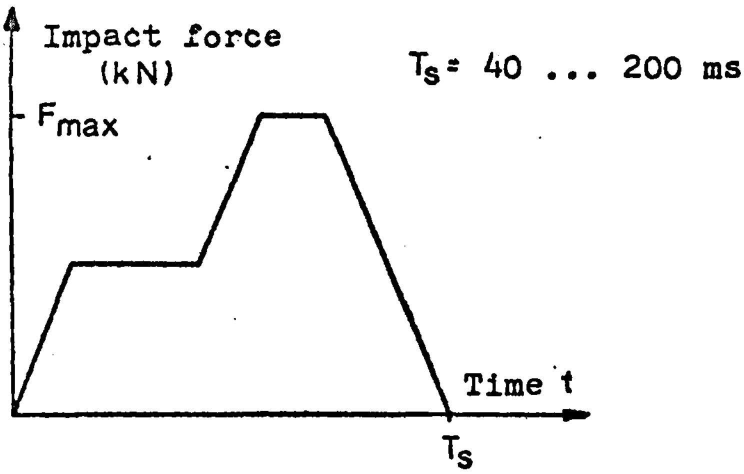


Fig. 3: Shape of impact force-time-curve prescribed to simulate aircraft-crash-effects on containment structures

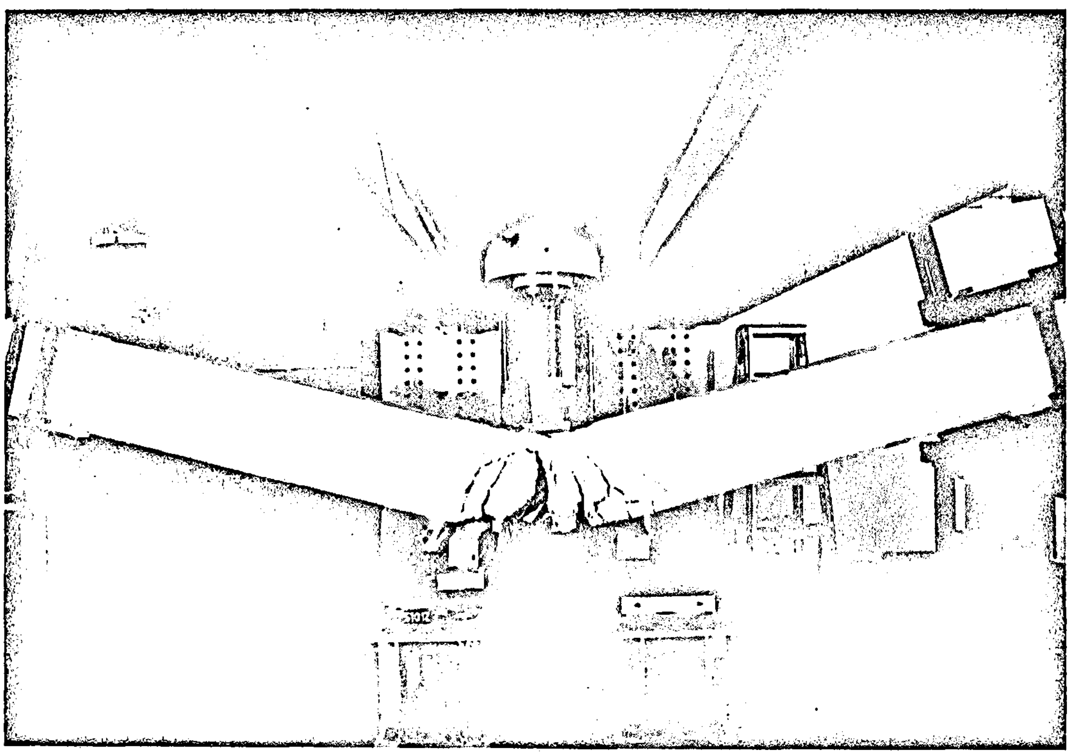
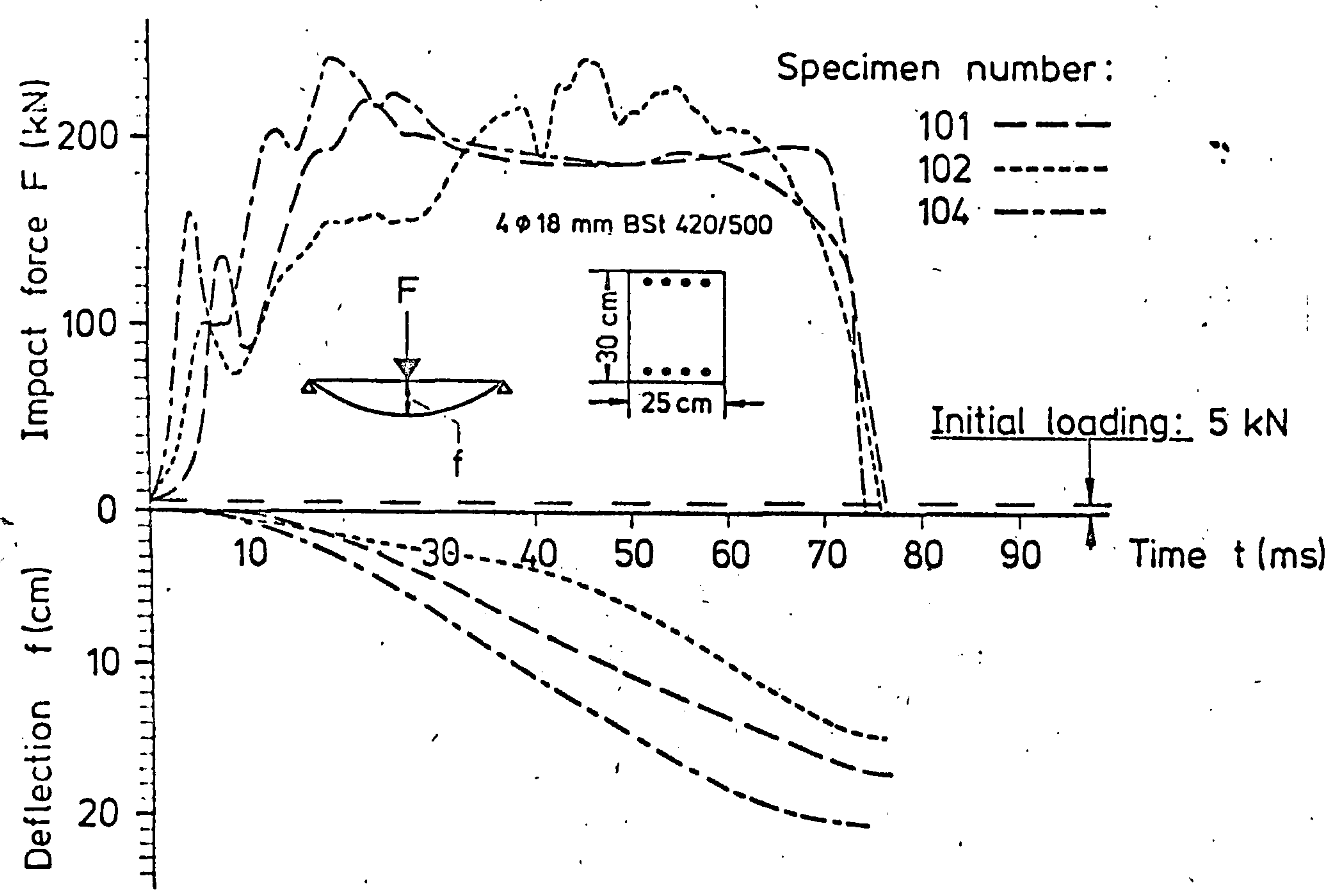


Fig.4: Reinforced concrete beam after test.
Impact test using a time-deflection ramp up to 17,6 cm within 75 ms. In the following static test the beam failed at 19,5 cm total deflection



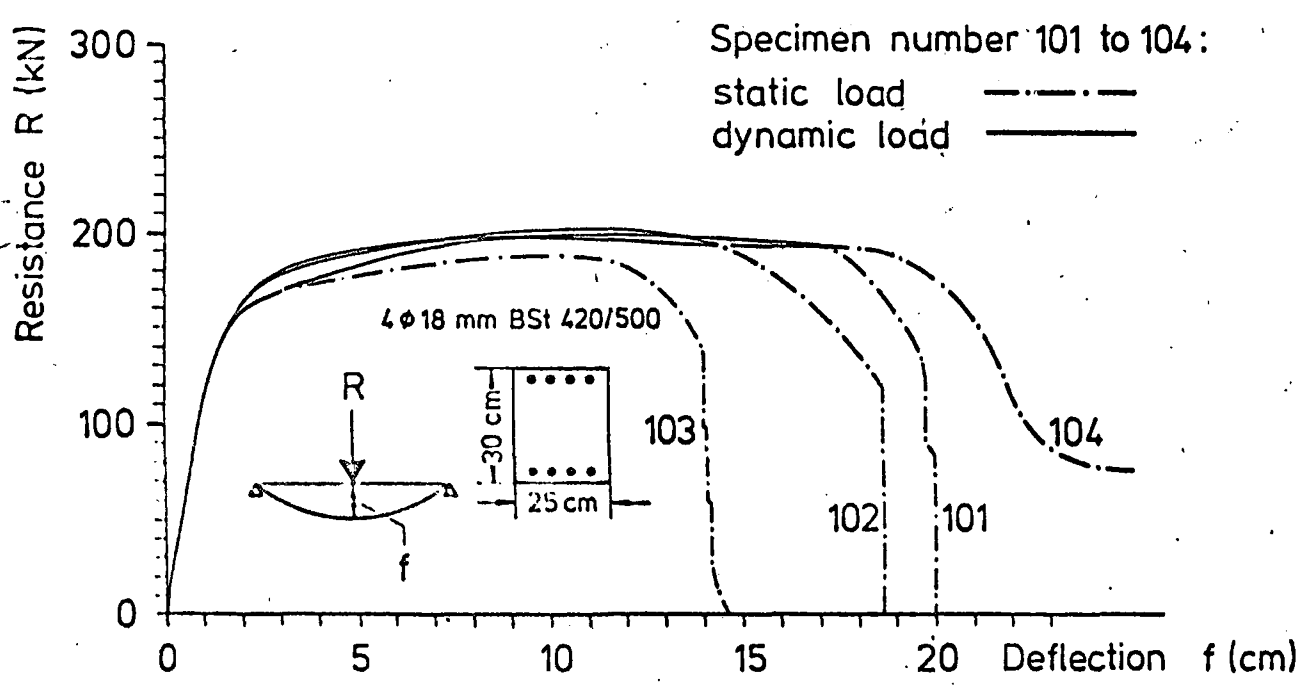


Fig. 5: Behaviour of reinforced concrete beams under impact load. The upper graphs show impact-load-time-curves and deflection-time-curves, the lower graphs resistance-deflection-curves evaluated from the results, which are partly shown in the upper graph.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.4.1978-31.12.1978	3.2	RS 337
Vorhaben/Project Title Grenztragfähigkeit von Stahlbetonbalken bei grosser Belastungsgeschwindigkeit Ultimate Bearing Capacity of Reinforced Concrete Beams under Time-Dependent Load		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Universität Karlsruhe
Arbeitsbeginn/Initiated 1.5.1978	Arbeitsende/Completed 30.4.1979	Leiter des Vorhabens/Project Leader Dr.-Ing.Henseleit
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General aim

Under special loads caused for example by impact of a fast flying military aircraft, earthquake or explosion, reinforced concrete structures are subject to stresses where a very high rate of loading velocity is given. In order to analyse a structure reliably and yet economically for such dynamic loads also, it is necessary to know the bearing capacity and the behaviour of the deformation of reinforced concrete for all these cases.

In addition, one has to know the rotation capacity of "plastic hinges", the resistance to punching and shear failure as well as the bond between steel and concrete near crushing to be able to predict the type of failure. Mathematical equations (here called constitutive relations) for structural elements of reinforced concrete are needed to describe the relation between stress resultants and distortion resultants.

2. Particular objectives

Based on the methods commonly used for nuclear power plant construction, reinforced concrete beams are designed, produced and loaded to failure by fast impacts of forces.

Depending on time, the following will be measured:

- Load, bearing reaction
- Deflection
- Steel strain without disturbing the bond properties
- Concrete strain

The test data are given in a prepared digital form and the results of the evaluation are given as graphs. Based on these results, a Finite Element computer programme, which is being further developed, will be checked.

An important secondary aim of generally practical relevance is the enlargement of knowledge on those dynamic material properties needed for the hypothetic constitutive relations.

The project RS 149/165 will profit from this with the aim of defining the free parameters of this simpler one-dimensional case.

3. Research programme

The following three major parameters will be analysed:
3.1 Rates of loading
3.2 Concrete strength
3.3 Percentage of reinforcement

Beyond the scope of this project with test specimens of relatively small dimensions, the basic results can be applied to structural elements with greater dimensions, like in nuclear power plant constructions.

Beams with the dimensions given in Fig. 1 will be used. The placing of the measuring equipment intended for the test beams can be taken from Fig. 2.

To shorten the test programme it has been planned, for the time being, to use only one kind of steel, i.e. the reinforcement steel BSt 420/500 RU particularly suitable for the assigned purpose.

Restrictions can be made also with regard to the concrete qualities to be tested, because in power plant construction only strength classes ≥ B25 are used. It is foreseen to choose the classes B25 and B45.

In detail, the following tests are planned:
3.1 Three static crushing tests (pilot tests) on the beam given in Fig. 1 with a strength class B25 will be performed.
3.2 Three cruching tests applying the load in the time of 100 ms will be performed on the beam given in Fig. 1 for the strength class B25.

3.3 Twelve crushing tests applying the load in the time of 40 ms will be performed on the beam given in Fig. 1.

For this series of tests, the following parameters are planned:

a) For the concrete strength class B25, three test specimens with the same percentage of reinforcement.
b) For the concrete strength class B45, three test specimens with three different percentages of reinforcement.

The restriction to one percentage of reinforcement in the strength class B25 seems justifiable, because it is assumed that sufficient knowledge will be gathered on the influence of the percentage of reinforcement from the tests with the strength class B45.

Under these conditions, the testing of some 18 test beams should be sufficient.

4. Experimental facilities, computer codes

The experiments will be made on the test floor of the Institut für Beton und Stahlbeton. Test floors as test arrangements for structures will enable the loading equipment (for example testing frames, hydraulic actuators, etc.) to be assembled piece by piece. The equipment can therefore be conveniently adjusted to the respective test conditions and test specimen dimensions. It is possible to generate any load-time-function or deflection-time-function with a recently bought electronic servo-hydraulics together with a hydraulic actuator. The exceptional loading cases "impact of a fast flying military aircraft", "earthquake" or "explosion" can thus be imitated in a very realistic way. In order to obtain the very high increase rates of loading especially in the loading case "impact of a fast flying military aircraft", the following pieces were obtained according to the contract dated 25/4/1978:

- Servovalve with a capacity of 1800 l/min
- Pressure storage
- Strain gauges
- Electronic elements
- Operation amplifier

A Finite Element computer programme will be set up to control the experimental results by a respective analysis. The test data will be analysed graphically using a computer code.

5. Progress to date

5.1 Three static crushing tests

5.2 Three tests applying the load in the time of 100 ms, concrete strength class B25, reinforcement percentage μ = 0,43%

5.3 Three tests applying the load in the time of 40 ms, concrete strength class B25, reinforcement percentage μ = 0,43%

5.4 Three tests applying the load in the time of 40 ms, concrete strength class B45, reinforcement percentage μ = 0,43%

5.5 Setting up and testing of a Finite Element computer code for the analysis of beams loaded by impacts

5.6 Setting up and testing of computer codes for the evaluation of the impact tests.

6. Results

Because the evaluation is not yet completed, no results can be given according to the programme of the project.

7. Next steps

7.1 Evaluation of the experiments

7.2 Further development of the computer codes

7.3 Steel tensile tests applying high velocity rates of strain.

8. Relation to other projects

RS 149/165 - load capacity of reinforced concrete plates applying high velocities of loading and research on resistance of concrete structures to impact by an aircraft.

RS 121 - Impact on reinforced concrete construction elements.

9. References

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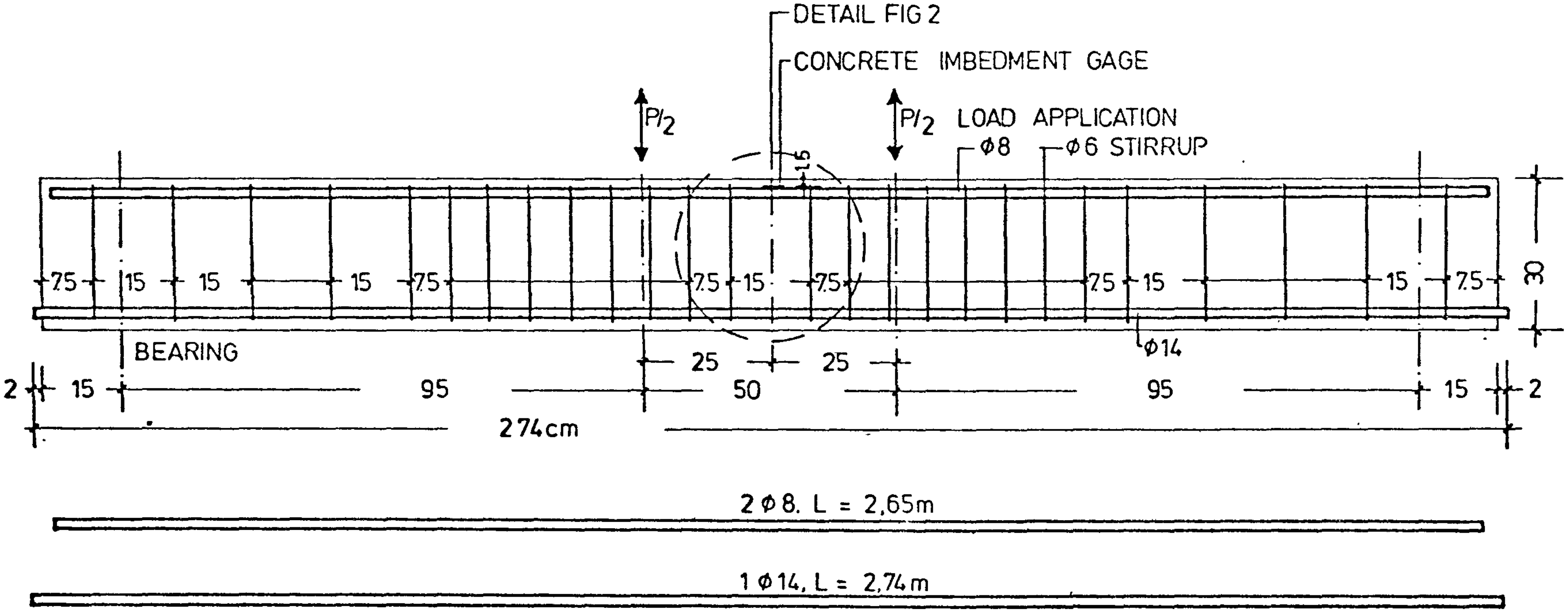
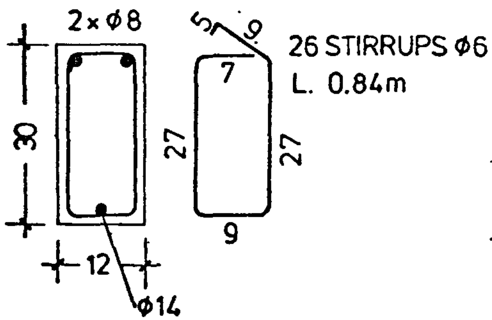
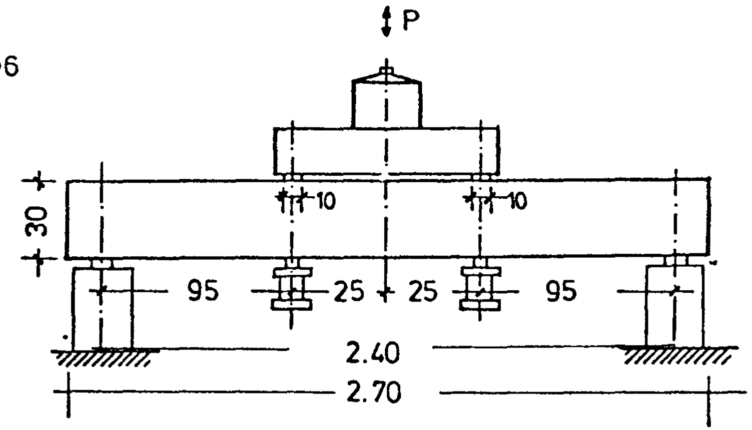




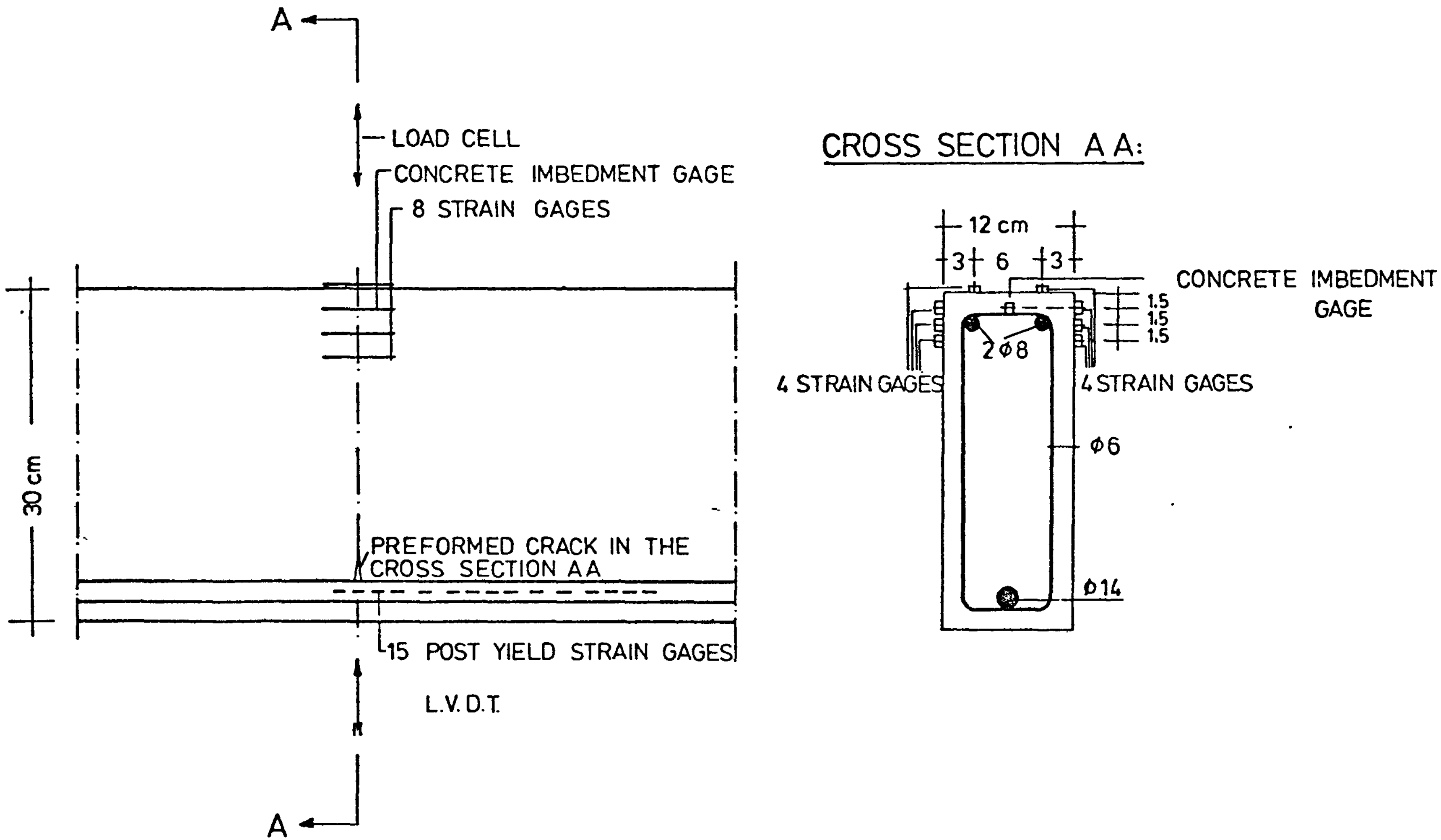


TEST SPECIMEN

B 25, BSt. 420/500 RU

PROJECT No. 7840180409
KARLSRUHE, AUG. 1978

FIG. 1



RS 337

DETAIL OF TEST SPECIMEN
IN THE MIDDLE OF THE BEAM

B 25 BSt. 420/500

PROJECT No. 7840180409
KARLSRUHE, AUG. 1978

FIG. 2

142.1.04/4111.06		3-2
TITRE COMPORTEMENT LOCAL DES ENCEINTES EN BETON SOUS L'IMPACT D'UN PROJECTILE RIGIDE.		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) LOCAL BEHAVIOUR OF REINFORCED CONCRETE WALLS UNDER HARD MISSILE IMPACT.		Organisme exécuteur CEA/DEMT-CEA/CESTA Responsable
Date de démarrage 1974	Etat actuel EN COURS	Scientifiques
Date d'achèvement 1979	Dernière mise à jour DECEMBRE 1978	

1. OBJECTIF GENERAL

- 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").

- Cette étude doit permettre également de mettre au point un programme de calcul utilisable dans un code aux éléments finis.

2. OBJECTIFS PARTICULIERS :

1/ Calculs d'évaluation préliminaires.
2/ Recherche de conditions typiques pour maquettage.
3/ Etude de l'influence du diamètre et de la masse de projectile.
4/ Etude de l'influence du feraillage.
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 du projectile.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME :

- Les essais sont réalisés au Centre d'Etudes Scientifiques et Techniques d'Aquitaine (CESTA) à l'aide d'un canon à air comprimé de Ø 300 mm sur des dalles de 1,46 x 1,46 m.

L'utilisation d'un sabot en bois à l'arrière du projectile permet d'obtenir les possibilités suivantes :

$100 < \emptyset < 300$ mm
$15 < M < 300$ kg

Dans chacun des essais on cherche à obtenir une vitesse d'impact voisine de celle qui est juste nécessaire pour perforer la dalle en béton.

- Le programme proprement dit a pour but, dans un premier temps, d'établir une formulation permettant de relier la vitesse critique (vitesse d'impact minimale pour laquelle le projectile traverse la dalle) aux caractéristiques de la dalle (épaisseur, densité et répartition du ferraillage, résistance du béton) et du projectile (masse, diamètre et vitesse).

- L'étape suivante (fin 79) consistera à mettre au point un programme de calcul utilisable dans un code aux éléments finis.

4. ETAT DE L'ETUDE

41. Avancement à ce jour

Les travaux effectués en 78 ont porté essentiellement sur les points suivants :

- Essais au CESTA pour étudier l'influence des paramètres suivants :
  . Augmentation de la résistance du béton (σ) en fonction du temps et conséquences éventuelles sur le comportement des structures vis-à-vis de la perforation,

  . Forme du projectile : application au projectile turbine du CP1.

- Définition d'un programme d'essais à basse vitesse (de l'ordre de 20 m/s) pour évaluer à surface égale l'influence de la forme de la section d'un projectile à nez plat et l'influence d'un nez de forme quelconque.

42. Résultats essentiels

- Bien que le rapport d'essai ne soit pas encore disponible, on peut dégager les orientations suivantes :

  . la "fragilité" des bétons à fort σ ($\sigma \gg 500$ bars) se confirme (diminution de 10 % de la vitesse de juste perforation quand on passe de 450 à 600-700 bars). Il y a lieu toutefois d'être très prudent car cette tendance doit être comparée et validée par les résultats des essais avec vieillissment naturel des bétons (tirs à 1, 2 et 5 ans).

  . Pour des projectiles de même surface mais de forme différente, on peut noter que la formule de juste perforation donne des résultats conservatifs pour des projectiles de forme triangulaire, rectangulaire.

  Le problème posé est l'évaluation du diamètre équivalent à prendre en compte pour un projectile de section et de forme données.

5. PROCHAINES ETAPES :

Les prochaines étapes prévues en 79 visent les buts suivants :

1/ au niveau de la formulation proprement dite :

- Influence de la résistance du béton pour apprécier la fragilisation due au vieillissement naturel (essai à 1 an).

- Influence de la section du projectile : cas particulier du projectile turbine pour évaluer le diamètre équivalent à une section rectangulaire donnée : Essais à basse vitesse programmés à Saclay.

- Influence de vitesses supérieures à 200 m/s.

2/ au niveau des modèles de calcul :

- Mise au point d'un modèle béton à introduire dans un code aux éléments finis pour calculer la pénétration et la perforation de dalles en béton non armé tout d'abord.

- Deuxième étape ensuite avec prise en compte du ferraillage.

6. RELATION AVEC D'AUTRES ETUDES :

Les relations sont les suivantes :

- avec EdF : les essais EdF sont terminés, mais il existe une liaison technique au niveau du groupe de travail DSN-SEPTEN sur la perforation du béton,

- avec les pays de la Communauté :

  Accord UKAEA-CEA : signé le 19 septembre 1978
  Accord BMFT-CEA : signé le 28 septembre 1978.

7. DOCUMENTS DE REFERENCE :

Les principaux documents établis en 78 sont les suivants :

- Rapport EMT 78-27 du 19/4/78 : Programme CAPRI XV, XVI.
- Rapport EMT 78-34 du 14/6/78 : Programme d'essais à basse vitesse.
- Rapport des essais 78 (à paraître)
- Rapport du CEBTP 912-7-020 du 25/10/78 : essais de résistance sur éprouvettes Ø 16 H 32
- Communication à l'American Society of Civil Engineers (ASCE) prévue lors de la National Convention in Boston (Avril 1979)
- Rapport DTech-STA 184 du 24/11/78 : résistance des bétons des dalles 64 à 67.

TNO-IBBC	CLASSIFICATION: 3.2/3.3/7.1.
TITLE: Responsieberekeningen voor reactorgebouw	COUNTRY: THE NETHERLANDS
	SPONSOR: Ministry of Social Affairs; Ministry of Public works (DIV); TNO-IBBC ORGANIZATION:
TITLE (ENGLISH LANGUAGE): Dynamic response of reactor structures (building and containment)	PROJECTLEADER: Kusters
INITIATED: June 1974 — LAST UPDATING: June 1979	SCIENTISTS: Kusters, de Groot, de Witte, Tolman.
STATUS: Progressing — COMPLETED: 1979	

General aim

Development and application of calculational tools for the evaluation of structural response of reactor buildings under dynamic loading conditions.

Particular Objectives

The calculations are directed to the evaluation of the effects of
(1) pressure/blastwaves of gas explosion in the vicinity of the site
(2) impact of striking aircraft on the reactor (containment) buildings
(3) investigation of the load-carrying capacity of the buildings in the vicinity of the reactor
(4) investigation of the load-carrying capacity of flat plate panels of reinforced concrete, subjected to local dynamic loads.

Computer program facilities

1. Finite element program DIANA
2. Special purpose program to calculated load-carrying capacity of flat plate planels of reinforced concrete, subjected to local dynamic loads [4].

Project status

The investigation to the load-carrying capacity is closed [4].
The investigation to the necessary computational methods to make available within DIANA the method to analyse the structural response of reactor buildings under dynamic loading conditions, including the non-linear behaviour of reinforced concrete is closed [5].
The implementation of this method has started.

Next steps

Investigation of the load-carrying capacity of the buildings in the vicinity of the reactor.
To finish the implementation of the computational method in the computer code of DIANA. To make a start of the evaluation of the implemented methods.

Relation to other projects

None.

Reference documents

1. "Mathematical description of the non-linear behaviour of reinforced concrete" by Ir. H. Geertsema, June 1976, TNO-IWECO, report nr. 11261/2.
2. "Responsieberekeningen aan een reactorgebouw voor belastingen t.g.v. een drukgolf en een neerstortend vliegtuig" ("Structural response analysis of a reactor building subjected to a blast wave and aircraft impact") by Ir. H. Geertsema, September 1976, TNO-IWECO, report nr. 11261/3.

3. "Onderzoek naar een mogelijke verbetering van het materiaalgedrag voor de berekening van gewapende betonkonstrukties", by Ir. G.M.A. Kusters and Ir. A.K. de Groot, February 1978, TNO-IBBC, report nr. B-78-60/64.7.0128.
4. "Gevelplaat opgevat als een elasto-plastisch één massaveersysteem" by Ir. A.K. de Groot, December 1978, TNO-IBBC, report nr. B-78-429/62.3.2014.
5. "Veiligheid van kernreactorgebouwen onder extreme belasting, eerste fase" by Ir. G.M.A. Kusters and ir. F.P. Tolman, April 1979, TNO-IBBC report, B-79-175/68.3.0158

Degree of availability

Through: Ministry of Social Affairs
TNO-IBBC

Budget Hfl. 530.000.-

Hfl. 470.000.- (1975, 1976, 1977, 1978)

Approx. Hfl. 150.000.- (1979)

	CLASSIFICATION:
TITLE (ORIGINAL LANGUAGE): THE EXPERIMENTAL AND THEORETICAL STUDY OF LOCAL EFFECTS IN THE IMPACT OF MISSILES OR STRUCTURES	COUNTRY: UK
	SPONSOR: SRD
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/AEEW UKAEA
	PROJECT LEADER: A R EDWARDS (M L BROWN DEPUTISING)
INITIATED: MAY 1977 / COMPLETED: 1980/1 APPROX	SCIENTISTS: SRD N CURTRESS; AEEW P BARR, A NIELSON
STATUS: IN PROGRESS / LAST UPDATING: 24 APRIL 1979	

1. GENERAL AIM

To develop theoretical methods for assessing the local effects of typical accidental missiles impacting upon structures. To validate these methods against experiment.

2. PARTICULAR OBJECTIVES

Work is to be continued on the input of missiles on concrete walls. This work was begun at AWRE Foulness. In particular scale modelling effects are to be investigated. This is to be done in co-operation with other European experimental programmes where possible. (France, Germany). The theoretical modelling of concrete behaviour is to be developed using the SRD finite difference code SARCASTIC as a framework. Impacts on metal plates in the below-ordnance (≤ 350 ms^{-1}) velocity region are to be studied. Experimental results are to be used to assess the performance of available finite element structural codes (CADROS/DPS, EURDYN, NONSAP) in this type of problem. A version of SARCASTIC, SARPLASTIC, is to be used as a vehicle for developing material modelling techniques where necessary.

3. EXPERIMENTAL FACILITIES AND PROGRAMME

Gravity drop tests are available and being used for light (tens of Kg) missiles at velocities up to about 20 ms^{-1}.

A compressed air missile launcher has been completed and is undergoing commissioning trials (January 1979). Initially missiles up to a few tens of kilogrammes at velocities up to about 200 ms^{-1} will be used. Maximum diameter will be 150 mm. At a later stage of the programme a 300 mm diameter launch barrel will be added. This will enable missiles of hundreds of kilogrammes to be launched at velocities up to around 250 ms^{-1}.

High speed transducers with associated data processing will be used together with high speed photography to monitor the impacts. These facilities are all available at present. A concrete construction and testing laboratory has been assembled and is capable of producing up to four concrete targets per week, with all associated quality control testing.

The initial programme of works consists of calibrating the launcher and running a series of confirmatory experiments for comparison with earlier work at Foulness. This is expected to take up to June or July 1979.

The programme for subsequent work is not yet finalised and will dep nd in part upon what, if any, arrangements are made for co-operation with other European research programmes.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.4.78 - 31.12.78	3.3	RS 286
Vorhaben/Project Title Diffraction of Shock Waves Analysis of Experimentally Ascertained Shock Waves on Reactor Buildings Beugung von Druckwellen Analytische Erfassung der experimentell ermittelten Druckwellen auf Reaktorgebäude		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor SDK Ingenieurunternehmen 7850 Lörrach
Arbeitsbeginn/Initiated 1.4.78	Arbeitsende/Completed 31.12.78	Leiter des Vorhabens/Project Leader Hofmann / Huber
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

An attempt is made to develop a concept for the analytical treatment of the diffraction of shock waves in the vicinity of reactor buildings - using latest engineering techniques. Results gained from the analytical calculations are compared with experimentally determined results. They should be sufficiently accurate for reactor safety purposes in connection with chemical explosions.

The experimental data are determined in model experiments carried out at the Ernst-Mach-Institut (RS 102-09).

2. Particular Objectives

Determination of the time-dependent pressure distribution on a simplified cylinder-model (plane problem), subjected to a pressure wave, having the form of a step-function. In a first step the problem is described by the linear differential equation for wave propagation, which includes diffraction and "regular reflection" phenomena.

If the comparison of the results with those of the experimental investigations, which are presently not known, show intolerable discrepancies, additional non-linearities and possibly flow-phenomena will be taken into account.

Another objective is to develop criteria which permit judgement of whether research efforts can be successfully extended to more complex problems as they exist in reality (location of buildings, realistic time function and spacial distribution of the pressure field).

3. Research Program

3.1 Preparations for analysis, simulation of the test model.

3.2 Coordination of work (EMI - SDK) to allow for comparable results.

3.3 Evaluation of time-dependent pressure distribution ($p(\underline{r},t)$) for the entire field, while pressure front is passing over the test body.

3.4 Evaluation, comparisons, report.

4. Computer Codes

Within the scope of this project computer codes are used, which were tested in several applications. The codes are based on the Finite-Element-Method.

5. Progress to Date

ref. 3.1 and 3.4 The determination of analysis-procedures, preparation of data and model to simulate the test-model were carried out first. In the course of the work a second calculation model was prepared, having a finer mesh. For comparison of analytical and experimental results, geometric data and points of comparison were checked. The time-dependent pressure distribution ($p(\underline{r},t)$) was determined for both calculation models.

The essential parts of the analytical work are documented.

6. Results

Analytical results, when plotted in a pressure-time graph, already show diffraction and "regular reflection" effects for a relatively coarse mesh.

7. Next Steps

ref. 3.4 Comparison of analytical and experimental results. This will be carried out by EMI, since the results are not presently known to SDK. Final work on the report.

8. Relation with Other Projects

RS 102-09

Ernst-Mach-Institut, Freiburg i.Br.

9. References

-

10. Degree of Availability of the Reports

-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	3.3	RS 318

Vorhaben/Project Title	
Vorbereitung Gasexplosionsforschungsprogramm Preparation of a research program on gas explosions	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Battelle-Institut e.V.

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.12.77	31.12.78	Geiger
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

The possible magnitude of the pressure loading of nuclear power plants due to external gas explosions can at present be estimated only with caution. To investigate the problems involved in the explosion of unconfined gas clouds to a greater depth and to improve the methods of safety assessment, a Gas Explosions subprogram within the External Events project of the BMFT Reactor Safety Research Program is to be established.

2. Particular Objectives

To prepare the Gas Explosions subprogram, the following tasks have to be accomplished:

- Preparation of the documents for the tendering procedure (detailed specification for each individual problem area)
- Actual tendering procedure.

3. Research Program

3.1. Preparation of the documents for the tendering procedure

- Selection and compilation of the individual investigations to be performed within the subprogram, giving reasons for the structure of the subprogram .
- Specifying each individual investigation and, in addition, the task of scientific management of the subproject, by precisely defining the respective tasks and the expected results

- Establishing a document giving general instructions for the bidders, especially with a view to dividing the proposed investigations into several phases and adapting them to other research programs

3.2. Performance of tendering

- Specification of the tendering procedure in close co-operation with BMFT and GRS/FB
- Distribution of the tender to the potential bidders
- Answering inquiries by the bidders.

4. Experimental Facilities, Computer Codes

-

5. Progress to Date

Re 3.1: The invitation to tender documents were drawn up and agreed by BMFT and GRS/Fb

Re 3.2: The invitation to tender documents were mailed to about 20 potential bidders and to about 20 official institutions interested in the subject by October 30, 1978. Additional copies of the documents were sent to interested institutions after that date. The invitation to tender was also published in several journals in December 1978 and January 1979.

6. Results

Re 3.1: Taking into consideration the investigations on gas explosions within other programs (PNP Safety Research Program and foreign programs), three problem areas were defined for the subprogram:

- Conceivable mechanisms for the initiation of detonation-like explosion modes.
- Relation between the characteristics of the impacting pressure wave and the structural response.

- Potential hazard of detonation of the real gas cloud.

Within these problem areas, 16 individual investigations (tasks) were defined by specifying the individual problems, the investigations envisaged to treat the problem, the expected results and, finally, the relation to other programs. Furthermore, the task of scientific management of the project was specified. The invitation to tender documents consist of the general invitation to tender letter and three enclosures. Enclosure 1 outlines the objectives and the structure of the subprogram, enclosure 2 specifies the individual tasks in the way described above, and enclosure 3 gives recommendations concerning the format of proposals.

Re 3.2: In general, several tenders were received for each of the individual tasks. The deadline for submitting tenders was fixed at January 19, 1979.

7. Next Steps

Extending the original objectives, the following tasks will be performed during the first quarter of 1979:

- Evaluation of the tenders received.
- Elaboration of a proposal for the awarding of contracts.

8. Relation to other Projects

PNP Safety research program, "Process Gas Release - Explosions in the Gasification Plant and Pressure Loading of the Containment" of subprogram; intended gas explosion research program of the Commission of the European Communities.

9. References

-

10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.Jan.78-31.Dez.78	3.3	RS 265
Vorhaben/Project Title Untersuchung über die Einwirkung schädlicher Stoffe auf Kernkraftwerke Analysis on the Effects of Dangerous Materials on Nuclear Power Plants		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Battelle-Institut e.V. Frankfurt/Main
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
July 1, 1977	March 31, 1979	Dr. H. J. Nikodem
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
continuing	Dec. 31, 1978	DM 485.470,--

1. General Aim

Identification of possible effects and their description, and consequences of the exposure of components of critical systems and personnel of nuclear power stations to dangerous materials; proposals for measures to reduce the risk.

2. Particular Objectives

2.1 Identification of possible points of attack by dangerous materials.

2.2 Description of mechanism from exposure to dangerous materials, determination of critical exposures.

2.3 Listing and evaluation of potential dangerous materials and their sources.

2.4 Selection and description of critical events.

2.5 Quantitative evaluation of critical events.

2.6 Catalogue of proposed measures to reduce the risk.

3. Research Program

3.1 Analysis of the reference plant Biblis B.

3.2 Determination and description of the effects of potentially dangerous materials.

3.3 Compilation of groups of dangerous materials.

3.4 Transport behaviour of dangerous materials.

3.5 Determination and description of external sources of dangerous materials.

3.6 Compilation of dangerous materials, their sources, and chains of events which lead to effects on sensitive elements.

3.7 Compilation of system failures in nuclear power plants which may occure by attacks of dangerous materials.

3.8 Identification of critical chains of events.

3.9 Quantification of external sources of dangerous materials.

3.10 Quantification of critical events.

3.11 Compilation of critical dangerous materials, sources and their risk potential.

3.12 Catalogue of proposed protective measures, estimates of the expected safety gains.

4. Experimental Facilities, Computer Codes

-

5. Progress to Date

To 3.1 - 3.6 The tasks of phase I of the project were completed. Dangerous materials were identified, their effects on sensitive elements were described. Scenarios were developed which describe the source of dangerous materials, its transport, and its effect on sensitive elements of the nuclear power station. Results were presented for comments.

6. Results

The preliminary results of phase I were reported to the sponsor, to experts, and to the manufacturer and the owner of the reference plant in a presentation and in a written form.

7. Next Steps

Completion of the project.

8. Relation with Other Projects

-

9. References

-

10. Degree of Availability of the Reports

-

122-2 - 02/4175-10		3-3
TITRE Agressions d'origine externe sur les installations nucléaires : explosions chimiques non confinées dues à un environnement industriel ou aux voies de communications.		Pays FRANCE Organisme Directeur CEA/DSN EdF/SEPTEN
TITLE (Anglais) External impacts on nuclear plants : unconfined chemical explosions due to industrial environment or to communication routes.		Organisme Exécuteur CEA/DSN/SESRS CEA/CESTA et DEMT ENSMA Responsable J. DUCO (CEA/DSN/FAR) T. GOBERT (EdF)
Date de démarrage 01/01/1976	Etat actuel en cours	Scientifiques MM. BROSSARD) ENSMA LEYER) M. PERROT CEA/CESTA M. ROCHEDEREUX CEA/DSN
Date d'achèvement 31/12/1981	Dernière mise à jour 02/01/1979	

1. OBJECTIF GENERAL

Protection des installations nucléaires contre des agressions d'origine externe : cas particulier des explosions chimiques de masses gazeuses dérivantes libérées par un accident dans un environnement industriel ou au niveau de voies de communication.

2. OBJECTIFS PARTICULIERS

2.1. - Recherche de lois d'échelle pour les caractéristiques de l'onde de pression aérienne engendrée par la détonation (au sens strict) d'un mélange air-hydrocarbure.
Si la non détonation ne peut être prouvée, ces lois seront utilisées dans les évaluations de sûreté.
Conditions pour une initiation en détonation.

2.2. - Recherche de lois d'échelle pour les caractéristiques des secousses telluriques induites par la détonation en surface (au sens strict) d'un mélange air-hydrocarbure.

2.3. - Recherche de l'effet de divers paramètres sur la cinétique d'une explosion : caractéristiques du mélange, géométrie, intensité et localisation de l'initiation, obstacles, confinement partiel, ... - Possibilités pour une déflagration d'évoluer vers une déflagration rapide ou une détonation.

2.4. - Recherche de modèles, utilisables dans les calculs de sûreté, représentant les caractéristiques de l'onde de pression aérienne dans les cas où il n'y a pas de détonation.

2.5. - Interaction onde incidente-structures : mise au point d'une méthode de calcul des charges dynamiques appliquées sur les éléments de bâtiments.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Les essais sont actuellèment réalisés en deux endroits différents :

a/ A l'Université de POITIERS, dans le Laboratoire d'Energétique et de Détonique de M. le Professeur N. MANSON.

Les effets de divers paramètres sur la cinétique de l'explosion (objectif 2-3) sont systématiquement analysés à petite échelle, les mélanges explosibles étant confinés à l'intérieur de bulles de savon hémisphériques de 20 cm de diamètre.

b/ Au Centre d'Etudes Scientifiques et Techniques d'Aquitaine (CESTA).

Des essais de détonation de mélanges air-hydrocarbures sont effectués sur des volumes de mélange de plusieurs dizaines de m^3. Des tirs de 1 000 m^3 sont envisageables sur le terrain d'essais.

Le but poursuivi est l'obtention de données expérimentales pour les objectifs 2-1, 2-2, 2-4 (ultérieurement) et 2-5.

Le programme est défini et réajusté périodiquement par un comité regroupant des experts du CEA, d'Electricité de France et de Gaz de France. Ce comité s'est adjoint, par contrat, le support scientifique des spécialistes de l'Université de POITIERS.

Programme actuel :

a/ A l'Université de POITIERS.

Objectif 2-3. Il s'agit d'essais exploratoires destinés à rechercher les paramètres influant de manière importante sur la cinétique des déflagrations Ces résultats devraient permettre d'orienter un programme d'essais à grande échelle.

Indépendamment du travail expérimental, les experts de l'Université de POITIERS assurent, à partir des essais au CESTA, la modélisation nécessaire à l'objectif 2-1, puis ultérieurement celle correspondant à l'objectif 2-4.

b/ Au CESTA.

Obtention de données expérimentales pour les objectifs 2-1 et 2-2, 2-4.(ultérieurement) et 2-5.

c/ Au DSN et à EdF

Travail de modélisation pour les objectifs 2-1, 2-2 et ultérieurement 2-4.

d/ Au DEMT

Modélisation pour l'objectif 2-5.

.../...

4. ETAT DE L'ETUDE

4.1. - Avancement à ce jour

a/ Objectifs 2-1 et 2-2

Les essais sont relatifs à des tirs de ballons sphériques ou hémisphériques en latex ou en mylar de volumes s'échelonnant de 1 à 215 m^3 de mélange gazeux explosible.
L'allumage se fait par explosif au centre des ballons.

5 essais ont été réalisés sur des mélanges air-propane ; malgré un amorçage par 500 g de plastic, le dépouillement des enregistrements a montré que l'obtention d'une détonation autonome est douteuse : des essais air-propane sur de plus grands volumes sont à envisager.

3 essais de détonation air-acétylène ont été effectués dans des ballons sphériques de 7 m^3 disposés à des hauteurs différentes par rapport au sol pour analyser l'effet de réflexion de ce dernier (capteurs de pression CELESCO et ENSMA).

4 essais de détonation air-éthylène ont été réalisés dans des ballons sphériques de 7 et 31 m^3, ainsi que 2 tirs de TNT (12 et 101 kg), pour établir les correspondances entre les réponses des trois types de capteurs de pression utilisés (CELESCO, ENSMA et KISTLER).

4 essais de détonation air-éthylène ont été effectués dans des ballons hémisphériques de 3 et 15 m^3, afin de rechercher les relations quant aux caractéristiques de l'onde de pression entre les explosions sphériques et hémisphériques (capteurs de pression ENSMA et KISTLER).

2 essais de détonation air-éthylène ont enfin été réalisés sur des ballons sphériques de 215 m^3 (capteurs ENSMA et KISTLER) pour étayer l'interprétation faite pour les 16 premiers essais de détonation réussis en 1976.

En ce qui concerne l'étude des conditions pour une initiation en détonation de certains mélanges air-hydrocarbures, un premier dispositif d'essais est en cours de mise au point : la cinétique de l'explosion sera suivie sur une distance de 10 à 20 m dans un tuyau rigide Ø 60 cm disposé dans le prolongement d'une chambre d'amorçage.

b/ Objectif 2-3

Les travaux en laboratoire n'ont pu commencer qu'en Octobre 1976. Le montage expérimental est terminé et diverses expériences d'orientation ont été réalisées (allumage de faible énergie par étincelle, mélanges oxygène-propane de rayons variables, mélanges concentriques de richesses variées, effets d'obstacles simples).

c/ Objectif 2-4

Etude bibliographique seulement.

.../...

d/ Objectif 2-5

Des maquettes au 1/50 représentant des éléments d'une centrale nucléaire ont été disposées dans le champ de pression lors de certains essais de détonation au CESTA.

4.2. - Résultats essentiels

a/ Objectif 2-1

Une première modélisation et une formule ont été proposées par l'ENSMA en ce qui concerne la surpression de crête de l'onde lancée dans l'environnement.
La validitié prouvée de la formule est actuellement limitée à une gamme de volumes assez restreinte.

b/ Objectif 2-2

L'instrumentation des essais en accéléromètres enterrés ayant été tardive et la cadence des tirs ralentie en 1978, les premiers résultats expérimentaux, bien que très encourageants, sont encore insuffisants pour permettre de tirer des conclusions quantitatives.

c/ Objectif 2-3

Une dissymétrie de la propagation de la flamme en géométrie hémisphérique a été observée pouvant provoquer des effets directionnels dans le champ de pression.
La célérité moyenne de la flamme augmente très rapidement avec le rayon initial de la charge jusqu'à atteindre 110 m/s pour un rayon de 10 cm (rappelons qu'il s'agit d'un mélange propane-oxygène).

Les effets d'une accélération fortuite du front de flamme (provoquée expérimentalement par une variation discrète de la richesse du mélange) sont d'autant plus importants que cette dernière se produit loin de la source d'inflammation.

Les simulations d'un mur vertical en limite du mélange explosible et d'un couloir placé sur le trajet de la flamme n'ont pas conduit, dans les conditions expérimentales, à des augmentations sensibles de la vitesse de flamme.

d/ Objectif 2-5

Les mesures effectuées sur les maquettes ont permis le développement (en cours) au DEMT du code ZEPHYR.

5. PROCHAINES ETAPES

a/ Objectif 2-1

Prolongement de l'interprétation des divers essais de détonation réalisés, dans le but de rechercher de nouvelles modélisations des caractéristiques de l'onde de pression importantes quant à la tenue des bâtiments (impulsion positive, durée de la phase positive, pic de surpression, etc...) ; ces modélisations concerneront essentiellement la gamme de surpressions de crête de 50 à 200 mbars et de-

.../...

vraient présenter une bonne crédibilité quant à leur extrapolation à de grands volumes explosibles.

Recherche (études en tuyau Ø 60 cm) des conditions d'initiation de mélanges d'air et d'hydrocarbures gazeux courants sur le plan industriel pouvant conduire à des déflagrations rapides et éventuellement à la détonation autonome.

b/ Objectif 2-2

Poursuite des mesures à l'occasion des essais de détonation.

Tentative de modélisation.

c/ Objectif 2-3

Poursuite des essais en laboratoire (programme 1979 financé par la C.C.E.). Définition et réalisation d'essais de déflagration à grande échelle (sphères, hémisphères, boyaux).

Une première tranche d'essais air-éthylène en ballons latex sphériques de 10 m^3 a été retenue début 1979 afin d'évaluer l'influence sur la cinétique de l'explosion de la puissance de l'amorçage.

d/ Objectif 2-4

Dépouillement des essais à grande échelle prévus en c/ et recherche de modélisations représentatives, en s'aidant de la bibliographie et d'études échangées dans le cadre d'accords internationaux (UKAEA, BMFT).

e/ Objectif 2-5

Un premier bilan de ce qui a été fait au CEA et de ce qui est disponible dans le cadre des accords d'échange sera effectué. La définition du programme ultérieur en découlera.

6. RELATIONS AVEC D'AUTRES ETUDES

- Formation et dispersion atmosphérique de nappes dérivantes de gaz ou d'aérosols, explosibles ou toxiques, suite à un accident sur une installation chimique ou nucléaire. Fiche 122-2-03.
- Identification et caractérisation des agressions externes dues aux activités humaines. Détermination de leur probabilité d'occurrence (122-2-01)

7. DOCUMENTS DE REFERENCE

11 Flame propagation through unconfined and confined hemispherical stratified gaseous mixtures. Communication au 17th International Symposium on combustion) Paper number 124.

122-2-03 / 4171-10 222-2-03		3.3.
TITRE Formation et dispersion atmosphérique de nappes dérivantes de gaz ou d'aérosols, explosibles ou toxiques, suite à un accident sur une installation chimique ou nucléaire.		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) Formation and atmospheric dispersion of drifting clouds of explosive or toxic gas or aerosols as a consequence of an accident on a chemical or on a nuclear plant.		Organisme Exécuteur CEA/DSN/SESRS Responsable J.P. MAIGNE DSN/SESRS/FAR
Date de démarrage 02/10/78	Etat actuel Lancement	Scientifiques G. DEVILLE-CAVELIN
Date d'achèvement 31/12/82	Dernière mise à jour 02/01/79	

1. OBJECTIF GENERAL

Gaz ou aérosols toxiques : détermination en fonction du temps du champ de concentration dans l'air, au niveau du sol. Dépôts sur le sol ou sur l'eau.

Gaz explosibles (hydrocarbures essentiellement) : détermination, en fonction du temps, des régions où l'hydrocarbure se trouve mélangé à l'air dans une fourchette donnée de concentration (fourchette d'inflammabilité) ; calcul de la masse correspondante d'hydrocarbure.

2. OBJECTIFS PARTICULIERS

2.1. Détermination du terme source

- Cas d'une rupture de gazoduc.
  En particulier, effet de la vitesse initiale du jet gazeux.

- Cas d'une rupture de cuve de gaz liquéfié.
  Effet des caractéristiques du stockage (sous pression, réfrigéré).
  Calcul du débit d'évaporation en cas d'épandage et d'évaporation simultanée sur le sol ou sur l'eau (surface limitée ou non).

- Cas d'une rupture de canalisation de transport sous pression de liquides (ammoniac, ...) : formation d'aérosols.

- Cas de rejets accidentels à l'atmosphère d'installations nucléaires : aérosols sodés (feux de sodium), rejets d'UF6 se transformant à l'air humide en UO_2F_2 et en acide fluorhydrique, ...

.../...

2.2. Dispersion atmosphérique

Modélisation des phases initiales où le gaz ou l'aérosol ne sont pas des polluants minoritaires.

- Cas des gaz lourds par rapport à l'air, des gaz légers, des gaz de densité variable (évolution en température), des aérosols liquides avec vaporisation éventuelle des gouttelettes, des aérosols solides se transformant chimiquement au contact de l'air, etc ...

Raccordement avec les modèles classiques de dispersion atmosphérique de polluant minoritaire.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Etude de faisabilité d'épandage d'éthylène liquéfié et de lâchers d'UF6 sur un terrain d'essais dépendant de CEA/DAM.

4. ETAT DE L'ETUDE :

Lancement.

5. PROCHAINES ETAPES

Poursuite de la compilation bibliographique. Bilan des informations obtenues dans le cadre de l'accord d'échanges CEA-UKAEA. Définition des études à entreprendre en priorité.

A noter que certains objectifs particuliers précédents, intéressent le Ministère de l'Industrie (Direction des Carburants), l'Union des Chambres Syndicales des Industries du Pétrole ou certaines industries du cycle du combustible nucléaire. Des participations financières extérieures sont à envisager sur des aspects partiels.

6. RELATIONS AVEC D'AUTRES ETUDES

Agressions d'origine externe sur les installations nucléaires : explosions chimiques non confinées dues à un environnement industriel ou aux voies de communication. Fiche 122-2-02.

Etude des transferts atmosphériques. Fiche 123-1-01.

7. DOCUMENTS DE REFERENCE

8. DEGRE DE DISPONIBILITE

TNO-IBBC	CLASSIFICATION: 3.2/3.3/7.1.
TITLE: Responsieberekeningen voor reactorgebouw	COUNTRY: THE NETHERLANDS
	SPONSOR: Ministry of Social Affairs; Ministry of Public works (DIV); TNO-IBBC ORGANIZATION:
TITLE (ENGLISH LANGUAGE): Dynamic response of reactor structures (building and containment).	PROJECTLEADER: Kusters
INITIATED: June 1974 / LAST UPDATING: June 1979	SCIENTISTS: Kusters, de Groot, de Witte, Tolman.
STATUS: Progressing / COMPLETED: 1979	

	CLASSIFICATION: 3.3
TITLE (ORIGINAL LANGUAGE): THEORETICAL STUDIES ON GAS CLOUD EXPLOSIONS A: VAPOURISATION OF HAZARDOUS LIQUIDS	COUNTRY: UK
	SPONSOR: SRD/HSE
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD UKAEA
	PROJECT LEADER: F BRISCOE
INITIATED: OCTOBER 1976 — COMPLETED: SEPTEMBER 1978	SCIENTISTS:
STATUS: COMPLETED — LAST UPDATING: APRIL 1979	

1. GENERAL AIM

To investigate unconfined gas cloud explosions which may present both an external hazard to nuclear installations and a direct hazard to general populations.

2. PARTICULAR OBJECTIVES

To investigate the vapourisation of hazardous liquids after spillage on land or sea; in particular, to predict the pool spread and total vapourisation rate. The models being developed cover instantaneous or continuous spillage onto bounded or unbounded surfaces of both crygenic liquids whose total vapourisation rate is dominated by the boiling process and liquids with higher boiling points whose total vapourisation rate is controlled by surface evaporation.

3. EXPERIMENTAL FACILITIES AND PROGRAMME

None available.

4. PROJECT STATUS

Nearing completion.

5. NEXT STEP

Prepare final documentation.

6. RELATION TO OTHER PROJECTS AND CODES

Theoretical studies on Gas Cloud Explosions B & C, SPILL.

7. REFERENCE DOCUMENTS

See SPILL.

	CLASSIFICATION: 3.3
TITLE (ORIGINAL LANGUAGE): THEORETICAL STUDIES ON GAS CLOUD EXPLOSIONS B: MECHANICAL EFFECTS, PHASE I	COUNTRY: UK
	SPONSOR: SRD
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD UKAEA
	PROJECT LEADER: F BRISCOE
INITIATED: OCTOBER 1975 / COMPLETED: SEPTEMBER 1978	SCIENTISTS:
STATUS: COMPLETED / LAST UPDATING: APRIL 1979	

1. GENERAL AIM

To investigate unconfined gas cloud explosions which may present both an external hazard to nuclear installations and a direct hazard to general populations.

2. PARTICULAR OBJECTIVES

To investigate the characteristics of the incident pressure waves generated by gas cloud explosions; in particular to predict the variation of peak overpressure, positive impulse, wave shape etc. with distance from the explosion centre.

3. EXPERIMENTAL FACILITIES AND PROGRAMME

N/A

4. PROJECT STATUS

N/A

5. NEXT STEPS

Prepare final documentation.

6. RELATION TO OTHER PROJECTS AND CODES

Theoretical Studies on Gas Cloud Explosions A, & C, EEC Sponsored Theoretical Studies of Gas Cloud Explosion Pressure Loadings, BKWAVE, GASEX 1, GASEX 2.

7. REFERENCE DOCUMENTS

See BKWAVE, GASEX 1 and GASEX 2.

	CLASSIFICATION: 3.3
TITLE (ORIGINAL LANGUAGE): THEORETICAL STUDIES OF GAS CLOUD EXPLOSIONS C: MECHANICAL EFFECTS, PHASE II	COUNTRY: UK SPONSOR: SRD/HSE
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD UKAEA PROJECT LEADER: F BRISCOE

INITIATED: APRIL 1978	COMPLETED:	SCIENTISTS:
STATUS: IN PROGRESS	LAST UPDATING: APRIL 1979	

1. GENERAL AIM

To investigate unconfined gas cloud explosions which may present both an external hazard to nuclear installations and a direct hazard to general populations.

2. PARTICULAR OBJECTIVES

To investigate the outstanding problems of gas cloud explosions, specifically:

(i) the mechanisms of flame acceleration from low initial flame speeds, following weak Units up to the high final flame speeds needed to produce severe damage.

(ii) the interaction of the non-sharp pressure waves generated by unconfined gas cloud explosions with structures; this information is required to assist the specification of imposed pressure loadings for use in subsequent calculations of structural response and will of value both in the prediction of hypothetical consequences and the analysis of previous incidents.

(iii) the effects of gas cloud shape on explosion pressure waves, since all present models are spherically-symmetric whereas real clouds will be asymmetric, especially pancake shaped.

3. EXPERIMENTAL FACILITIES AND PROGRAMME

N/A

4. PROJECT STATUS

Just started. No results on (i) and (iii) to date. Preliminary results on (ii) obtained during EEC sponsored study.

5. NEXT STEPS

Prepared detailed programme of work on item (i). Initial studies on items (ii) and (iii) in progress.

6. RELATION TO OTHER PROJECTS AND CODES

Theoretical Studies on Gas Cloud Explosions A & B Sponsored Theoretical Studies of Gas Cloud Explosion Pressure Loadings, BKWAVE, GASEX1, GASEX2, PCAKE.

7. REFERENCE DOCUMENTS

Nuclex 78 Paper summarises results to date on items (ii) and (iii).

	CLASSIFICATION: 3.1 - 3.5
TITLE (ORIGINAL LANGUAGE): Studi di ingegneria del sito	COUNTRY: ITALY SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Studies of site engineering	ORGANISATION: CNEN PROJECT LEADER: Laboratorio Ingegneria Sito
INITIATED: Nov. 1974 (present phase) / COMPLETED:	SCIENTISTS: M. Basili, V. Cagnetti, A. Fontanive
STATUS: In progress / LAST UPDATING: July 1979	

4. POWER TRANSIENTS

146-1 - 01/4151-10		4
TITRE Développement des moyens de calcul nécessaires à l'étude des transitoires anormaux sur les réacteurs PWR.		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) Development of computer codes necessary to study abnormal transient conditions on PWR.		Organisme exécuteur CEA/DSN-SETS Responsable
Date de démarrage 01/01/74	Etat actuel en cours	Scientifiques
Date d'achèvement 31/12/80	Dernière mise à jour 1/79	

1. OBJECTIF GENERAL

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.

2. 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 de représentations mathématiques des phénomènes physiques.
Etude des A.T.W.S.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

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4 - ETAT DE L'ETUDE

1) Avancement à ce jour

- Des modèles ponctuels décrivent le comportement des différents éléments constitutifs du circuit primaire y compris certains systèmes de régulation et de protection.
  Un modèle de générateur de vapeur axial permettant de calculer entre autre l'évolution du niveau d'eau dans le générateur de vapeur et un modèle de pressuriseur fonctionnant indifféremment en simple phase (vapeur ou liquide) et en double phase sont intégrés à la version actuelle.
- Calcul de la répartition des débits dans les lignes vapeur lors d'une rupture de tuyauterie vapeur.
- Calcul des débits de décharge en phase liquide en vapeur aux soupapes du pressuriseur par les méthodes de MOODY, FAUSKE et du modèle homogène équilibré.
- Couplage d'un modèle de générateur de vapeur axial décrivant la réduction de l'échange primaire secondaire lors du dénoyage des épingles.
  Le module utilisé lors de l'étude des A.T.W.S. décrit en particulier la dynamique de l'assèchement du générateur de vapeur en s'appuyant sur un jeu de corrélations d'échange variables tout au long du transitoire.
- Prise en compte d'une forme axiale de flux et calcul de l'effet modérateur à partir de sa densité.
- Calcul de la puissance résiduelle.
- Couplage d'un modèle de pompe permettant à partir de courbes caractéristiques de décrire le comportement de la pompe en circulation naturelle.
- Traitement simplifié de la double phase limité aux faibles taux de vide.
- Couplage d'un module de sauvegarde permettant la reprise des calculs.

2) Résultats essentiels

- Etude de transitoires accidentels de classe 2 et 3 concernant la centrale de Fessenheim

  Le calcul des accidents suivants :

  - Retrait incontrôlé des grappes de réglage en puissance,
  - perte totale de la charge électrique,
  - augmentation excessive de la charge,
  - perte totale de débit primaire,

  révèle un bon accord avec les calculs présentés par le constructeur dans le rapport provisoire de sûreté.

- Etude de transitoires de dimensionnement du circuit primaire

  Dans ce domaine, l'étude de la surpression consécutive à une perte de la charge électrique a été effectuée sur la centrale de Fessenheim

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- Etudes des A.T.W.S. :
  - Etude de la perte de l'eau alimentaire avec déclenchement de la turbine.
    Cette étude a montré que, dans la mesure où l'on adopte des hypothèses communes en ce qui concerne le modèle de dégradation de l'échange de chaleur aux générateurs de vapeur, nos résultats restent comparables à ceux obtenus par EDF et FRAMATOME.
  - Etude de la perte de l'eau alimentaire sans déclenchement de la turbine.
    Ce jeu d'hypothèses conduit à un pic de pression primaire beaucoup plus élevé que le cas précédent.
  - Etude de l'ouverture intempestive d'une soupape du pressuriseur.
    Le code permet entre autre d'étudier le début de l'accident pendant lequel le D.N.B. passe par un minimum.
  - Etude de la perte de l'eau alimentaire menée à partir d'un modèle de G.V. axial.
    Cette étude s'appuyant sur des corrélations d'échange fonction du taux de vide calculé au secondaire du G.V., n'a pas remis en cause les résultats obtenus à partir d'un modèle ponctuel de G.V. Cependant une étude de sensibilité nécessite des développements ultérieurs.
- Etude hors dimensionnement :
  - Etude de la perte totale des alimentations électriques et de la défaillance de l'A.S.G.
    Le code permet d'étudier le comportement du circuit primaire tant que le G.V. n'est pas complètement asséché.

5 - PROCHAINES ETAPES

- Etude d'un modèle de corrélation du niveau d'eau dans les générateurs de vapeur.
- Elaboration d'un modèle coeur tenant compte d'un mélange imparfait de l'eau des différentes branches et d'une répartition dissymétrique des températures dans le coeur.
- Traitement de la double phase dans le plenum supérieur.
- Amélioration du modèle de G.V. axial.
- Traitement explicite de 3 boucles.
- Amélioration informatique.

6 - RELATIONS AVEC D'AUTRES ETUDES

- Etude des transitoires anormaux.
- Etude des A.T.W.S.
- Etude hors dimensionnement.

7 - DOCUMENTS DE REFERENCE

M. TRAN TUC VI
Analyse de fonctionnement d'un PWR : le code "SIRENE"
Note SERMA

JP. MERLE
Etude des transitoires accidentels - centrale de Fessenheim
Note technique SETSSR n° 88, Mai 1976

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Un modèle d'études dynamiques de pressuriseur.
Note SERMA-SPM n° 169 T.

JP. MERLE
Adaptation du code SIRENE à l'étude des transitoires accidentels des centrales à eau pressurisée.
Note SETS n° 69, Mai 1975

JP. MERLE
Régulation de puissance dans le code de fonctionnement pour les PWR
Note SERMA n° 381 T, Janvier 1974

JP. MERLE
Perte de l'eau alimentaire normale sans arrêt d'urgence
Note SETS 203, 1977 .

Perte de l'eau alimentaire sans arrêt d'urgence et sans déclenchement de la turbine.
Note SETS n° 78/210

M. Yann BOARETTO
Etude de sûreté des réacteurs nucléaires à eau ordinaire pressurisée.
Note SETSSR n° 80, Juin 1978

Classification 4.1

Title 1 ANDYCAP: Tre dimensional dynamisk model af kogendevands reaktor kerne.	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 ANDYCAP: 3-D-dynamical model of a BWR-core.	Project leader: P. Skjerk Christensen
Initiated: 1969 Completed: 1972 Status: in use, being improved Last updating: Currently	Scientists: P. Skjerk Christensen

1. General aim

The purpose of the model is to describe and follow transients in a BWR core due to pertubations of process variables in timescale 1-100 seconds.

2. Particular objectives

The project is particularly aimed at normal and abnormal conditions in the reactor. The model is based on a three dimensional nodal description of the core as the neutronic part whereas the hydraulics model consists of a number of parallel one dimensional channels coupled at the lower and upper plenum. A recirculation loop containing a pump is included. In practical calculations the number of nodes has to be limited to some 2000, and the number of hydraulic channels to 30, due to the computer time which on a CDC-6600 is a factor of 100 times the reactor time, strongly depending on the character of the transient. The transients can be initiated by control rod movement, steam load disturbance, feed water disturbance, and main circulation pump disturbance.

3. Experimental facilities

4. Project status

1. Progress to date: A version of the code is in use
2. Essential results:

5. Next steps

Work is in progress directed to speed up the code.

6. Relation with other projects

7. Reference documents

To be issued

8. Degree of availability

Not available

Classification: 4.1

Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: NORHAV - Three-Dimensional Transient Calculation Program for the PWR Core (ANTI)	Organization: Risø National Laboratory
Initiated date: 1977 Completed date: Status: in progress.	Scientists: Anne Margrethe Larsen

1. General aim

Development of a three-dimensional computer program for the calculation of transients in the PWR core.

2. Particular objectives

The computer program should be able to describe PWR transients where the spatial power distribution is important, covering the range from operational transients to design basis accidents (Rod ejection, ATWS). The neutronics part is a three-dimensional nodal theory program, and the hydraulic model is a transient subchannel program which was originally intended for blowdown calculations. The program is planned to deal only with the reactor core, so the boundary conditions at the core inlet and outlet will have to be specified.

3. Experimental facilities

4. Project status

The programming is in progress. The steady state part is running (in the debugging phase).

5. Next steps

Programming of the dynamics. Documentation.

6. Relation with other projects

The nodal theory routines are the same as in the ANDYCAP program (ANDYCAP: 3-D dynamical model of a BWR-core, classification 4.1).

The hydraulic model is the TINA program (NORHAV-P(B)WR blow-down computer program, classification 1.1).

7. Reference documents

8. Degree of availability

Available on exchange basis when completed.

Classification 4.1

Title 1 Kontrolstavs udskydning i en kogendevands-reaktor		COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 Rod ejection transients in BWR		Project leader: P. Skjerk Christensen
Initiated: 1974 Status: in progress	Completed: 1976 Last updating: currently	Scientists: B. Thorlaksen

1. General aim

The aim is to construct a model which describes the reactor transient following the ejection of one of the control rods in a BWR. The model is an extension of the ANDYCAP code which describes the dynamics of a 3-D nodal model of a BWR-core and pressure vessel. The movement of the control rod is calculated assuming critical flow in the guide tube, and the velocity limiter on the rod has been taken into account.

2. Particular objectives

1. The power distribution is calculated in three dimensions, but the local peaking factor has to be found by adequate box calculations.
2. The fuel model includes the possibility of a steam explosion initiated by molten fuel.
3. The consequences of the transient pertinent to the core and pressure vessel are calculated.

3. Experimental facilities

4. Project Status

1. Progress to date: A model for the movement of a control rod after a failure of the housing has been made. Several transients have been analysed

2. Essential results.

5. Next steps

6. Relation with other projects

7. Reference documents

-B. Thorlaksen, Analysis of Control Rod Ejection Accidents in Large Boiling Water Reactors (Risø Report 344).

8. Degree of availability

The thesis is freely available.

Classification 4.1, 4.2, 4.3

Title 1 PWR-stations dynamik model	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 PWR: A PWR power plant dynamics model	Project leader:
Initiated: 1972 Completed: 1974 Status: in use, being improved Last updating: Currently	Scientists: P. la Cour Christensen P. Skjerk Christensen

1. General aim

The goal of the project is to describe and follow transients in a power plant comprising a PWR. The transients may be initiated by any process variable in- or outside the plant.

2. Particular objectives

The plant model must be able to calculate the transients in real time which however limits the number of space meshes. Furthermore, the model must be able to perform interactive calculations which means that the user is able to study immediately the results of his perturbations on the model. At last, the model must be able to serve as a tool used by investigation of control systems.

The model includes a one-dimensional core model and a single cooling loop comprising a circulation pump, a steam generator of the U-tube type, a pressurizer, and a boron injection system. The neutronic model is based on diffusion theory with a single prompt and three delayed neutron groups. The steady state is found by purely digital calculations while the transients are calculated mainly by analogue elements while some neutronic solutions still are calculated by digital techniques.

3. Experimental facilities

4. Project status

1. Progress to date: The models for the steady state and the transients are finished and the two parts have been coupled together. Simple transients have been run.

   A more detailed version with two primary loops, a turbine and a feedwater system has been programmed for simulation by means of a digital simulation system.

2. Essential results:

5. Next steps

6. Relation with other projects

7. Reference documents

   Risö report no. 318.

8. Degree of availability

Classification 4.1, 4.2, 4.3

Title 1 BWR-stations dynamik model	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 Development of a Dynamic Model of a BWR Nuclear Power Plant.	Project leader: P. Skjerk Christensen
Initiated: 1973 Completed: 1976 Status: in use Last updating: currently	Scientists: E. Nonbøl

1. General aim

The purpose of the project is to develop a dynamic model of a nuclear power plant based on a BWR reactor which simulates various transients occurring during normal operating conditions.

2. Particular objectives

The model includes a boiling water reactor, high- and low pressure turbines, moisture separator, reheater, condenser, feedwater heaters and feedwater pump. It is one-dimensional except for the nuclear part of the reactor which is based on point kinetics equations. A great deal of attention has been devoted to the model of the turbine and the feedwater heaters.

3. Experimental facilities

4. Project status

1. Progress to date: The model is finished. Several transients have been run. The kinetic model has been improved by introducing a one-dimensional part.
2. Essential results.

5. Next steps

6. Relation with other project

7. Reference documents

Risø Report No. 335, Risø Report No. 336.

8. Degree of availability

	CLASSIFICATION: 4.1
TITLE (ORIGINAL LANGUAGE): Three-dimensional transient analysis in thermal power reactors: an extensive comparison between finite difference and space-time synthesis method.	COUNTRY: Italy SPONSOR: ENEL
TITLE (ENGLISH LANGUAGE):	ORGANISATION: ENEL - CRTN PROJECT LEADER: F. Di Pasquantonio

INITIATED: January 1977	COMPLETED: -	SCIENTISTS: E. Brega, F. Di Pasquantonio, E. Salina
STATUS: a) completed; b) in progress	LAST UPDATING: July 1979	

The purpose of this work was to discuss some numerical experiments in three-dimensional nuclear reactor dynamics with emphasis on the comparison between the Non Symmetric Alternating Direction Explicit (NSADE) finite difference method, as programmed in the 3DKIN code (a proper exponential transformation of the time dependent equations is also included there), and space-time synthesis methods.

The main conclusion is that for time step sizes which are not small enough, the NSADE solution shows oscillations having no physical meaning. These oscillations are particularly evident when large cores with non-symmetric perturbations are dealt with, and can be made to disappear only at the expense of step sizes that are small beyond any expectation. This sensitivity of the NSADE scheme to time step size is far greater than we imagined before our experiments and it may be an inherent limitation of the method, not only in production work but also in the validation of more approximate methods such as synthesis and quasi-static methods, even though relatively coarse spatial mesh grids are used.

In support of these conclusions, two transient problems, both simulating a control rod drop accident in a BWR core, at a speed of about 3.5 m/sec, are described here: the first is a prompt supercritical transient with a symmetric perturbation (four control rods moving at symmetric locations) and the second one is a delayed supercritical transient with an unsymmetric strongly localized perturbation (only one control rod is dropped). In the latter problem, the power oscillations, caused by the NSADE method, disappear only when the step sizes Δt are so small that about 1,500 time steps sometimes are needed for a mere doubling of the power level ($\Delta t < 0.0005$ sec). On the other hand we found that the same accuracy could be attained by the synthesis calculation with time step sizes varying from 0.003 ÷ 0.01 sec., i.e. ten times as long.

a) It is important to note that our activity in this field can been considered concluded from the point of view of comparison between finite difference and space-time synthesis.

b) However our work continues in the field of three-dimensions transient analysis and presently we are also working by means of nodal methods.

(ENEL-CRTN, Bastioni Porta Volta 10, I-20121 Milano)

45.897

TH-Delft	CLASSIFICATION : 4.1. 4.2.4.3
TITLE : Ontwikkeling van een hybried computermodel voor de simulatie van storingen en ongevallen in een drukwaterreactor.	COUNTRY: THE NETHERLANDS SPONSOR : Ministry of Social Affairs ORGANIZATION : TH-Delft
TITLE (ENGLISH LANGUAGE): Development of a hybrid-computermodel for the simulation of transient and accident conditions in a PWR.	PROJECTLEADER : Latzko
INITIATED :October 1974 LAST UPDATING : May 1978 STATUS :in progress COMPLETED : End 1978	SCIENTISTS : Bruens

General aim

Development of a calculational tool which can compute the plant response to various transients and accident conditions (excl. LOCA) for a PWR. Provide the Possibility to evaluate the effectiveness of control and protection systems under these conditions.

Particular Objectives

Development of a hybrid-computermodel of a PWR. The nuclear core and the steam generator will be the basic modules. These and the other parts of the primary and secundary system will be modelled such that they can be easily adapted to any type of PWR.

Experimental facilities and programma: -

Project Status

The following simulation programs are finished:
- hybrid reactor core model describing the neutron-kinetics and the thermal behaviour
- natural circulation steam generator computer modules describing the thermal/hydraulic behaviour.
- coupling of reactor core model with steam generator model.
- digital model of the pressurizer.

Next steps

- digital models of the turbine, reheater and generator
- digital model of the preheater.
- coupling of the different computer programs.

Relations with other projects: -

Reference documents: -

Degree of availability

through Ministry of Social Affairs.

Budget: -

Personnel: 40 manmonths

Classification 4.1, 4.2, 4.3

Title 1 BWR-stations dynamik model	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 Development of a Dynamic Model of a BWR Nuclear Power Plant.	Project leader: P. Skjerk Christensen
Initiated: 1973 Completed: 1976 Status: in use Last updating: currently	Scientists: E. Nonbøl

Classification 4.1, 4.2, 4.3

Title 1 PWR-stations dynamik model	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 PWR: A PWR power plant dynamics model	Project leader:
Initiated: 1972 Completed: 1974 Status: in use, being improved Last updating: Currently	Scientists: P. la Cour Christensen P. Skjerk Christensen

TH-Delft	CLASSIFICATION : 4.1. 4.2.4.3
TITLE : Ontwikkeling van een hybried computermodel voor de simulatie van storingen en ongevallen in een drukwaterreactor.	COUNTRY: THE NETHERLANDS
	SPONSOR : Ministry of Social Affairs ORGANIZATION : TH-Delft
TITLE (ENGLISH LANGUAGE): Development of a hybrid-computermodel for the simulation of transient and accident conditions in a PWR.	PROJECTLEADER : Latzko
INITIATED :October 1974 — LAST UPDATING : May 1978	SCIENTISTS : Bruens
STATUS :in progress — COMPLETED : End 1978	

Classification 4.1, 4.2, 4.3

Title 1 BWR-stations dynamik model	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 Development of a Dynamic Model of a BWR Nuclear Power Plant.	Project leader: P. Skjerk Christensen
Initiated: 1973 Completed: 1976 Status: in use Last updating: currently	Scientists: E. Nonbøl

Classification 4.1, 4.2, 4.3

Title 1 PWR-stations dynamik model	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 PWR: A PWR power plant dynamics model	Project leader:
Initiated: 1972 Completed: 1974 Status: in use, being improved Last updating: Currently	Scientists: P. la Cour Christensen P. Skjerk Christensen

	CLASSIFICATION: 4.3
TITLE (ORIGINAL LANGUAGE): Instabilità di canali in parallelo durante l'avviamento del reattore CIRENE	COUNTRY: ITALY SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Parallel-channel instability during the start-up of CIRENE reactor	ORGANISATION: CISE PROJECT LEADER: R. Granzini, R. Ravetta
INITIATED: 1977 / COMPLETED: 1979 STATUS: in progress / LAST UPDATING: June 1979	SCIENTISTS: F. Frenquellucci, G. Masini, C. Medich

1. General aim: to determine the parallel-channel instability thresholds of CIRENE reactor during start-up.

2. Particular objectives: to obtain experimental data to validate the instability analysis code in conditions of positive quality and low flowrate at the channel inlet.

3. Experimental facilities and programme: the experiments are carried out on the CIRCE loop simulating in a closed circuit two full-scale power channels in parallel, with the respective two liquid feeders, two steam feeders, and two risers. The start-up conditions of CIRENE reactor will be investigated.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Parallel-channel instability during the start-up of CIRENE reactor	4.3

4. Project status: the experimental loop has been completely set-up. The first set of experiments have been completed, their analysis is in progress.
5. Next steps: Definitions of specifications of finaltests (second set) on the basis of the conclusion of the above mentioned analysis. These tests will be initiated in July '79 and terminated in the same year.
6. Relation with other projects: steady-state and transient tests to study the thermoydraulic behaviour of CIRENE power channels during reactor start-up.
7. Reference documents: none.
8. Degree of availability: to a limited extent.

(CISE, C.P.3986, I-20100 Milano)

TH-Delft	CLASSIFICATION : 4.1. 4.2.4.3
TITLE : Ontwikkeling van een hybried computermodel voor de simulatie van storingen en ongevallen in een drukwaterreactor.	COUNTRY: THE NETHERLANDS SPONSOR : Ministry of Social Affairs ORGANIZATION : TH-Delft
TITLE (ENGLISH LANGUAGE): Development of a hybrid-computermodel for the simulation of transient and accident conditions in a PWR.	PROJECTLEADER : Latzko
INITIATED :October 1974 — LAST UPDATING : May 1978 STATUS :in progress — COMPLETED : End 1978	SCIENTISTS : Bruens

	CLASSIFICATION: 4-8
TITLE (ORIGINAL LANGUAGE): SIMULAZIONE E ANALISI DINAMICA DI IMPIANTI NUCLEARI	COUNTRY: ITALY
	SPONSOR: C.N.E.N.
TITLE (ENGLISH LANGUAGE): SIMULATION AND DYNAMIC ANALYSIS OF NUCLEAR POWER PLANTS	ORGANISATION: C.N.E.N.
	PROJECT LEADER: M.DI BARTOLOMEO
INITIATED: 1962 / COMPLETED:	SCIENTISTS: T.G. BISERNA, F. CIAMPA, O. MODONESI
STATUS: In progress / LAST UPDATING: June 1979	

1. - General aim

Concern the development and the implementation of simulation models of nuclear power plants for dynamic analysis and accident studies.

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. To test control systems.

3. - Experimental facilities and programme

1 Hybrid computer - EAI - 8945
1 Hybrid computer - EAI - PACER - 700

4. - Project status

Hybrid models of particular components of BWR and PWR reactors have been implemented to verify their dynamic behaviour.

5. - Reference documents

1) P.Giordano - A.Mathis - G.Melucci
Dynamics and control studies for a steam-generating pressure-tube reactor
Doc. CNEN RT/ING(65)13 - Sept. 1965.

2) P.Giordano - A.Mathis - B.Rimini
Analog methods for studying the space-time dynamics of nuclear reactors.
Proceedings of 5th Congress of the Int.Ass.for An.Comp. Losanna 1967.

TITLE (ENGLISH LANGUAGE): SIMULATION AND DYNAMIC ANALYSIS OF NUCLEAR POWER PLANTS	CLASSIFICATION: 4-8

3) A.Mathis
The use of hybrid computers in the Italian CNEN nuclear program. Conference on "The Effective Use of Computers in the Nuclear Industry".
Knoxville, Tenn. (USA) April 21-23, 1969

4) P.Giordano - A.Mathis - O.Modonesi
Use of analog and hybrid computers in the design of CIRENE type nuclear power plant.
Enlarged Halden Programme Group Meeting on Computer Control.
LOEN (Norway) - May 29th - June 2nd, 1972

5) T.G.Biserna - M.Di Bartolomeo
Un modello ibrido per l'analisi di incidenti di un impianto nucleare
Rivista di Informatica - numero speciale - novembre 1972.

6. - Degree of availability

Besides the equipments mentioned there is also a know-how in dynamic models implementations.

5. BEHAVIOUR, TRANSPORT AND RELEASE OF RADIOACTIVE SUBSTANCES

Classification 5

Title 1	
Dosisbelastninger i A-kraftværker	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 Radiation Doses in Nuclear Power Plants	Project leader: Kurt Lauridsen
Initiated: February 1974 Completed: Status: Progressing Last updating:	Scientists: Kurt Lauridsen

1. General Aim. To study the radiation doses received by power plant personnel and the factors influencing the size of these doses.

2. Particular objectives. Development of a mathematical model for the transport of radioactive material in the coolant circuit of a BWR-plant, and calculation of the radiation fields outside the components of the coolant circuit.

3. Experimental facilities None

4. Project status

4.1. Progress to date. Three computer codes have been developed, FICOPI, INAPI, and SHIELD, all in FORTRAN.

FICOPI calculates the inventories of radioactive fission and corrosion products in the components of the coolant circuit as a function of power history.

INAPI calculates the inventories of radioactive nuclides created by activation of the coolant itself.

SHIELD is a simple shielding code based on point-kernel technique.

4.2. Essential results

5. Next steps. Further testing of the inventory calculations still has to be done in order to verify the models. A more sophisticated shielding code is being considered.

6. Relation with other projects. No formal relations to other projects are established, but interfaces exist with studies performed at Risø on the subjects: Systems delineation and power plant operation.

7. Reference documents Kurt Lauridsen, Development of a Model for the Assessment of Radiation Fields Around Nuclear Power Plant Components, Risø Report No. 353 (1977), 106 pp.

8. Availability. The project information is freely available.

	CLASSIFICATION: 5
TITLE (ORIGINAL LANGUAGE): Sampling, monitoring and analysis of PWR circuits (C 7.1)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Winfrith and Harwell)
	PROJECT LEADER: D J Ferrett (Winfrith) J B Sayers (Harwell)
INITIATED: March 1978	COMPLETED:	SCIENTISTS:
STATUS: In progress	LAST UPDATING: March 1979	

Background

There is considerable experience at Winfrith with sampling systems, monitoring and analytical methods from HP water circuits for reactors. It is necessary to apply this to PWR problems by liaison with plant operators and trials at Power Stations and to assess the performance of sampling systems by autoclave and loop studies.

Objectives

1. To assess PWR sampling efficiency and associated effects from autoclave and theoretical studies at Harwell.

2. To apply this work to loop studies in the DIDO Water Loop at Harwell.

3. To liaise with PWR operators regarding sampling, monitoring and analytical problems relevant to the chemistry and man-rem problems.

4. To develop appropriate analytical techniques to support other programmes when required, eg techniques for monitoring decontamination procedures, etc.

Programme

Autoclave studies in laboratories at Harwell. Experiments in DIDO Water Loop at Harwell. Analytical development work at Winfrith.

Facilities

Autoclave and DIDO Water Loop at Harwell.
Equipment developed on WSGHWR for power plant applications.
Laboratories at Winfrith for analytical development work.

Reference Documents

Classification 5/14/16/18

Title 1	Prediction of Reactor Releases	Country	UK
		Sponsor	CEGB
Title 2	Building Entrainment Effects	Organisation	Research Divn. Berkeley Nuclear Laboratories
Initiated	1974 Completed	Project Leaders	Dr. H. F. Macdonald Dr. B. M. Wheatley
Status	Continuing Last updating		

1. General Aim

Improvement of models used to predict the environmental consequences of routine operational and accidental releases of radioactivity from reactor systems of interest to the CEGB (GCR & LWR).

2. Specific Objectives

a) To update earlier fuel inventory, atmospheric dispersion and individual and collective dose models. The improved models, together with a simple food chain model, will form the modules of an improved environmental code NECTAR (Nuclear Environmental Consequences, Transport of Activity and Risk). This will be used to study reactor releases and irradiated fuel handling and waste management problems. The results of this work will provide a basis for assessing both the risks associated with power reactor operation, in relation to other risks of modern life, and the implications of alternative frequency/release limit criteria used in reactor siting.

b) To devise improved methods for predicting the effects of building entrainment on low-level reactor discharges. An experimental programme to determine the concentration patterns of low-level atmospheric discharges in the neighbourhood of buildings will be commenced. Full-scale data and wind tunnel model results will be used to investigate the feasibility of developing generalised building entrainment models.

3. Facilities and Overall Programme

This work forms part of continuing studies on present and future reactor systems of interest to the CEGB which have resulted in the development of a range of computer programs describing the build-up of activity in reactor fuels, the environmental consequences of reactor releases, and the fate of inhaled and ingested radioactivity. These programs are available for general use on the Board's IBM computer based in London and are widely used throughout the UK and abroad.

4. Project Status

Earlier studies on accidental releases from GCR have been extended to include severe hypothetical LMFBR accidents. As part of the development work for NECTAR, an improved fuel inventory code RICE has been developed, while comparisons of various Gaussian and diffusivity models for the prediction of atmospheric dispersion have been carried out. A review paper describing methods of prediction of accidental releases from nuclear installations has been published.

/5.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	5.1	RS 285

Vorhaben/Project Title	
Untersuchung der Jod-Exhalation aus UO_2 unter stationären und transienten Bedingungen Investigation of the Iodine Release from UO_2 Under Steady-state and Transient Conditions	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik B 222, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 12. 77	31. 8. 80	Dr. Peehs
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 78	

1. General Aim

In LWR's, the amount of fission-Cesium generated during burn-up is approx. 10 times higher than that of fission Iodine. Since Cs and Iodine form a more stable Iodide than the comparable Zirconium-Iodides, it is necessary to consider the related chemical equilibria based on the actual Cs/I-release to evaluate the effect of Iodine on the loss of ductility of Zr-cladding.

After experimental data on the Iodine and Cesium release from fuel are obtained, it will be possible to determine the effect of Iodine-caused loss of ductility on the fuel rod performance during transient and especially LOCA conditions. This will be performed by using the computed Cs/I-inventory of the fuel, the measured release characteristics and the results from ISCC of Zry.

2. Particular Objectives

The fission Cesium and Iodine integrated in the UO_2 lattice can only migrate to a free surface (outer and inner pellet surface and/or fracture) by diffusion.

It is the objectives of this program to investigate on the I/Cs release characteristics under the following conditions:

- I/Cs release under steady-state conditions (isochronous-isothermal heat-up) enclosed table. They are exclusively fuel rods from the ramp test program with varying burn-ups. Detailed pre-irradiation characterizations are available of the fuel as well as detailed post-irradiation analyses for all microsections listed in the table 1 below.

7. Next Steps

To 3.2 The installed test equipment will be subjected to another cold test. In that connection also calibration tests will be carried out with inactive substances. After completion of the test phase the installed test equipment is to be taken to the hot laboratory.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Table 1

Archives No.	Microsection No.	Burn-up
BE 317 KWO V	Microsection 44.317.0601 MK 02 Microsection 44.317.0601 MK 01	∿ 5.0 GWd/tU ∿ 10.0 GWd/tU
BE 317 KWO VI	Microsection 48.317.0602 half completed	∿ 24.0 GWd/tU
BE 317 KWO VII	Microsection 53.317.0609 MK 01	∿ 32.0 GWd/tU
BE 280 KWO VI	Microsection 48.280.2804 MK 01	∿ 29.0 GWd/tU
BE 250 KWO VII	Microsection 48.250.0201 MK 01 Microsection 48.250.0601 MK 01	∿ 32.0 GWd/tU ∿ 32.0 GWd/tU

149-1-03 /4141 20 5-1

TITRE Expérience de fonctionnement PWR : Synthèse des campagnes de mesures concernant les transferts de radioactivité.	Pays France Organisme Directeur CEA/DSN
TITLE (Anglais) Operating experience in PWR	Organisme exécuteur CEA/DSN/SESRS Responsable DSN/SESRS/FAR

Date de démarrage 1977	Etat actuel en cours	Scientifiques
Date d'achèvement 1982	Dernière mise à jour Décembre 1978	

1. OBJECTIF GENERAL

Le but essentiel de cette étude est une meilleure connaissance de l'origine, la nature, l'importance et la répartition de l'activité dans les circuits primaires des réacteurs et dans les enceintes qui les entourent. La détermination des mécanismes de production et de transferts des isotopes radioactifs dans les différents milieux se fait par mesure in situ. Les résultats doivent eux-mêmes être utilisés pour qualifier des codes de calcul représentatifs des phénomènes.

2. OBJECTIFS PARTICULIERS

Détermination des sources radioactives en fonctionnement normal ou en cas d'accident qui sont à l'origine :

- de la contamination des locaux et enceintes
- des rejets gazeux vers l'environnement
- des effluents liquides destinés à être traités
- des rejets liquides après traitement
- **de la production de déchets solides.**

Toutes ces informations constituent un support indispensable à l'analyse de sûreté. L'interprétation des résultats devrait permettre en particulier d'évaluer :

- le débit de fuite primaire dans le bâtiment réacteur
- le débit de fuite primaire-secondaire
- les performances des circuits d'épuration

.../

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Néant.

4. ETAT DE L'ETUDE

1) Avancement à ce jour

On dispose des résultats de plusieurs campagnes de mesures effectuées à Fessenheim I et à Tihange I par le DSN/SESTR et le DRE/SEN. EDF a par ailleurs établi un bilan tritium pour les 2 premiers cycles de fonctionnement de Tihange I et pour le début du 1er cycle de fonctionnement des 2 tranches de Fessenheim.

2) Résultats essentiels

a) Bilan tritium : d'après le bilan tritium établi par EDF/SEPTEN sur les 2 premiers cycles de Tihange I et en début de cycle sur les 2 tranches de Fessenheim, le taux de diffusion du tritium à travers les gaines est d'environ 1 %.

b) Contamination du circuit primaire par les produits de corrosion

L'activité des produits de corrosion croît lors des arrêts à froid, on observe notamment une montée du Co 58 à 1 ou 2 Ci/t à Tihange I et à Fessenheim. Des mesures effectuées à Fessenheim I par EDF/ER et le DRE/SEN ont montré que le circuit RCV est une source de produits de corrosion (confirmation des prévision du code PACTOLE). Les mesures effectuées lors du 1er arrêt pour rechargement du réacteur Tihange I montrent qu'on a un assez bon accord calcul PACTOLE-expérience, en ce qui concerne les activités déposées.

c) Contamination du circuit primaire par les produits de fission :

- Tihange I : le réacteur a fonctionné avec des ruptures de gaines au cours des 2 premiers cycles. Le DRE/SEN (à l'aide du code PROFIP-3) et EDF/SEPTEN ont élaboré chacun une méthode de détermination du taux de rupture de gaine à partir des mesures d'activité des produits de fission dans l'eau primaire. Les résultats sont très encourageants et le DRE/SEN a en particulier prévu le nombre de ruptures de gaine à la fin du 2ème cycle. Le 3ème cycle a débuté sans rupture de gaine.

- Fessenheim : la tranche I a fonctionné sans rupture de gaine jusqu'en août 1978 ; durant cette période l'activité des produits de fission dans l'eau primaire est due uniquement à la contamination superficielle des gaines estimée à 0,3 g d'uranium enrichi à 3,1 % (calcul PROFIP). Une campagne de mesures effectuée en août 1978 indique, qu'à cette date, le combustible de la tranche 1 présente probablement un ou plusieurs défauts de tailles très réduites.
L'activité des produits de fission mesurée sur la tranche 2 est due uniquement à la contamination des gaines évaluée à 0,2 g d'uranium enrichi à 3,1 % (calcul PROFIP).

.../

d) Transfert de contamination au-delà de la 2ème barrière :

Les campagnes de mesures effectuées par le DSN/SESTR à Tihange et à Fessenheim montrent que la contamination de l'atmosphère des enceintes est très faible. Les mesures effectuées dans le bâtiment réacteur de Fessenheim I ont mis en évidence la présence de Xe 133, Xe 135, I 131 et de tritium qui se trouve principalement sous forme tritium gaz.

5. PROCHAINES ETAPES

a) Bilan tritium

Le bilan tritium sera poursuivi sur Tihange I (3ème cycle) afin de déterminer l'influence éventuelle du taux de combustion sur le taux de sortie du tritium ; il en sera de même à Fessenheim 1 et 2.

b) Contamination du circuit primaire par les produits de corrosion

- Essai d'injection d'eau oxygénée au cours d'un arrêt à froid à Fessenheim pour solubiliser le Co 58 et l'éliminer par épuration du fluide primaire
- Détermination quantitative de l'apport des produits de corrosion à partir du circuit RCV
- Bilan de l'activité des dépôts de produits de corrosion sur les parois hors flux (mesures prévues à chaque arrêt pour rechargement)
- L'ensemble des résultats expérimentaux seront confrontés aux calculs prévisionnels du code PACTOLE

c) Contamination du circuit primaire par les produits de fission

- Poursuite des mesures d'activité des produits de fission dans l'eau primaire à Tihange et Fessenheim en vue de qualifier le code de calcul PROFIP-3
- Qualification et comparaison des méthodes de détermination du taux de rupture de gaine

d) Transfert de contamination au-delà de la 2ème barrière

- Poursuite des mesures de contamination de l'atmosphère des bâtiments réacteur et combustible en multipliant les points de mesure afin de vérifier si la contamination est répartie de manière homogène dans les enceintes. Ces mesures permettront, lorsque le niveau de contamination sera suffisamment élevé, de déterminer le débit de fuite primaire dans le bâtiment réacteur
- Evaluation du débit de fuite primaire-secondaire à partir des mesures d'activité du Cs 137
- Détermination de l'efficacité des chaînes de traitement du TEP et du TEU
- Détermination du taux de fuite du système TEG

6. RELATIONS AVEC D'AUTRES ETUDES

L'étude de fonctionnement des réacteurs PWR est directement liée aux programmes expérimentaux sur réacteurs et sur boucles.

.../

- Fiche DSN/SESTR 149-1-06 et 159-1-03 Transfert de la contamination dans les réacteurs en service
- Fiche DRE/SEN 149-1-07 Contamination des circuits primaires : suivi de Fessenheim
- Fiche 149-1-01 Synthèse des programmes d'étude sur la contamination des circuits primaires des PWR
- Fiche 140-1-02 Evaluation du taux de relâchement des produits de fission à partir d'une rupture de gaine dans un réacteur à eau ordinaire en fonction de la puissance linéaire (Expériences EDITH)
- Fiche 140-1-03 Evaluation du taux de relâchement des produits de fission à partir d'une rupture de gaine dans un réacteur à eau ordinaire sous l'influence des cyclages de puissance (expériences CYFON)
- Fiche 140-1-04 Evaluation du taux de relâchement des produits de fission hors d'une rupture de gaine déclenchée en fonctionnement normal dans un réacteur à eau sous pression (expériences CRUSIFON).

7. DOCUMENTS DE REFERENCE

- Comptes rendus des réunions du Groupe de Travail "Transfert de contamination à Fessenheim"
- J-J. SEVEON - Etat d'avancement des études relatives au transfert de la contamination dans les réacteurs à eau pressurisée - Note technique SESRS N° 27 - Octobre 1978

8. DEGRE DE DISPONIBILITE

Diffusion restreinte.

140.1.02		5-1
TITRE EDITH - Evaluation du taux de relâchement des produits de fission à partir d'une rupture de gaine dans un réacteur à eau ordinaire en fonction de la puissance linéaire.		**Pays** FRANCE **Organisme Directeur** CEA/DgCS et EdF
TITLE (Anglais) EDITH - Assessment of the fission products release rate from a cladding defect in a pressurized water reactor in terms of the linear heat rating.		**Organisme exécuteur** CEA/DMECN/DMG **Responsable**
Date de démarrage 1/1/76	**Etat actuel** EN COURS ACTION PROLONGEE	**Scientifiques**
Date d'achèvement 31/12/79	**Dernière mise à jour** MISE A JOUR N° 1 DU 3.1.79	

1. OBJECTIF GENERAL :

Etablir une relation entre le taux de dégagement des produits de fission hors d'un crayon présentant une rupture de gaine de section donnée et située au droit d'un bouchon et la puissance linéique de ce crayon en fonctionnement normal, en début de vie et en régime stable (10 à 40 $KW.m^{-1}$).

2. OBJECTIFS PARTICULIERS :

1. Pour un état donné de la charge de combustible d'un réacteur, déterminer et quantifier les paramètres permettant d'évaluer le terme source à prévoir dans l'hypothèse d'un fonctionnement en base, et notamment l'influence de la puissance linéique.

2. Eventuellement, apporter une contribution à l'étude de la dégradation d'un crayon présentant une rupture de gaine dans les mêmes conditions de fonctionnement.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME :

Jusqu'à présent, ces essais ont été réalisés dans le dispositif BOUFFON installé dans le réacteur SILOE à Grenoble. Il s'agit d'un bouilleur à thermosiphon en circuit fermé comportant un système de prélèvement pour effectuer hors pile les mesures de contamination.

Les essais prévus en 1979 seront, en principe, effectués dans le dispositif BOUFFON-JET. Ce dernier, installé dans le même réacteur, est également à thermosyphon, mais il est, de plus, accéléré par jet de vapeur. Les conditions thermohydrauliques y sont plus représentatives de celles d'un réacteur de puissance.

4. ETAT DE L'ETUDE :

1. Avancement à ce jour :

La première partie du programme, comportant deux expériences, J51 et EDITH 1, est terminée.

2. Résultats essentiels :

1/ Le dégagement des produits de fission hors du crayon est d'autant plus important que la puissance est élevée. Cependant, il paraît devenir indépendant de la puissance linéique au-delà de 20 $KW.m^{-1}$.

2/ Le taux de dégagement des halogènes est environ dix fois moins élevé que celui des gaz rares.

3/ A basse puissance linéique (20 $KW.m^{-1}$), le comportement des halogènes et des gaz n'est pas le même qu'à puissance linéique plus élevée. Les modes de transport hors du crayon sont probablement différents.

5. PROCHAINES ETAPES :

Ces résultats ont été jugés suffisamment importants pour qu'une prolongation du programme soit décidée. Deux nouvelles irradiations, EDITH 2 et EDITH 4 seront donc réalisées en 1979 et auront pour objectif d'étudier l'influence de l'emplacement de la rupture et du taux de combustion.

6. RELATION AVEC D'AUTRES ETUDES :

Cette étude doit être utilisée pour l'analyse de sûreté des réacteurs à eau ordinaire sous pression utilisés en base (évaluation du terme source de contamination du circuit primaire). A ce titre, elle est en relation avec les études du DRE sur la mise au point du code PROFIP, ainsi qu'avec les activités du Groupe Mixte EdF-CEA de suivi de la contamination dans la centrale de FESSENHEIM. De plus, elle est connexe aux programmes CYFON et CRUSIFON, la synthèse de l'ensemble de ces trois programmes étant faite dans le cadre de la fiche 140.1.05 (modélisation).

7. DOCUMENTS DE REFERENCE :

Compte rendu DMG 54/77 du 9 juin 1977 : examens métallographiques du crayon BOUFFON J51.
Compte rendu DMG 98/78 du 11 septembre 1978 : contamination du circuit primaire d'un réacteur à eau sous pression par des crayons combustibles présentant des défauts de fabrication.
Compte rendu DMG 108/78 du 2 octobre 1978 : évolution du programme expérimental concernant le comportement des crayons combustibles défectueux dans les réacteurs à eau.
Compte rendu DMG 144/78 du 14 décembre 1978 : fission product release from a PWR defected fuel rod. Effect of thermal cycling.

8. DEGRE DE DISPONIBILITE DES DOCUMENTS :

Documents internes. Toutefois, les comptes rendus DMG 98/78 et 114/78 ont fait l'objet de publications aux colloques de l'AIEA à Prague et de l'ANS/ENS à Bruxelles.

140.1.03		5-1
TITRE CYFON : EVALUATION DU TAUX DE RELACHEMENT DES PRODUITS DE FISSION A PARTIR D'UNE RUPTURE DE GAINE DANS UN REACTEUR A EAU ORDINAIRE SOUS L'INFLUENCE DES CYCLAGES DE PUISSANCE.		Pays FRANCE Organisme Directeur CEA/DgCS et EdF
TITLE (Anglais) CYFON : ASSESSMENT OF THE FISISON PRODUCTS RELEASE RATE FROM A CLADDING DEFECT IN A PRESSURIZED WATER REACTOR UNDER POWER CYCLING OPERATION.		Organisme exécuteur CEA/DMECN/DMG Responsable
Date de démarrage 1/1/76	Etat actuel ACTION SUSPENDUE	Scientifiques
Date d'achèvement 31/12/79	Dernière mise à jour MISE À JOUR N° 1 AU 2/1/79	

1. OBJECTIF GENERAL :

Déterminer le taux de dégagement des produits de fission hors d'un crayon présentant une rupture de gaine dans des conditions de cyclage représentatives du suivi de réseau (cyclage de type 15/7).

2. OBJECTIFS PARTICULIERS :

1/ Pour un état donné de la charge du combustible d'un réacteur, déterminer des paramètres permettant d'évaluer le terme source à prévoir dans l'hypothèse d'un fonctionnement en suivi de réseau.

2/ Eventuellement, apporter une contribution à l'étude de la dégradation d'un crayon présentant une rupture de gaine dans les mêmes conditions de fonctionnement.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME :

Ces essais ont été réalisés dans le dispositif BOUFFON installé dans le réacteur SILOE à Grenoble. Il s'agit d'un bouilleur à thermosyphon en circuit fermé comportant un système de prélèvement pour effectuer, hors pile, les mesures de contamination.

4. ETAT DE L'ETUDE :

1. Avancement à ce jour :

Au cours de l'année 1978, une troisième irradiation, CYFON 3, a été réalisée.

2. Résultats essentiels : (cf. fiche 140.1.05)

1/ Les cyclages périodiques du type 15/7 provoquent des bouffées de produits de fission lors des variations de puissance ayant pour conséquence des pics d'activité dans le circuit primaire. Ces bouffées se produisent à la baisse de puissance pour des cyclages entre 20 et 40 $KW.m^{-1}$ et à la montée pour des cyclages plus profonds (12 à 40 $KW.m^{-1}$).

2/ Les bouffées principales contiennent des gaz rares et des halogènes. Les bouffées secondaires surtout des gaz rares.

3/ Au cours de l'expérience CYFON 3, une rupture secondaire très importante s'est produite, avec perte de combustible dans le caloporteur.

5. PROCHAINES ETAPES :

L'action étant suspendue, il n'est pas prévu d'essai en 1979. Cependant, la reprise de ces essais pourrait être envisagée dans la boucle JET sur des crayons présentant une rupture au droit du combustible.

6. RELATION AVEC D'AUTRES ETUDES :

Cette étude doit être utilisée pour l'analyse de sûreté des réacteurs à eau ordinaire sous pression utilisés en suivi de charge (évaluation du terme source de contamination du circuit primaire). A ce titre, elle est en relation avec les études du DRE sur la mise au point du code PROFIP, ainsi qu'avec les activités du Groupe mixte EdF-CEA de suivi de la contamination dans la centrale de FESSENHEIM. De plus, elle est connexe aux programmes EDITH et CRUSIFON, la synthèse de l'ensemble de ces trois programmes étant faite dans le cadre de la fiche 140.1.05 (modélisation).

7. DOCUMENTS DE REFERENCE :

Compte rendu DMG 103/77 : émission de produits de fission par un crayon présentant une fuite de conductance élevée en partie haute et soumis à cyclage de puissance. Expérience CYFON 1, du 30 novembre 1977.
Compte rendu DMG 104/77 : examens après irradiation de CYFON 1, du 30 novembre 1977.
Compte rendu DMG 19/78 : émission de produits de fission par un crayon présentant une fuite de conductance élevée en partie haute et soumis à cyclage de puissance. Expérience CYFON 2, du 8 février 1978.
Compte rendu DMG 99/78 : analyse de l'émission des produits de fission par un crayon présentant un défaut d'étanchéité. Effet du cyclage thermique du 12 septembre 1978.

8. DEGRE DE DISPONIBILITE DES DOCUMENTS :

Notes internes, à l'exception de la quatrième, communiquée à la Conférence Internationale de la Sûreté des Réacteurs Nucléaires de Puissance, 16-19 octobre 1978, Bruxelles.

140.1.04		5-1
TITRE CRUSIFON : EVALUATION DU TAUX DE RELACHEMENT DES PRODUITS DE FISSION HORS D'UNE RUPTURE DE GAINE DECLENCHEE EN FONCTIONNEMENT NORMAL DANS UN REACTEUR A EAU SOUS PRESSION.		Pays FRANCE **Organisme Directeur** CEA/DgCS et EDF
TITLE (Anglais) CRUSIFON : ASSESSMENT OF THE FISSION PRODUCTS RELEASE RATE FROM A CLADDING DEFECT TRIGGERED UNDER NORMAL OPERATION IN A PRESSURIZED WATER REACTOR.		Organisme exécuteur CEA/DMECN/DMG Responsable
Date de démarrage 1/1/76	Etat actuel EN COURS	Scientifiques
Date d'achèvement 31/12/80	Dernière mise à jour MISE A JOUR N° 1 AU 3.1.79	

1. OBJECTIF GENERAL :

Déterminer le taux de dégagement des produits de fission hors de crayons préirradiés à différents taux de combustion représentatifs du début de vie (2000 MWJ/T) et de la fin de vie (au-delà de 20.000 MWJ/T) subissant une rupture de gaine en fonctionnement normal.

2. OBJECTIFS PARTICULIERS :

1/ Déterminer l'influence du taux de combustion.
2/ Déterminer l'influence du niveau de puissance linéique en palier avant l'essai (25 et 40 $KW.m^{-1}$).
3/ Déterminer l'influence de paramètres géométriques (jeu résiduel, longueur du crayon).

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME :

Les premiers de ces essais ont été réalisés dans le dispositif BOUFFON installé dans le réacteur SILOE à Grenoble. Il s'agit d'un bouilleur à thermosyphon en circuit fermé comportant un système de prélèvement pour effectuer, hors pile, les mesures de contamination.

Le programme de 1979 pourrait être réalisé, au moins en partie dans le dispositif BOUFFON-JET. Ce dernier, installé dans le même réacteur, est également à thermosyphon, mais il est, de plus, accéléré par jet de vapeur. Les conditions thermohydrauliques y sont plus représentatives de celles d'un réacteur de puissance.

4. ETAT DE L'ETUDE :

1. Avancement à ce jour :

Au cours de l'année 1978, un essai additionnel, CRUSIFON 1bis, doublant l'essai CRUSIFON 1 a été réalisé. Deux autres essais sont en cours d'irradiation : CRUSIFON 3 et 4.

2. Résultats essentiels :

L'apparition de la rupture s'accompagne d'une forte bouffée de produits de fission, suivie d'une décroissance progressive. A chaque baisse de puissance ultérieure, le même phénomène se reproduit. Le pic d'activité initial est interprété comme la vidange, dans le circuit primaire, des gaz de fission contenus dans le volume libre.

5. PROCHAINES ETAPES :

En 1979, il est prévu deux nouveaux essais, CRUSIFON 2bis, doublant CRUSIFON 2 et CRUSIFON 5.

6. RELATION AVEC D'AUTRES ETUDES :

Cette étude doit être utilisée pour l'analyse de sûreté des réacteurs à eau sous pression (évaluation du terme source de contamination du circuit primaire). A ce titre, elle est en relation avec les études du DRE sur la mise au point du code PROFIP, ainsi qu'avec les activités du Groupe Mixte EDF-CEA de suivi de la contamination dans la Centrale de FESSENHEIM. De plus, elle est connexe aux programmes CYFON et EDITH, la synthèse de l'ensemble de ces trois programmes étant faite dans le cadre de la fiche 140.1.05 (modélisation).

7. DOCUMENTS DE REFERENCE :

Compte rendu DMG 69/77 : programme concerté CEA-EDF sur le comportement des éléments combustibles défectueux ou rompus en régime normal et en régime transitoire. Expériences CRUSIFON, du 8 août 1977.
Compte rendu DMG 73/78 : examens après irradiation de CRUSIFON 2, du 26 juin 1978.

8. DEGRE DE DISPONIBILITE DES DOCUMENTS :

Notes internes non diffusables sous leur forme actuelle.

140.1.05		5-1
TITRE MODELISATION DES RESULTATS DES PROGRAMMES EDITH, CYFON ET CRUSIFON EN VUE DE LEUR UTILISATION DANS DES CODES DE CONTAMINATION DU CIRCUIT PRIMAIRE.		**Pays** FRANCE **Organisme Directeur** CEA/DgCS et EDF
TITLE (Anglais) MODELLING OF THE FINDINGS FROM PROGRAMS EDITH, CYFON AND CRUSIFON FOR USE IN PRIMARY CIRCUIT CONTAMINATION CODES.		**Organisme exécuteur** CEA/DMECN/DMG **Responsable**
Date de démarrage 1/1/78	**Etat actuel** EN COURS	**Scientifiques**
Date d'achèvement 31.12.80	**Dernière mise à jour** 21/12/78 (nouvelle fiche)	

1. OBJECTIF GENERAL :

Modéliser les résultats expérimentaux obtenus à la suite des essais des programmes EDITH, CYFON et CRU IFON, en vue de leur utilisation comme module de terme source dans les codes d'évaluation de la part de contamination du circuit primaire imputable aux produits de fission, le code PROFIP et les codes EDF notamment. Proposer une réglementation propre à maintenir les doses reçues par le personnel d'exploitation ainsi que les rejets dans des limites acceptables.

2. OBJECTIFS PARTICULIERS :

1/ Modéliser les rejets dans le circuit primaire à partir de crayons présentant une rupture de gaine en fonctionnement à puissance constante, mais en incluant les démarrages et les arrêts ;

2/ Modéliser les rejets en exploitation en suivi de charge.

3/ Réaliser des essais supplémentaires pour préciser le profil thermique des crayons rompus (EDITH 3, FURET 28, FURET 29).

4/ Valider les modèles proposés pour des crayons de longueur normale.

5/ Proposer une mise en forme des résultats utilisable en vue de la rédaction d'un texte réglementaire.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME :

Sans objet pour la partie de l'action relative à la modélisation proprement dite.

4. ETAT DE L'ETUDE :

1. Avancement à ce jour :

Une première analyse des résultats des programmes EDITH et CYFON a été faite.

2. Résultats essentiels :

En fonctionnement permanent, le taux d'activité relâché croît avec la puissance linéique du crayon jusqu'à 20 $KW.m^{-1}$. Au-delà, il ne paraît pas relié de façon sensible à la puissance.

Les cyclages de puissance provoquent un très fort accroissement de la fraction relâchée. Des pics d'activité apparaissent au moment des variations de puissance. L'importance et la position de ces pics dépendent des valeurs des paliers hauts et bas. Le taux de relâchement paraît déterminé par l'équilibre entre l'eau et la vapeur dans le jeu entre le combustible et la gaine.

5. PROCHAINES ETAPES :

1/ Préciser le profil thermique des crayons rompus.
2/ Fournir une modélisation chiffrée des phénomènes observés.
3/ Transposer cette modélisation à des crayons de longueur standard, dans des conditions d'exploitation représentatives d'un réacteur de puissance.

6. RELATION AVEC D'AUTRES ETUDES :

En dehors des programmes EDITH, CYFON et CRUSIFON déjà nommés, cette action est en relation avec les études du DRE sur la mise au point du code PROFIP, ainsi qu'avec les activités du Groupe mixte EDF-CEA de suivi de la contamination dans la Centrale de FESSENHEIM.

7. DOCUMENTS DE REFERENCE :

Aucun document n'a encore été publié à ce jour.

8. DEGRE DE DISPONIBILITE DES DOCUMENTS :

Sans objet.

	CLASSIFICATION: 5.1
TITLE (ORIGINAL LANGUAGE): Analisi di prodotti di fissione emettitori γ a vita breve presenti nel fluido primario di un reattore LWR	COUNTRY: ITALY SPONSOR: ENEL
TITLE (ENGLISH LANGUAGE): Analysis of short-lived γ-emitter fission products in the main coolant of Light Water Reactors	ORGANISATION: Politecnico di Milano (*) PROJECT LEADER: G.Sandrelli -(ENEL) M.Mandelli-Bettoni (Politecnico)

INITIATED:	COMPLETED:	SCIENTISTS:
1974	1981	A.Foglio Para - Politecnico M.Mandelli-Bettoni " G.Sandrelli - ENEL
STATUS: In progress	LAST UPDATING: July 1979	

(*) Istituto di Ingegneria Nucleare - CESNEF

1. General aim

To improve the most common method used in a nuclear power plant LWR, operating in normal conditions, for the detection of failed fuel elements, based on the measurement of short-lived γ-emitter fission products in the main coolant and in the off-gases.

2. Particular objectives

2.1 To select in the γ-spectra emitted by samples of main coolant and off-gases a number of lines certainly free from interferences due either to long-lived fission products or to activation products.

2.2 To check the relialibity of available techniques for automatic γ-spectrum unfolding.

2.3 To reanalyze thoroughly the theory of fission product release from fuel elements, adapting it to the complex operating conditions of a power plant.

3. Experimental facilities

All measurements were performed at ENEL nuclear power plant.

4. Project status and essential results

The points 2.1 and 2.2 have been completed. The results of the study carried out according to the point 2.3 have been successfully applied to the analysis of the radioiodines and radioactive fission gases in the main coolant at Trino Vercellese power plant, operating at full load and in stretch-out conditions. Also the agreement between theory and measurements at Garigliano plant turned out to be fully satisfactory.

5. Next steps

The same procedure will be applied to the analysis of fission products in the main coolant and in the off-gases at Caorso power plant.

45.897 ./.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Analysis of short-lived γ -emitter fission products in the main coolant of Light Water Reactors	5.1

Afterwards this research will be extended to the analysis of the fission products released at the reactor shut-down.

6. Reference documents
Internal progress reports.

7. Degree of availability
To a limited extent, available with the authorization of the sponsoring organization.

ENEL- CRTN, Bastioni Porta Volta 10, I-20121 Milano

Politecnico di Milano-CESNEF, Via Ponzio 34/3, I-20133 Milano

Berichtszeitraum/Period 1.1. - 31.12.1978	Klassifikation/Classification 5.2	Kennzeichen/Project Number 06.01.11/03A (PNS 4315)
Vorhaben/Project Title Versuche zur Erfassung und Begrenzung der Freisetzung von Spalt- und Aktivierungsprodukten beim Kernschmelzen Experiments on Determination and Limitation of Fission and Activation Product Release during Core Meltdown		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe, Projekt Nukleare Sicherheit, IRCH
Arbeitsbeginn/Initiated 1973	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader Dr. H. Albrecht
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating Dec. 1978	Bewilligte Mittel/Funds

1. General Aim

Determination of the release fraction of the radioactive core inventory for various coremelting conditions.

2. Particular Objectives

Quantitative investigation of the release of fission and activation products during core heat-up and from a liquid melt, including also concrete; characterization of the physical and chemical behavior of the released products; development of techniques for reducing the release.

3. Research Program

3.1 Experiments with 30 g of inactive corium to investigate the melt behavior during induction heating in a ThO_2 crucible and to measure the release-fractions of the main components of the melt (Fe, Cr, Mn, Ni, Zr, U) as a function of temperature, atmosphere, and pressure.

3.2 Experiments with 30 g of corium containing activated steel and Zircaloy for preparation of tests with active fissium (see 3.3) and to measure release fractions of those elements which can not be analyzed precisely in the tests with inactive corium (e.g. Sn, Zr).

3.3 Release experiments with masses of 30 g - 3 kg corium containing slightly active fissium with simulated burn-up in the range of 10.000 - 50.000 MWd/t; same parameters as under 3.1.

3.4 Release experiments with fissium/corium and additions of CaO, SiO_2, concrete, and other materials.

4. Experimental Facilities, Computer Codes
   - Melting furnace SASCHA
   - transport and collection system for the released products
   - facility for production of slightly active fissium (FIFA)
   - several computer codes for evaluation of gamma-ray spectra using a PDP-11 or IBM-370 computer.

5. Progress to Date
   a) Reconstruction of the experimental facility:
      In future release experiments with melt masses of 100 g up to more than 1 kg, the concentration of the aerosol particles above the melt is expected to be about one order of magnitude higher than in previous experiments with 30 g corium. The transport and collection system was reconstructed, therefore, by increasing the diameter of the tubes, valves, and membrane filters. In addition, the electric power of the high frequency generator has been increased to 250 kW. It is now possible to melt fuel rod specimens of 65 mm length which contain 6 pellets in closed Zircaloy tubes. Thus, the influence of chemical reactions between the cladding material and the fission products or the fuel is more realistically covered than in previous experiments.

   b) Completion of the fissium production facility FIFA:
      After installation of additional devices such as a welding apparatus for enclosing the fissium pellets in Zircaloy tubes, a series of tests was carried out for optimizing the temperature program during sintering, measurement of the weight loss during sintering, and determination of the pellet homogeneity.

   c) Release Experiments:
      Five release experiments have been carried out under air with maximum temperatures ranging from 1920°C to 2400°C. The mass of the corium samples was 300 g in two cases and otherwise 148 g. For I, Cs, Se, and Cd, temperature-dependent release curves could be determined by use of an online measurement technique. For Zr, Mo, Ru, Ag, Sb, Ba, Ce, Nd as well as for Fe, Cr, Co, and Mn, the integral release was measured for each experiment.
      Additional experimental parameters were:
      system pressure: 1-2 bar
      heat-up rate : 1-2°C/sec
      gas flow rate : 10 l/min

simulated burn-up
of the fissium pellets: 40 000 MWd/t

6. Results

The results summarized below correspond to the point 3.3 of the research program:

a) The release of cesium and iodine is less than 1 % for temperatures up to 1200°C. It can be concluded, therefore, that both elements are present in the fuel as compounds. A total release is obtained for I up to 2000°C and for Cs up to 2400°C when realistic heat-up rates are applied.

b) The behavior of selenium is quite different if a reaction with the zircaloy cladding material can take place than in the case where the fuel pellet is not encapsulated in zircaloy. In the second case the fractional release was 24 % for temperatures up to 1800°C whereas it was one order of magnitude less in the first case. For tellurium, a similar chemical reaction and hence a similar reduction of release is expected, since Te is a homologous element of Se.

c) The fractional release of ruthenium up to 2400°C was 0.8 % which indicates that high volatile oxides (RuO_2, RuO_3, RuO_4) have been formed only to a small extent. This was not caused by a lack of oxygen supply within the crucible but rather by the shielding effect of the cladding, by the thermal instability of these oxides, and by the heat-up rate (cf.a)).

d) The chemical form of barium is apparently not metallic, because the fractional release up to 2400°C was only 2 % and the boiling point of elemental barium is 1638°C.

7. Next Steps

a) Continuation of the experiments in air with melt masses of 150/300 g of corium

b) Preparations for release experiments
   - in steam atmosphere with samples in the same mass range as above
   - with increased masses of 1-3 kg and with addition of concrete to the corium samples.

8. Relation with other Projects

PNS 4316: Development and Operation of Facilities for the Investigation of Fission Product Release in anLWR Core Metdown Accident

9. References

Report KFK-2600 (1978) pp. 41, 429 (in German)
pp. 95 (in English)
Report KFK-2700 (1978) pp. 32, 4300-19 (in German)
pp. 87 (in English)

H. Albrecht, V. Matschoß, H. Wild:
Experimente zur Erfassung der Freisetzung von Spalt- und Aktivierungsprodukten beim Coreschmelzen, Reaktortagung, 4.-7. April, Hannover

H. Albrecht, V. Matschoß, H. Wild:
Release of Fission and Activation Products during LWR Core Meltdown, ENS/ANS Topical Meeting on Nuclear Power Reactor Safety, Brussels, Oct. 16-19, 1978

H. Albrecht, A.P. Malinauskas:
Spaltprodukt-Quellterme bei Kühlmittelverluststörfällen und hypothetischen Kernschmelzenunfällen, Projekt Nukleare Sicherheit, Jahreskolloquium, 28./29.11.1978, KFK-2770 (in press).

10. Degree of Availability of the Reports

Unrestricted distribution.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	5.2	06.01.11/04A (PNS 4316)
Vorhaben/Project Title Entwicklung und Betrieb von Anlagen zur Untersuchung der Spaltproduktfreisetzung beim LWR-Kernschmelzenunfall Development and Operation of Facilities for the Investigation of Fission Product Release in an LWR Core Meltdown Accident		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KFK Projekt Nukleare Sicherheit, Abt.Ingenieurtechnik
Arbeitsbeginn/Initiated 1972	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader Perinić
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Investigation into the release and transport of radioactive materials under different core meltdown conditions (see project PNS 4315).

2. Particular Objectives

Development of technical means for performing tests allowing to record and limit the release of fission and activation products during core meltdowns.

3. Research Program

3.1 Development of a melting facility for generating simulated core melts (CORIUM).

3.2 Development of a facility for generating nuclear fuel material with a simulated burnup (FISSIUM).

3.3 Development of a measuring technique for monitoring the molten pool temperatures.

3.4 Development of melting crucible configurations allowing to retain corium melts.

3.5 Operation of the experimental facilities.

4. Experimental Facilities

4.1 Melting facility for low activity specimens (SASCHA). It is heated by direct coupling of the melt in the induction field of the inductor. Working frequency 50 kHz, terminal power 250 kW, molten mass 5.0 kg.

4.2 Vacuum melting furnace, resistance heated, max. 3000°C, 1 l volume.

4.3 Induction facility 500 kHz, 12 kW.

4.4 Facility for the fabrication of nuclear fuel with simulated burnup (FIFA). Nominal throughput per month 700 g of fissium pellets.

5. Progress to Date

Ad 3.1 Conversion work was performed in the SASCHA facility. A new coaxial power lead-in was developed.

Ad 3.5 Test operation was begun in late September 1978 in the SASCHA facility.

6. Results

Ad 3.1 The SASCHA facility was converted for operation with corium melts on a kilogram scale. The coaxial power lead-in installed lends itself for operation of the facility under air and inert gas atmospheres, respectively. However, for operation under a water vapor atmosphere a new coaxial power lead-in had to be designed. Unlike the old version, which had been cast in silicone rubber, all electrically insulating components are now made of Al_2O_3 ceramics. Completion of the new coaxial system is expected for the first quarter of 1979.

Ad 3.2 The work was completed on schedule.

Ad 3.3 The work was completed on schedule.

Ad 3.4 The work was completed on schedule for melting crucibles up to 50 mm in diameter.

Ad 3.5 Test operation of the SASCHA facility was begun in late September 1978; the plant has so far performed 14 melting experiments after conversion.

The FIFA facility is operated by the KTB Department of KfK.

7. Next Steps

Ad 3.1 The new coaxial power lead-in will be installed in the SASCHA facility. Afterwards the facility will be accepted, trial operation and test operation with corium melts on a kilogram scale will be started.

8. Relation with Other Projects

PNS 4315: Experiments on Determination and Limitation of Fission and Activation Product Release During Core Meltdown

9. References

-

10. Degree of Availability of the Reports

-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01. - 31.12.1978	5.2	06.01.13/03A (PNS 4333)
Vorhaben/Project Title Theoretische Arbeiten zur Beschreibung der Aktivitätsfreisetzung beim Kernschmelzen Analytical Work on the Prediction of Activity Release in a Core Meltdown Accident		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe Projekt Nukleare Sicherheit INR
Arbeitsbeginn/Initiated 01.01.1978	Arbeitsende/Completed 31.12.1979	Leiter des Vorhabens/Project Leader Dr. E. Fischer
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Development of a computer code to describe the release, transport and sedimentation of aerosols in core melt-down accidents.

2. Particular Objectives

Development of theoretical models for the description of the activity release in the different phases of a core melt-down accident. The experimental results obtained in the melting plant SASCHA shall be used to guide the modeling work.

3. Research Program

Identification of the physical and chemical processes which determine the fission product release during the different phases of the accident. Development of models suitable to describe these processes. Check of the models by experimental results obtained at the melting plant SASCHA.

4. Experimental Facilities

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5. Progress to Date

The fission products ruthenium and cesium may contribute significantly to the damage potential caused by a core meltdown accident /1/. Therefore, the release of Ru and Cs was estimated, taking into account the chemical reactions they may undergo.

A model for the release from non-boiling, homogeneous melts was developed /2/. An analytic expression for the mass transfer from a liquid pool to a moving gas atmosphere was used to predict experimental release rates. This model was used to analyze experimental data from SASCHA experiments with fissium (release of Sb and Cs), and with corium E (release of several elements in the melt).

6. Results

The analysis of the SASCHA experiment S-172 with fissium gave the following result: For Sb, the calculated release was 0.8 times the experimental value, for Cs it was a factor of 6.4 higher /2/. It was assumed in the calculation that Cs was present as a metal in the melt. If this is not correct, the overestimation can be explained. However, recently an experiment with corium E /3/ was also analyzed. The release of different elements (Fe, Cr, Mn, Sb, Sn) was overestimated, on the average, by one order of magnitude.

7. Next Steps

An attempt will be made to explain the existing overestimation of the release. It may be necessary to modify the model. Further on, it is planned to study the influence of the atmosphere on the release.

8. Relation to other Projects

PNS 4315

9. References

/1/ Reactor Safety Study, Appendix VI
WASH-1400 (1975)

/2/ W. Breitung: "Zur analytischen Beschreibung der in SASCHA-Experimenten gemessenen Spaltproduktfreisetzung"
PNS-Halbjahresbericht 1978/1, KFK-2700 (1978)

/3/ H. Albrecht, V. Matschoss, H. Wild
Investigation of Activity Release during LWR Core Meltdown
Nucl. Tech. 40 (1978), 278

10. Availibility of the Reports

Unlimited availibility

148.1.01		5-2
TITRE FLASH : EVALUATION DU TAUX D'EMISSION DES PRODUITS DE FISSION AU COURS D'UN ACCIDENT DE PERTE DU REFRIGERANT PRIMAIRE DANS UN REACTEUR A EAU SOUS PRESSION.		Pays FRANCE Organisme Directeur CEA/DgCS et EDF
TITLE (Anglais) FLASH : ASSESSMENT OF THE FISSION PRODUCTS RELEASE RATE DURING A LOSS OF COOLANT ACCIDENT IN A PRESSURIZED WATER REACTOR.		Organisme exécuteur CEA/DMECN/DMG Responsable
Date de démarrage 1/1/77	Etat actuel EN COURS	Scientifiques
Date d'achèvement 31/12/81	Dernière mise à jour MISE A JOUR N° 1 AU 13.12.78	

1. OBJECTIF GENERAL :

Déterminer le taux de relâchement des produits de fission hors d'un crayon subissant une rupture de gaine lors d'un accident de perte du réfrigérant primaire ayant pour conséquence une température de gaine comprise entre 800 et 1200 °C.

2. OBJECTIFS PARTICULIERS :

1. Etudier le relâchement par lavage ou effet thermique des produits de fission condensables, notamment l'iode, le césium et le tellure, initialement piégés par dépôt dans le jeu.
2. Vérifier les hypothèses usuelles de dégagement des gaz rares radioactifs à partir de l'émission hors de la gaine rompue.
3. Etudier l'effet du transitoire sur les taux de dégagement.
4. Etudier le dégagement post-accidentel.
5. Etudier, si possible, le taux d'immobilisation des produits de fission sous forme de composés.
6. Modéliser les résultats obtenus.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME :

Ces essais sont réalisés dans un dispositif composé d'une partie basse GRIFFON et d'une partie haute FLASH, et installé dans le réacteur SILOE à Grenoble.

4. ETAT DE L'ETUDE :

1. Avancement à ce jour :

L'étude neutronique et thermique du dispositif est terminée. Elle a montré la faisabilité du scénario suivant :

Puissance : 40 $KW.m^{-1}$, pression : 13 MPa, température de gaine : 330 °C.

Chasse de l'eau primaire pour simuler l'accident, suivie d'une dépressurisation primaire jusqu'à 2 MPa.

La maquette munie d'un crayon électrique a fonctionné en juillet 1978. Elle a permi de valider les calculs ; à partir du scénario ci-dessus, on obtient une évolution de la température jusqu'à 1000 °C. en 3 minutes.

2. Résultats essentiels :

Sans objet : la phase expérimentale du programme n'a pas encore débuté.

5. PROCHAINES ETAPES :

La première irradiation est prévue dans le cours du premier trimestre de 1979.

6. RELATION AVEC D'AUTRES ETUDES :

Cette étude doit être utilisée pour l'analyse de sûreté des réacteurs à eau ordinaire sous pression (évaluation du terme source de contamination lors d'un accident de perte du réfrigérant primaire). A ce titre, elle est complémentaire du projet PHEBUS.

148-1-03			5-2
TITRE MIGRATION DE LA CONTAMINATION DANS LES REACTEURS A EAU ORDINAIRE EN SITUATION ACCIDENTELLE			Pays FRANCE Organisme Directeur CEA-DSN
TITLE (Anglais)			Organisme exécuteur DSN-SRS-SESTR/CADARACHE Responsable
Date de démarrage 1976	Etat actuel En cours		Scientifiques
Date d'achèvement 1981	Dernière mise à jour 15.12.78		

1. OBJECTIF GENERAL

L'action a été redéfinie lors d'une réunion SESRS/SESTR. Les thèmes d'étude proposés correspondent aux besoins identifiés pour préciser les conséquences radiologiques des accidents de dimensionnement et des accidents hypothétiques.

2. OBJECTIFS PARTICULIERS

1/ Comportement de l'iode dans l'enceinte

1.1. - Détermination de l'isotherme d'adsorption de l'iode par le béton et l'acier peint ou non en fonction de pression température (vapeur préchauffée) et mesure du coefficient de partage atmosphère eau (vapeur saturante).

1.2. - Taux de remise en suspension de l'iode adsorbé sur une paroi ou dans l'eau en cas de dépressurisation (cinétique).

1.3. - Détermination de la répartition des iodes pénétrants - non pénétrant dans une enceinte adiabatique, (pression de vapeur sèche ou saturante, 140° C - 170° C) ; recherche de l'état d'équilibre, cinétique d'atteinte de l'état d'équilibre.

Paramètres : - masse d'iode injectée,
- forme d'iode injectée,
- présence de matériaux organiques,
- rapport surface/volume,
- irradiation.

2/ - Participation au programme PHEBUS : mesure de l'émission de radioactivité dans un accident de perte de réfrigérant. Mesures des prosuits de fission émis, mesure d'hydrogène, étude du comportement des P.F dans l'enceinte de dépressurisation.

3/ - Définition d'un projet d'expérience visant à préciser le terme d'émission de produits de fission en cas de cisaillement d'un assemblage de combustible irradié, lors d'un accident de manutention.

3. INSTALLATION EXPERIMENTALES ET PROGRAMME

- Cuve 400 litres, pression 10 bars.

PROGRAMME :

POINT 1.1 et 1.2	1979-80
POINT 1.3	1979-81
POINT 2.	lié au planning SES
POINT 3.	1979

4. ETAT D'AVENCEMENT

1/ Expérience Pirée-Manutention

L'étude théorique a montré que ce paramètre, qui détermine la surface d'échange est le plus important par rapport aux paramètres : température, concentration. Un montage expérimental simulant la partie haute d'un assemblage combustible a été réalisé et est en cours d'essais.

2/ Phébus

Les divers appareils sont réalisés (May Pack) ou en cours de réalisation (coquilles chauffantes). Le banc d'essais est terminé.

3/ Formes d'iode

La conclusion de l'étude théorique est la suivante : la présence d'iodométhane (ICH_3) dans l'eau d'un PWR ou dans l'enceinte est improbable.

La mise au point de la recherche des formes d'iode dans l'eau des PWR s'es poursuivie. Après les essais à haute température nous allons explorer la gamme ... C.

.../...

4/ Dépôt d'iode sur le béton

Les générateurs de vapeur et d'iode moléculaire ont été réalisés et étalonnés. L'étude est étendue aux accidents hors dimensionnement : cas ... rupture de l'enceinte, les dépôts sont-ils réémis ?

5. PROCHAINES ETAPES (1979)

1/ Détermination de l'isotherme d'adsorption de l'iode par le béton en fonction de la température et de l'humidité relative :

Les essais seront effectués dans un réacteur en verre à la pression atmosphérique.

1° Semestre : étude en faisant varier la température de 40°C à 170°C avec de l'air sec.

2° Semestre : même étude en faisant varier l'humidité relative.

2/ Réémission de l'iode adsorbé par balayage d'air.

3/ Détermination de la répartition des iodes pénétrants non pénétrants dans une enceinte.

Une phase préparatoire à l'étude complète consistera à étudier le comportement des May Packs en fonction de la variation du terme source.

La réalisation des équipements nécessaires à l'étude complète (cuve et matériel de mesure associé) pourra être entreprise en fonction des disponibilités financières.

4/ Phébus

Les essais programmés cette année par le SES seront effectués à l'aide de combustible vierge à faible puissance. La probabilité de rupture de gaine est faible.

On se contentera donc de suivre les essais.

On effectuera les étalonnages nécessaires pour les mesures en spectro γ qui seront effectuées par la suite.

.../...

7. DOCUMENTS DE REFERENCE

J.M. VINSON - N. SESTR 19 - 13/10/78
Etudes des possibilités de formation de l'iodométhane dans les réacteurs à eau pressurisée

B. BOSCA - J.M. VINSON - G. MINGUELLA - J.M. DUBOIS
Etude par une méthode électrochimique des formes de l'iode dans l'eau primaire des réacteurs pressurisés
SESTR 12/78.

J. PORCHERON
Expérience PIGME - Accident de manutention de combustible irradié. Influence des conditions de rupture et de la rétention de l'iode dans la gaine.
Nt SESTR 20.

Mise au point et étalonnage d'un générateur d'iode moléculaire
R. SESTR 78/209.

IWP-1-04 / 4151

5-2

TITRE NATURE CHIMIQUE DES PRODUITS DE FISSION DANS LE COMBUSTIBLE DES REACTEURS A EAU.		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) CHEMICAL COMPOSITION OF FISSION PRODUCTS IN LWR FUEL ELEMENT		Organisme exécuteur CEA/DMG Responsable
Date de démarrage 1.79	Etat actuel	Scientifiques
Date d'achèvement 12.79	Dernière mise à jour 1.79	

1. OBJECTIF GENERAL

Connaissance de la nature chimique des produits de fission pendant les différentes phases des accidents hors dimensionnement entraînant la fusion du coeur.

2. OBJECTIFS PARTICULIERS :

Effectuer une étude bibliographique d'une part, et, d'autre part de rassembler l'ensemble des connaissances actuellement acquises sur ce sujet au cours de travaux antérieurs. Cette étude serait limitée au combustible des réacteurs à eau. Elle serait complétée par une étude de faisabilité d'une expérience en pile pouvant aller jusqu'à la fusion des pastilles du combustible en atmosphère contrôlée.

N.V. KEMA	CLASSIFICATION : 5.2
TITLE : Berekening van hoeveelheden radioactiviteit vrijkomend bij een ernstig reactorongeval.	COUNTRY: THE NETHERLANDS
	SPONSOR : KEMA ORGANIZATION : KEMA
TITLE (ENGLISH LANGUAGE): Calculation of the quantities of radioactivity released as a result of a serious reactor accident	PROJECTLEADER :
INITIATED : - LAST UPDATING : 1978	SCIENTISTS : K.P. Termaat
STATUS : - COMPLETED : 1977	

General aim

In analysing the risk of a nuclear power plant for the surrounding population one has, among other things, to consider the quantity and the nuclide spectrum of the radioactivity released to the environment as a result of a reactor accident with a non zero probability.

Particular objectives

Calculations are performed with the programme CORRAL developed by Battelle NW USA. Minor changes were introduced.
The basic objective has been to contribute in the risk analysis referring to an enlarging nuclear power programme in this country.

Experimental facilities

Not applicable.

Project status

Necessary calculations have been performed. An improvement in applied input data to the programme CORRAL could be usefull.

Next steps

Not applicable.

Relation to other projects

Equivalent calculations are performed in the USA as a contribution to the Rasmussen-study (WASH-1400).

Reference documents

See 2 and 6.

Degree of availability

Internal report.

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	5.3	RS 209
Vorhaben/Project Title Aktivierte Korrosionsprodukte in LWR-Kreisläufen Aktivated Corrosion Products in LWR Loops		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 45, Erlangen
Arbeitsbeginn/Initiated 1. 1. 76	Arbeitsende/Completed 31. 3. 81	Leiter des Vorhabens/Project Leader Dr. Neeb
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 78	Bewilligte Mittel/Funds

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. 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 date for PWRs and BWRs.

3.2 Data balancing in order to identify radionuclide sources.

3.3 Evaluation of the actual Co-contents in the construction materials.

3.4 Improvement of analytical methods.

3.5 Specific PWR tests.

3.5.1 Variation of operation parameters.

3.5.2 Exchange behaviour of deposit- and protective layers.

3.6 Specific BWR tests.

3.6.1 Compilation of loop surfaces of various materials and their metal erosion rates.

3.6.2 Contamination influence of high-Co-Containing materials in a neutron field.

3.7 List of PWR and BWR contamination models.

4. Experimental Facilities

The necessary test facilities for radio-chemistry, the analysis and measurement techniques, and the coolant chemistry and hot cell techniques are available. All measurements necessary will be performed in KWU laboratories.

Provision of various analysis samples (system- and fuel assembly deposits), the collection of several data points are closely tied in with reactor refuelling shutdowns.

5. Progress to Date

To 3.1 In some PWR and BWR plants, aerosol and reactor water samples are taken from the coolant once a month during an operating cycle under constant-load conditions and analyzed for the specific activities of 55-Fe and 60-Co. The first results are available for KKS.

To 3.4 The investigations for developing an analytical method for the specific ^{63}Ni activity are drawing to a close. Research has begun into possibilities for analyzing specific ^{59}Ni activity.

To 3.5 Primary water samples were taken during the shutdowns for the 1st and 2nd refuelings in Biblis-A and analyzed for specific activities of 55-Fe and 60-Co. These operations have been concluded.
Particulates have been taken from different systems by the Stade plant laboratory during the shutdowns for the 4th and 5th refuelings and in the start-up phases for the 5th and 6th cycles. The specific activities of 55-Fe, 59-Fe and 60-Co were determined from these samples.
In KWO 3 primary water samples were taken during plant start-up for the 8th refueling and analyzed separately for particulates > 0.45 μm and "soluble" corrosion

products < 0.45 μm.
Specific ^{55}Fe, ^{59}Fe and ^{60}Co activities and the Ni contents were determined in the primary coolant samples drawn in KWO upstream and downstream of the particulate filters during undisturbed constant-load operation.

In GKN, samples were taken of the deposits in the following primary system components during the shutdown for the 1st refueling: pressurizer safety valve, No. 1 steam generator manhole cover, reactor coolant and residual heat removal pumps. The deposit samples were analyzed for specific activities of Fe-55 and Co-60. The results of the Fe and Fe-55 analyses are available; the Co analyses have not yet been completed.

To 3.6 Reactor water samples were taken from the KWW cycle during constant load phases. Specific activities of ^{60}Co, ^{55}Fe and ^{59}Fe were determined on one of the reactor coolant samples.

6. Results

To 3.4 Nickel has to be separated chemically to determine the ^{63}Ni activity. Afterwards, ^{63}Ni activity can be determined by the fluid scintillation method. With this, the ^{59}Ni activity interferes, so that this radionuclide has also to be determined. Preliminary studies have so far been concluded, that a start can be made on determining the analytical limits, accuracy and reproducibility of the method.

To 3.5 The investigations carried out until now show that in PWR plants, specific activities of ^{60}Co, ^{55}Fe and ^{59}Fe in Particulates > 0.45 μm are clearly higher than in the so-called "soluble" portion of corrosion products.

To 3.6 In contrast to this, the analyses of the BWR coolant sample showed an approximately equal distribution of the specific ^{60}Co, ^{55}Fe and ^{59}Fe activities to the particulates > 0.45 μm and the "soluble" portion. Whether this was a chance result for the BWR sample or really a different distribution of specific activities, must remain a matter reserved to

further studies.
It can be deduced from the divergence of the specific ^{59}Fe and ^{55}Fe activities of particulates > 0.45 µm, that the specific Fe activities have not arisen from one single activation in the neutron field but that multiple activations with decay times interspersed between them outside the neutron field probably played a part.

7. Next Steps

To 3.5 + 3.6 Further determinations of specific activities of different radionuclides in samples from BWR and PWR loops.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	5 3	RS 204
Vorhaben/Project Title Dosisabbau Dose Reduction		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 312, Erlangen
Arbeitsbeginn/Initiated 1. 4. 76	Arbeitsende/Completed 30. 9. 80	Leiter des Vorhabens/Project Leader Dr. Wille
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 78	Bewilligte Mittel/Funds

1. General Aim

Improvement of currently used units and development of advanced techniques to handle gaseous and liquid radioactive waste, their optimization and the application of both lab and operating plant experience to technically mature systems.

2. Particular Objectives

2.1 Gaseous Activities

In order to reduce radiation exposure, the quantities of long-lived nuclides which are released by leakage and upon reactor shutdown will be reduced as much as possible by continuous extraction of radioactive gases from the primary coolant.
In addition, further improvement of the measurement and control of hydrogen and oxygen in the off-gas system will be pursued.

2.2 Water-Soluble Activities

By use of a Caesium-specific Ion exchange resin, an increased service life and minimization of Cs-build-up in the primary coolant, thereby leading to a reduction of personnel exposure, will be achieved. In addition, waste water evaporator distillates from NPP will be decontaminated with I. T. to achieve optimum waste water purification

2.3 Decontamination

Decontamination methods will be further developed to the point of application in a NPP, so as to reduce exposure of personnel during repair work.

3. Research Program

3.1 Gas Treatment

Laboratory test and evaluation of a noble gas separating test facility and development of an overall applicable system, examination of oxygen- and hydrogen measuring apparatus and selection of measurement procedure.

3.2 Water-Soluble Active Wastes

Examinations of decontamination of primary coolant and waste water evaporator distillates with filters and Ion-exchangers; magnitude and origin of corrosion products in the primary coolant (primary circuit and emergency systems).

3.3 Decontamination of Units and Containers

Decontamination of large-scale tanks in NPPs by means of special methods; preparation and treatment of decontamination solutions.

4. Experimental Facilities

Tests will be performed at model test stands available either in Erlangen or in NPPs.

5. Progress to Date

To 3.1 Cold tests were performed for the operational modes of purging with a gas free of noble gas and for the motive gas mode. The last two tests were performed with a purge gas flow with a krypton/xenon ratio that could be the same as the ratio during power plant operation found empirically in the least favorable conditions.

Constant gas flow control by the low-temperature system was by way of trial altered to being a function of the gas flow pressure at the inlet and not a function of the flow rate, as was the case until now.

New calibration curves had to be drawn for the gas chromatograph because of the higher concentration of noble gas in the regenerated fluid. The adsorber unit was dismantled

and activated charcoal samples for analysis.
Necessary maintenance jobs and reconstruction were carried out, including the preparation of radioactivity measuring points so that, as far as was possible, a change could be made from manual operation to automatic.

The dismantled adsorber unit was refilled with activated charcoal and assembled using for flask 1-3 used activated charcoal but for flask 4-7 fresh activated charcoal.
Two measuring heads were made to measure radioactivity and lead plates ordered to shield the measuring points. A separate instrument line was laid for ß-measurement so that a changeover can be made from ß-measurement to γ-measurement depending on the measuring gas radioactivity. Adsorber unit cooling with liquid nitrogen is now undertaken by a control circuit, so that constant charcoal temperatures can be expected during operation. The experimental plant has been moved to the radio chemistry section and a helium leak test already performed.

To 3.2 In December, a particle counter was tested in Gösgen Nuclear Power Plant near the seal water filters during the 2nd hot trial operation. All resin samples taken in Biblis B NPP in May and July were analyzed with regard to nuclides.

To 3.3 The reconditioning of decontamination solutions containing complexing agents was examined:

- Development of quantitative analysis methods to include the complexing agents
- Studies concerning the treatment of original decontamination solutions from the chemical two-stage decontamination process
- Studies concerning the treatment of diluted decontamination solutions such as reinsing and washing solutions, etc.

Preparation of a study of the construction of a mobile decontamination system.

All stages of the process should be automated as far as possible in order to reduce the radiation exposure of the decontamination staff. This is achieved by the remote indication of the following criteria: liquid levels temperatures and pH values of solutions; positions of distributor valves, pums and heaters.
All above-mentioned parameters can be corrected by remote control operation and positions changed on individual components.
A test program was prepared to optimize the single-stage decontamination (soft decontamination) process

1. Oxide solving tests as a function of the acid concentration, pH value and temperature.
2. Definition of corrosion rates of structural materials used in the primary circuit under the conditions found in subsection 1.
3. Temperature behaviour of acids used in autoclave tests.
4. Finding the most suitable ion exchanger for decontaminating the decontamination solution produced.
5. Testing of all results obtained in laboratory; (nonradioactive).
6. Transfer of the process to the nuclear area.

The chemical two-stage decontamination process was employed for decontaminating 4 reactor coolant pumps and 2 pressurizer heating rod bundles in Biblis Nuclear Power Plant, Unit A.

6. Results

To 3.1 The retention capacity for xenon does not present any problem at all compared with that for krypton, so that the static capacity for krypton is the factor governing system design.

Prolonged purging and motive gas operation exert a major negative influence on the static working capacity, like the absolute flow of noble gas with which the irradiated charcoal was loaded.

The system operated in a faultfree manner while the five tests were being made.

Pressure control proved to be more advantageous during changeover to motive gas operation and especially during system startup and rundown, as the possibility of the refrigeration machine freezing up during rundown was excluded.

To 3.2 An evaluation has not yet been made.

To 3.3 The complexing agents can be destroyed entirely by $KMnO_4$ in the original decontamination solution.
Good separation factors were achieved for ^{134}, ^{137}Cs, ^{124}Sb, ^{54}Mn and ^{65}Zn after complex destruction in different precipitation stages.
A study was made of the construction of a mobile decontamination system. The experience gained in decontamination operations in nuclear power plants was used as the design basis.
A test program was prepared for optimizing the soft decontamination process.
In the decontamination of the reactor coolant pumps and heater rod bundles, the dose rate could be lowered from max. 60 R/h to $\leq$ 1 R/h.

7. Next Steps

To 3.1 Trial operation with active krypton can probably be started early next year.

To 3.2 Allocation to operating data has still to be clarified.

To 3.3 Mobile decontamination system
Design of components needed for the mobile decontamination system.
Soft decontamination processes.
Start of series examinations for oxide solubility.

8. Relation with Other Projects

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9. References

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10. Degree of Availability ---

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	5.3	RS 238

Vorhaben/Project Title	
Entwicklung eines Systems zur Absaugung von Anlagenteilen und Armaturen Development of a Suction System for Installations and Fittings	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 312, Frankfurt

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 10. 76	31. 3. 78	H. Queiser
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	31. 12. 78	

1. General Aim

Previously, there were no requirements for retention of noble gases comming up with the vent-gases of the liquid-waste-tanks and the gland-leak-off-system.

Due to the extremely low release rates now demanded by the licensing authorities, however, noble gas retention systems must be developed and connected into activity paths previously thought to be negligible as far as activity is concerned.

2. Particular Objectives

Investigations will be made on activity-release, -paths and -concentration as well as on the development of systems for activity collection and retention.

In particular, noble gas evolution and -release from iodine-containing ion-exchange resins, e.g. reactor water clean-up-filter, will be investigated for a certain period of time after resin discharge.

3. Research Program

3.1 Experimental determination of the noble gas release from ion-exchange resins in nuclear power plants during transportation of suspended loaded resins to the waste tanks and under the operating conditions present in these tanks (temperature, pressure, movement, filling- and discharge rhythm, etc).

3.2 Verification of the calculated noble gas concentration and sources by measurements performed in the components of the water clean-up-systems in nuclear power plants.

3.3 Determination of iodine and noble gas paths in the reactorwater- and -steam circuit and the auxilliary systems.

3.4 Determination of the fluid motions in the water decontamination systems with respect to quantities and time sequence.

3.5 Investigations on the gas flow in the water clean-up systems.

3.6 Investigations on possible gas-flow-retention within water clean-up system containers.

3.7 Experimental investigation of noble gas adsorption observed on charcoal under the following conditions:

a) loading with steam-saturated gas flows at higher temperatures

b) Operation of charcoal columns during alternating loads (counter flow) as well as different steam saturation degrees of gas flow observed at R.T. and elevated temperature.

c) loading with steam saturated gas flow at batch-wise operation and higher temperatures.

3.8 Experimental measurements and investigations of leakages observed in glands and leak-off systems as well as possible reduction of air-in-leakages and steam contents in a nuclear power plant in operation.

Concept and optimization of systems with charcoal filter designed for:

a) collection of gland leakages

b) treatment of motive air quantities from water clean-up system containers with higher activity

c) exhausting from the H_2 concentration limitation system

d) degasification of the primary circuit after plant shut down

e) coolant filter cleaning by means of purging air.

4. Test Facilities

Tests will be performed by the "Bergbauforschung/Essen GmbH".

5. Progress to Date

To 3.6 Determination of the activity flow in the air circulation system considering plant related data and the gas distribution conditions present in the various containers.

To 3.7 The "Bergbauforschung/Essen" (Mining Research Ass.), will perform tests on loading and reflooding of activated carbon, considering the specific design of the apparatus.

To 3.8 Concepts of a component suction system, consisting of the gland leak-off, containment, ventilation- and deaerating systems, penetration suction were completed in the context of system-, control and instrumentation technology.

6. Results

To 3.6 Investigations on the mixing conditions present in the air phase of different containers showed that with partial degrees of filling the complete space will not participate in the air exchange. The range within the container hardly - or not at all - exceeds the clearance volume, so that any this volume can be regarded as aerated.

7. Next steps

Project completed.

8. Relation with Other Projects

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9. References

V. Tiegs, W. Rüffer, P. Sydow
Development of a Suction System for Installations and Fittings
Final Report BMFT RS 238, November 1978

10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78-31.12.78	5.3	06.01.11/01A (PNS 4311)

Vorhaben/Project Title	
Untersuchungen zur Wechselwirkung von Spaltprodukten und Aerosolen in LWR-Containments Investigations on the interactions between fission products and aerosols in the atmosphere of LWR-containments	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Kernforschungszentrum Karlsruhe GmbH (KFK) Projekt Nukleare Sicherheit (PNS) - LAF I

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.77	31.12.81	Dr. W. Schöck
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	December 1978	

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 m³ 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 Mod3) calculates the aerosol removal processes coagulation, sedimentation, thermophoresis and diffusion as well as steam condensation and the significant thermodynamic functions for realistic PWR core melt sequences.

5. Progress to Date

The experiments concerning the condensation of steam onto aerosol particles were continued varying particle size, temperature, composition of the atmosphere to cover the whole field of parameters necessary and to improve the statistics of the experimental results. Furthermore a formula was developed to take the condensation shape factor into account.

6. Results

The tendency shown already in the former experiments was confirmed that the heterogeneous condensation of the steam onto the particles is dominated by their geometrical structure and their material. This fact can be seen by the different condensation behavior of new and aged aerosol. The condensation onto new aerosol begins earlier and involves smaller particles than in the case of an aged aerosol. This can be explained by the aggregate structure which is far more distinct in the case of the new aerosol than in the case of the aged aerosol with a more spherical shape. Nevertheless, the calculations made parallel to the experiments show that the difference between the condensation velocities of the aerosols and of water droplets can be accounted for by a single probably material dependent shape factor. The insertion of the shape factor into the model NAUA-Mod3 did not change the results very much compared to calculations without the shape factor. If this fact is generally valid or depends only on the special steam source function which was used in the calculations has to be investigated in the future.

7. Next Steps

To improve the shape factor model some more experiments with soluble aerosol particles are necessary. The main subject of the near future work, however, will be the quantification of the influence of wall effects on the aerosol system. Especially the partitioning of steam between walls and particles will be measured.

8. Relation with other Projects

PNS 4243

9. References

(1) Bunz, H., Schöck, W., Aerosol behavior in the condensing steam atmosphere of a post accident LWR containment, Intern. Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, October 16-19, 1978

(2) Bunz, H., Schöck, W., The behavior of highly concentrated aerosols in a condensing steam atmosphere, 6. Fachtagung der Gesellschaft für Aerosolforschung, Wien, 26.-28.9.1978

(3) Bunz, H., Schöck, W., Abbau freigesetzter Spaltprodukte in LWR-Sicherheitsbehältern, Jahreskolloquium 1978 des Projekts Nukleare Sicherheit, Karlsruhe, 28.-29.11.1978. In: KfK-2770 (Nov. 78)

10. Availability of Reports

Unrestricted distribution

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	5.3	06.02.03 (PNS 4621)
Vorhaben/Project Title Krypton- und Xenon-Entfernung aus der Abluft kerntechnischer Anlagen Separation of Krypton and Xenon from the Offgas of Nuclear Facilities		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe Projekt Nukleare Sicherheit IHCH, IT, IRCH
Arbeitsbeginn/Initiated Januar 1972	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader R.v.Ammon, E.Hutter, R.D.Penzhorn
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Development of the separation technic for the fission product Kr-85 out of the fuel reprocessing off gas. Safe storage for about 100 years.

2. Particular Objectives

Pilot tests for:

- Catalytic reduction of O_2 and NO_x with H_2
- Adsorption of H_2O, CO_2 (NH_3, NO_x and other impurities)
- Separation of Kr and Xe out of a N_2-stream by cryogenic distillation
- Concentration and storage of Kr-85

3. Research Program

3.1 Column behavior and separation factors achievable in the cryogenic distillation of the N_2-Ar-Kr-Xe system

3.1.1 Prevention of plugging by desublimation

3.1.2 Accumulation of impurities, e.g. O_2, CH_4, NO

3.2 Prepurification of the offgas

3.2.1 Development and testing of catalytic methods for the reduction of O_2 and NO_x with H_2 and for the oxidation of hydrocarbons

3.2.2 Determination of the adsorption and desorption behavior of Kr, Xe, H_2O, CO_2, NO_x and NH_3 on various adsorbents

3.3 Ozon safety

3.3.1 Development of analytical methods for the surveillance of ozone

3.3.2 Investigation of the behavior of ozone under cryogenic conditions and determination of phase diagrams.

3.4 Storage of Kr-85

3.4.1 Development of a containment for the storage of Kr-85

3.4.2 Solidification of fission noble gases.

-2-

4. Experimental Facilities, Computer Codes

5. Work Completed

Two test campaigns of six weeks duration each without interruption were conducted at the test facilities ADAMO and KRETA. The anayltical instrumentation was extended by a sampling station of the liquid bottom phase of the first column and by the determination of argon as an additional gas component.

ADAMO was operated separately twice for a two and four weeks period, respectively, for the determination of the adsorption and desorption of the rare gases and of the breakthrough of CO_2 and H_2O through the adsorption beds. A radioactive tracer technique using Kr-85 was applied for analysis.

The laboratory studies for the collection of dynamic adsorption data were continued using Kr and Xe as adsorptives at low temperatures and N_2O and CO_2 at room temperature.

The laboratory studies on the minimization of gas components passing the reduction catalyst forming potential hazards in the first cryogenic column, e.g. reduction of CO_2 to CO and CH_4, were continued. The trace analytical instrumentation was improved.

The catalytic test facility REDUKTION was delivered by the supplier.

The reaction velocity of the thermal decomposition of ozone was measured in glass and stainless steel as a function of the surface/volume-relationship at various temperatures.

To calibrate the IRCH cryostat, several p-T-x-measurements were carried out on the CHF_3/O_2 system. The explosion limits of several ozone/oxygen/noble gas mixtures were measured.

First results on the corrosion of Rb and Rb_2O on stainless steel (1.4306 and 1.4301) and welds (1.4316) at 100 bar and 150 ^{o}C after 3000 k were obtained.

The fixation of noble gases in zeolithes was investigated as an alternative to the storage of Kr-85 in pressure steel cylinders.

6. Results

Operations leading to a temperature increase in the vicinity of the feed entry into the first KRETA column reduce the tendency of xenon to desublime according with theory. Two such operations are: addition of 1 % argon to the feed gas (temperature increase at the sieve plate above the feed entry: 1 - 2 K) and pressure increase from 5 to 6 bar (temperature increase: 2 - 3 K). The upper limit of xenon concentration in the feed gas where the column can be operated continuously without perturbations is in the vicinity of 800 ppm (by vol.) at 5 bar without the addition of argon. Temporarily higher xenon concentrations can be fed to the column perturbations, e.g. the concentrations of a simulated fuel dissolution in the WAK (1600 ppm at maximum). Xenon freezing from the liquid phase was not observed, not even from a bottom product containing 80 % Xe and 20 % Kr.

From the mass balance of the rare gases fed into the KRETA plant during one campaign and bottled at the end, a Kr decontamination factor of 25 for the whole plant was obtained. Main points of leakage are the sampling lines.

During the test runs at the ADAMO facility the fast and complete desorption of Kr from silicagel and molecular sieve 10 A during the purging step at room temperature was ascertained.

The breakthrough times for CO_2 and H_2O through an ADAMO adsorption line follow the expectations from literature data.

Laboratory tests on the adsorption capacity of molecular sieves for Kr showed an increase by a factor of about 3 at $-50^{\circ}C$ compared to room temperature.

The dynamic adsorption coefficient of N_2O is of the same order of magnitude as of CO_2 and NH_3. Its adsorptive retention therefore poses no problem.

The reduction of CO_2 with H_2 at the reduction catalyst depends strongly on H_2 and O_2 concentrations and on temperature: CO formation increases with increasing [illegible] formation decreases. Under the operational conditions to be [illegible] in a real offgas system ($550^{\circ}C$, concentration of O_2: 2 %, excess of H_2: 1000 ppm) the formation of CH_4 is $\leq$ 1 ppm on ruthenium and oxidic catalysts.

Whereas ozone decomposes homogeneously at temperatures above 100 °C after the 2nd order, the reaction becomes of the order 1.5 at temperatures below 25 °C. The reaction is very slow in glas vessels. In metal surfaces a catalytic process markedly accelerates the reaction rate.

To test the chemical analysis of ozone under hot conditions, it was necessary to investigate the adsorption of Kr-85 on active charcoal as a function of the container dimensions, the flow velocity and the composition of the feed gas.

On the basis of p-T-x-measurements for the CFH_3/O_2 system, the cryostat from the IRCH was calibrated.

A total of 27 sample bubbles filled to half height with Rb, loaded with a pressure of 100 bar Ar have been kept for more than 8000 h at 150 and 200 °C. The parameters were: temperature, concentration of Rb_2O (simulation of an O_2 impurity) and time. First results on some of the materials show that the mechanical properties (yield point, tensile strength and point of max. elongation) are not significantly altered. Metallographic tests however indicate that in the presence of O_2 an intercristalline corrosions becomes apparent.

More than 30 different zeolites were loaded with noble gas and the thermal stability, radiation resistance and resistance to humidity of the obtained product investigated. Several zeolites were found that process a very high thermal stability. This was found from ling term experiments at 200 and 400 °C. The negative effects became apparent, when some of the samples were submitted to a radiation dose of $3 \cdot 10^8$ rad.

7. Next Steps

Presently constructive changes at the interior of the first KRETA column (sieve plates) near the feed entrance are being carried out in order to prevent plugging by desubliming Xenon, if the feed gas contains higher Xenon concentrations. After completion of these changes the plant will be operated again.

The accumulation of oxygen contained in the feed gas in small amounts will be studied experimentally in the plant. The results are to be compared with c calculations.

The laboratory study for the collection of adsorption data are continued. Additionally, the possible formation of solid salts from NO_x and NH_3 will be investigated.

As soon as the hall and the gas storage for the pilot plant REDUCTION at the new site is constructed, the plant will be mounted and prepared for the start of operation.

The laboratory studies concerning catalytical offgas cleaning methods will be continued. They will include the oxidation of hydrocarbons.

Primary objectives of the future work is to test the analysis of ozone under hot conditions. Furthermore, it should be investigated whether ozone is completely soluble in Xe and Kr at low temperatures. The effect of HC on the explosion limits of ozone rich gaseous mixtures will be examined.

The Rb corrosion experiments as well as work destined to develop a Kr-85 storage vessel will be continued. Additional materials will be examined for Rb corrosion. The possible effect of radiation as well as other up to now not investigated impurities (H_2O, N_2) will be incorporated into the research program.

The work on the fixation of Kr in zeolites will be continued, with the objective of improving the loading conditions as well as the thermal stability and the heat transport form the loaded pellets.

8. Relations with Other Projects

KFA/ICT Jülich, C.E.N. Mol (Belgien), Fraunhofer-Institut für die Chemie der Treib- und Explosivstoffe, Pfinztal, Institut für Verfahrenstechnik der TU Berlin, Uni Karlsruhe, Uni Hamburg.

9. References

H.Jüntgen, H.J.Schröter, R.v.Ammon, C.H.Leichsenring, Kerntechnik 20, 450 (1978)

R.v.Ammon, H.G. Burkhardt, E.Hutter, G.Neffe,
15th DOE Air Cleaning Conf., Boston, Mass., August 7 - 10, 1978

R.v.Ammon et al., Deutsches Atomforum, Reaktortagung 1978, Hannover

R.-D. Penzhorn, K.Gunther, P. Schuster;
15th DOE Nuclear Air Cleaning Conference, Boston, Mass., August 7 - 10, 1978

10. Availability: free (KfK/LA)

Berichtszeitraum/Period 1.1. - 31.12.1978	Klassifikation/Classification 5.3	Kennzeichen/Project Number 06.01.14/02A (PNS 4415)
Vorhaben/Project Title Entwicklung von Abluftfiltern für Unfallbedingungen Development and Improvement of Exhaust Air Filters for Accident Conditions		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KfK/PNS LAF II
Arbeitsbeginn/Initiated 1978	Arbeitsende/Completed 1981	Leiter des Vorhabens/Project Leader H.-G. Dillmann
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 1978	Bewilligte Mittel/Funds

1. General Aim

Removal of fission products from the post-accident atmosphere of the safety containment by means of air filters including under conditions of venting the containment.

2. Particular Objectives

Removal of aerosols at high air humidity and elevated temperatures. Optimization of droplet separators.

3. Test Program

3.1. Testing of aerosol filters at elevated temperatures and high humidities of the air to determine their removal efficiency and mechanical properties.

3.2. Design of aerosol filter elements and filter systems and, possibly, new developments of aerosol filter elements together with industrial firms.

3.3. Optimization of droplet separators.

4. Experimental Facilities

Laboratory scale experimental facilities and technical scale test rig for investigation of true scale aerosol filter elements.

5. Work Performed

ad 3.1. Literature research concerning investigations of aerosol filters made at elevated temperatures, high humidities, pressure pulses, loadings and exposures to radiation; installation and commissioning of the AgI plasma aerosol generator together with all the associated supply units at the technical scale test rig.

ad 3.2. Experimental determination of the temperature rise within the test section due to the electric power supply of the aerosol generator, as a function of the air velocity, temperature and relative humidity.

ad 3.3. Optimization was completed with a mat combination consisting of 22 µm and 8 µm fibres.

5. Results Obtained

ad 3.1. The literature research which includes the most recent results, has shown that single investigations have partly been made relative to extreme impacts on aerosol filters. However, there are no measured data available yet about the effect of the combined influences of elevates temperature, high air humidity and condensate impact to be anticipated in accidents of nuclear power stations on the filtering efficiency of aerosol filter elements.

ad 3.2. The influence of temperature rise on the maximum decrease of the relative air humidity is 2 % in dry air of 30 ^{o}C and less than 0.5 % in humid air of 90 ^{o}C.

ad 3.3. Droplet separators of 22 µm and 8 µm metallic fibres, 12 mm thick each, yield removal efficiencies > 99.9 % with a pressure drop < 500 Pa.

7. Plans for Future Work

Recording the size distribution and evaluating the mass concentration of the AgI testing aerosol. Measurements on commercial filters in a view of their resistance to humidity, temperature and pressure.

8. Relation with Other Projects

PNS 06.01.14/01A, 03A

9. Literature

KFK 2600, 2800

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	5.3	06.02.02 (PNS 4611)
Vorhaben/Project Title Entwicklung von Abluftfiltern für Wiederaufarbeitungs-anlagen Development of Off-Gas Filters for Reprocessing Plants		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe GmbH LAF II, IT
Arbeitsbeginn/Initiated July 1971	Arbeitsende/Completed 1982	Leiter des Vorhabens/Project Leader Dr. Furrer, K. Jannakos
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

To remove fission product iodine and other contaminations from off-gas of fuel reprocessing plants a filter facility for the head end part is under development capable to retain the fission product iodine, including ^{129}I, in a form ready for final storage.

2. Particular Objectives

Filters habe 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 rig (no radioactivity above tracer level) simulating head end off-gas conditions for dummy components, PASSAT, 250 m^3/h of volumetric gas flow. 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 1400 t of heavy metal per year).

5. Progress to Date

to 3.1. In laboratory scale tests the influence was investigated of nitrogen monoxide in pure nitrogen on the removal efficiency of the AC 6120-12 % Ag iodine sorption material.

to 3.2. The PASSAT prototype off-gas purification system equipped with remotely operated droplet separators, aerosol filters and iodine filters started operation for dissolver off-gas purification under simulated dissolver off-gas conditions: The function of the gas simulation facility, sample introduction and collection were tested and calibrated and the testing program was started subsequently.

6. Results

The PASSAT facility was commissioned without major technical problems so that it was possible to start the test program. Equipment for droplet generation and measurement was calibrated. First investigations of the removal by prototype fiber package units, using droplets of about 1 - 10 µm diameter, yielded high retention factors for water droplets.

At nitrogen monoxide concentrations up to 1 vol.% in pure nitrogen as the carrier gas and with 24 h of exposure at test bed temperatures of 130°C and 150°C noticeable effects have not been found on the loading capacity and iodine removal of the AC 6120-12 % Ag iodine sorption material.

7. Next Steps

Continuation of the PASSAT test program including the investigations of the prototype fiber package removal units and the category S aerosol filters.

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

PNS Semi Annual Reports KFK 2600, KFK 2700.

Furrer, J.; Wilhelm, J.G.; Jannakos, K.: Aerosol- and Iodine Removal System for the Dissolver Off-Gas in a Large Reprocessing Plant. 15. DOE Air Cleaning Conf., Boston, Mass., August 7-10, 1978.

Berichtszeitraum/Period 1.1. - 31.12.1978	Klassifikation/Classification 5.3	Kennzeichen/Project Number PNS 06.01.14/03A (4416)
Vorhaben/Project Title Abluftfilterung an Reaktoren (Alterung und Vergiftung von Jod-Sorptionsmaterialien) Off Gas Filtering in Reactor Stations (Ageing and Poisoning of Iodine Sorption Materials)		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe GmbH PNS, LAF II
Arbeitsbeginn/Initiated 1973	Arbeitsende/Completed 1979	Leiter des Vorhabens/Project Leader J. Furrer
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Improvement of iodine sorption filters.

2. Particular Objectives

Development of improved iodine sorption filters for extended operational periods.

3. Research Program

3.1. Gas chromatographic analysis of the influent and effluent of iodine sorption filters in nuclear power plants (nukes). Determination of the dependency of the removal efficiency for radioiodine on the nature and loading of poisoning components.

3.2. Development of an improved iodine sorption filter including in-place regeneration of the iodine sorption material.

4. Experimental Facilities

to 3.1. Gas chromatographic equipment. Test rigs in nukes for iodine sorption materials. Test apparatus for measurements of the removal efficiencies of sorption materials under simulated conditions of filter operation.

to 3.2. Test rig for whole filter units for a throughput of up to 2000 m^3/h.

5. Progress to Date

to 3.1. In addition, the ten most important filter pollutants in the room exhaust air were measured quantitatively during a six months period using a continuously operating gas chromatograph.

to 3.2. Loading of activated carbon for radioiodine removal together with the room exhaust air of a PWR nuclear power station furnished further results on the operation of a multiway sorption (MWS) filter.

6. Results

to 3.1. Evaluation of the gas chromatograms of the room exhaust air showed that during normal operation of the nuclear power station the concentration of the organic solvent amounts to about 6 - 10 mg/m^3. During or shortly after the phase of intervention of the nuclear power station solvent concentrations up to 63 mg/m^3 were measured.

to 3.2. Only after ten months of exposure the zone of the solvents with low volatility had penetrated into the second carbon bed of a 2-stage test filter. The integral removal efficiency of the second carbon bed decreased from 99.999 to 99.1 %.

7. Next Steps

The loading of test filters designed similar to MWS filters, the determination of the removal efficiencies and the gas chromatographic analysis of the filter eluates and the room exhaust air will be continued so as to obtain more data which are required for optimization. This work will be supplemented by the continuous gas chromatographic measurement of filter pollutants in the room exhaust air.

8. Relation to Other Projects

Results of PNS 4415 may contribute to the program described here.

9. References

PNS-Semi Annual Report (KFK 2600, KFK 2700).

10. Degree of Availability

Reports and publications are available without restriction.

Berichtszeitraum/Period 1.1. - 31.12.1978	Klassifikation/Classification 5.3	Kennzeichen/Project Number RS 221
Vorhaben/Project Title Bestimmung der Jodkomponenten in der Abluft kerntechnischer Anlagen Determination of the Iodine Species in the Exhaust Air of Nuclear Installations		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe Projekt Nukleare Sicherheit LAF II
Arbeitsbeginn/Initiated 1.12.1976	Arbeitsende/Completed 30.6.1979	Leiter des Vorhabens/Project Leader H. Deuber
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

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 concept of nuclear installations.

2. Particular Objectives

Determination of the radioiodine species elemental iodine (I_2), particulate iodine, and organic iodine (CH_3I).
(Assumed ratio of the thyroid doses caused by the release of equal amounts of these species in the FRG 100 : 10 : 1.)

3. Research Program

3.1. Laboratory tests: development of selective sorption materials for radioiodine species samplers.

3.2. In situ tests: operation of radioiodine species samplers in the exhaust air and stack effluent.

4. Experimentel 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 and stack effluent.

5. Progress to Date

to 3.1. Ascertainment of the removal efficiencies of several iodine sorbents for ^{131}I in the form of I_2, CH_3I, and HIO (hypoiodous acid).

to 3.2. Measurements with radioiodine species samplers in the exhaust air and stack effluent of PWRs (PWR 2 and PWR 3).

6. Results

to 3.1. The removal efficiency of the I_2 sorbent DSM 11 was high for ^{131}I in the form of I_2 and low for ^{131}I in the form of CH_3I even at extreme conditions (2 % and 98 % R.H. at 40 ^oC). ^{131}I in the form of HIO was retained by DSM 11 to a negligible extent in the relevant range of parameters and under the conditions indicated above, except for cases of low temperatures, relative humidities and short purging times. Thus, a sufficient separation of the iodine species I_2 and HIO is generally attained.

HIO was removed selectively by the sorbent IPH (from NES, U.S.A.) in a wide range of parameters; however, at a high relative humidity (80 %) or a long purging time (1 week) strong desorption of HIO from this material was observed. Desorption of HIO at a long purging time was also found in experiments on the removal of HIO by the KI impregnated carbon CG 0.8. The removal of HIO by iodine sorbents was practically not influenced by the impregnants of the sorbents.

to 3.2. In the stack effluent of PWR 2 the fraction of organic ^{131}I was predominant during power operation, that of elemental ^{131}I during refueling outage. In total (measurement time 1 year) the fraction of organic ^{131}I prevailed. The contrary held for the stack effluent of PWR 3. The annual ^{131}I stack release amounted to 0,5 m Ci at PWR 2 and to 5 m Ci at PWR 3. (Thus, the release of elemental ^{131}I was considerably lower than 50 % of the permitted total ^{131}I release.) At PWR 3 during power operation the main source of elemental ^{131}I was the hood exhaust. During refueling outage the mixture of the annular compartment exhaust and the containment purge contributed most of the elemental ^{131}I.

7. Next Steps

to 3.1. The development of a radioiodine species sampler applicable to the exhaust air of reprocessing plants will be continued.

to 3.2. Measurements with radioiodine species samplers will be carried out in the exhaust air of a reprocessing plant and of a BWR.

8. Relation with other Projects

Ageing and Poisoning of iodine sorbents.

9. References

KFK Ext. 30/78-1;
KFK 2600, p. 130 - 155.

10. Degree of Availability of the Reports

Literature Department, KFK, Postfach 3640, 7500 Karlsruhe 1.
Unrestricted.

	CLASSIFICATION: 5.3
TITLE (ORIGINAL LANGUAGE): Trattamento dei gas nobili radioattivi prodotti per fissione	COUNTRY: ITALY SPONSOR: CNR-CNEN-CCE
TITLE (ENGLISH LANGUAGE): Fission produced radioactive noble gases treatment	ORGANISATION: University of Pisa PROJECT LEADER: Giorgio CURZIO
INITIATED: July 1970 / COMPLETED:	SCIENTISTS: Franca CASTELLANI, Alberto GENTILI, Leonardo PIEVE
STATUS: in progress / LAST UPDATING: June 1979	

1. General aim
   Theoretical and experimental research on the general problems involved in the production, release, treatment and storage of radioactive noble gases.

2. Particular objectives
   - 2.1 Decontamination of the reactor off-gas by means of adsorption delay beds.
     - 2.1.1 Characterization of the adsorbent media and delay beds in laboratory scale.
     - 2.1.2 Characterization of the delay beds in real scale.
     - 2.1.3 Acceptance tests of delay beds in laboratory and industrial scale.
     - 2.1.4 Periodical tests of delay beds.
   - 2.2 Decontamination of the reactor off-gas by means of cryogenic techniques and other methods.
   - 2.3 Decontamination of the reprocessing plant off-gas.
     - 2.3.1 Cryogenic distillation.
     - 2.3.2 Adsorption in organic solvent.
     - 2.3.3 Production of solid matrix and other methods.
     - 2.3.4 Experimental determination of the concentration of noble gases and other gaseous radwaste in the off-gas of a reprocessing plant.
   - 2.4 Storage of Kr^{85}.
     - 2.4.1 Storage by bottling.
     - 2.4.2 Storage by bottling in containers filled with adsorbent media.
     - 2.4.3 Storage by implantation in solid matrixes.
   - 2.5 Cost-benefit analysis
     - 2.5.1 Cost-benefit analysis of nuclear reactors off-gas decontamination systems.
     - 2.5.2 Cost-benefit analysis of reprocessing plants off-gas decontamination systems.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Fission produced radioactive noble gases treatment. -2-	5.3

2.6 Experimental determination of the Kr^{85} concentration in air.
   2.6.1 Environment concentration determination.
   2.6.2 Concentration measures in the neighborough of nuclear plants.

3. Experimental facilities
   a- Charcoal bed testing facility in dynamic conditions.
   b- Nuclear detection devices.
   c- Granular charcoal testing facilities in dynamic and static adsorption conditions.
   d- Facilities for off-gas characteristic adsorption conditions.

4. Project status
   Items 2.1, 2.2 and 2.5 are completed and need only a periodical up to dating.
   Items 2.3, 2.4 are in development and will be completed in the frame of a study concerning the philosophy of Kr^{85} management and will be completed in the 1980.
   Items 2.6 is at the starting point and will be developed in the next year.

5. Next steps: with reference to the objectives listed in point 2.

6. Relation to other projects and codes
   "Testing of the filter systems used in nuclear plants for particle and iodine removal"

7. Reference documents

1- CURZIO G. GENTILI A.
"Ritenzione di gas nobili su letti di carbone attivo"
Atti del XVI Congresso Nazionale dell'A.I.F.S.P.R., Firenze, Settembre 1970. Firenze 1971.

2- CURZIO G., GENTILI A., MAINARDI C., PELLUNGRINI P.
"Ritenzione dei gas nobili radioattivi prodotti per fissione negli impianti nucleari"
Tip. Edit. Pisana, Pisa 1972.

3- CURZIO G., GENTILI A.
"Determazione 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 à 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.

TITLE (ENGLISH LANGUAGE): Fission produced radioactive noble gases treatment. -3-	CLASSIFICATION: 5.3

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"
La Termotecnica, Vol. XXXI, 3 (1977).

11- CASTELLANI F., CURZIO G., GENTILI A.
"Facilities for Conditioning 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"
Istituto di Impianti Nucleari, Università di Pisa, 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"
Istituto di Impianti Nucleari, Università di Pisa, RP 251(76).

14- CASTELLANI F., CURZIO G., GENTILI A.
"Analisi comparativa tra i sistemi di trattamento dei gas nobili radioattivi prodotti negli impianti nucleari"
Istituto di Impianti Nucleari, Università di Pisa, RP 280(77).

15- CASTELLANI F., CURZIO G., GENTILI A.
"Stoccaggio di gas nobili radioattivi in bombole o contenenti carbone attivo: valutazione dei costi"
Istituto di Impianti Nucleari, Università di Pisa, RP 283(77).

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Fission produced radioactive noble gases treatment. -4-	5.3

16- 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"
Istituto di Impianti Nucleari, Università di Pisa, RP 284(77).

17- CASTELLANI F., CURZIO G., GENTILI A.
"Collaudo del sistema di ritenzione dei gas nobili radioattivi della centrale ENEL-IV di Caorso"
Istituto di Impianti Nucleari, Università di Pisa, RP 285(77).

18- CASTELLANI F., CURZIO G., GENTILI A.
"Metodi di controllo dell'efficienza di un sistema a fuoriuscita ritardata dei gas nobili radioattivi prodotti per fissione"
Istituto di Impianti Nucleari, Università di Pisa, RP 295(77).

19- CASTELLANI F., CURZIO G., GENTILI A.
"Caratterizzazione di materiali adsorbenti di gas nobili radioattivi"
Istituto di Impianti Nucleari, Università di Pisa, RP 310(78), RP 315(78), RP 334(78).

20- CASTELLANI F., CURZIO G., GENTILI A.
"Le piégeage du Kr^{85} rejeté par les usines de retraitement de combustible"
Seminar on Radioactive Effluents from Nuclear Fuel Reprocessing Plants, Karlsruhe 22-25 nov. 1977.

21- CASTELLANI F., CURZIO G., GENTILI A.
"Caorso off-gas system: acceptance test of the noble gases delay charcoal beds"
5th Technical Meeting, Nuclex 78, Basel 3-7 ottobre 1978.

22- CASTELLANI F., CURZIO G., GENTILI A.
"Caratterizzazione del rilascio dei radionuclidi in fase gassosa nel corso delle operazioni di taglio e di dissoluzione presso l'impianto pilota Eurex di elementi di combustibile irraggiati in reattore del tipo CANDU"
Istituto di Impianti Nucleari, Università di Pisa, RL 332(79).

8. Degree of availability
All the papers and reports listed above are free; availability of reports related to the work in progress is subjected to the authorization of the sponsoring organizations.

(Istituto Impianti Nucleari, Università, V. Diotisalvi 2, I-56100 Pisa)

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Testing of the filter systems used in nuclear plants for particle and iodine removal -2-	5.3

granular beds so that H.T. and P. tests can be easily run;
- to perform efficiency tests on commercial HEPA filters with monodisperse aerosol to compare their results with polidisperse aerosols ones:
  . first step: single filters
  . further steps: cascade
- to set up the apparatus related to the tests mentioned in point 2.d

4) Project status

All the items of the program are in progress; it can be noticed that a monodisperse aerosol generation device was built and calibrated (diameter range 0.2÷1,1μm; nuclei concentration $\simeq 10^6$n/cm^3; outlet air flow-rate $\simeq$200 l/h).
So some preliminary tests on HEPA filters were run

5) Next steps

To test HEPA filters under heavy environmental conditions.

6) Relation to other projects
"Fission Produced radioactive noble gases treatment".

7) Reference documents

1. LANZA S.,MAZZINI M. et alii.
   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.
2. 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 Nov. 1976.
3. 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 Nov. 1976.
4. 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 Ott.1976.
5. BENIGNI E., LANZA S., MAZZINI M.
   Messa a punto di un circuito di prova di assorbitori dello iodio
   FI-KW-0002 - Pisa 1976.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Testing of the filter systems used in nu clear plants for particle and iodine re- moval -3-	5.3

6. ISTITUTO IMPIANTI NUCLEARI
   Descrizione del circuito prova filtri per particelle
   FP-KW-0006 - Pisa 1978

7. CUCCURU A., LANZA S.
   Generazione di vapori di F-112 mediante gorgogliamento di aria in soluzioni azeotropiche. Parte I: scelta del solvente
   ISTITUTO IMPIANTI NUCLEARI - FI-IW-0004 - Pisa 1977.

8. BENIGNI E., MAZZINI M.
   Dispositivo di generazione di vapori di Freon 112
   ISTITUTO IMPIANTI NUCLEARI - FI-KW-0003 - Pisa 1977.

9. CUCCURU A., LANZA S., MAZZINI M.
   Generazione di vapori di F-112 mediante gorgogliamento di aria in soluzioni azeotropiche. Parte II: taratura del generatore
   ISTITUTO IMPIANTI NUCLEARI - FI-IW-0005 - Pisa 1977.

8) Degree of availability

The references 1 ÷ 4 are free: the next ones are restricted and may be available with the authorization of the CNEN.

(Istituto Impianti Nucleari, Università, V. Diotisalvi 2, I-56100 Pisa)

	CLASSIFICATION: 5.3
TITLE (ORIGINAL LANGUAGE): Tecniche per la misura della capacità di ritenzione di sistemi di filtrazione per iodio e derivati iodoorganici	COUNTRY: ITALY SPONSOR: ENEL
TITLE (ENGLISH LANGUAGE): Techniques for Testing Charcoal Absorbers for Iodine and its Derivatives	ORGANISATION: Polytechnic Institute of Milan PROJECT LEADER: G. SANDRELLI
INITIATED: 1970 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: June 1979	

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.

(G. SANDRELLI, ENEL- CRTN, Bastioni Porta Volta 10, I-20121 Milano)

	CLASSIFICATION: 5.3
TITLE (ORIGINAL LANGUAGE): Decontamination (C 2.1)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Winfrith and Harwell)
	PROJECT LEADER: D J Ferrett (Winfrith) J B Sayers (Harwell)
INITIATED: March 1978 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: March 1979	

Background

Surfaces and components exposed to PWR coolants pick up activity during operation. For inspection, maintenance and access purposes it will be necessary to reduce the level of surface-held activity. This can be carried out by chemical or physical processes. Optimisation related to PWR conditions is required.

Objectives

1. To review PWR decontamination experience and methodology.

2. To undertake experimental studies to assess the efficacy of circuit and component surface decontamination with full strength and regenerable reagents.

3. To assess the corrosion effects of chemical reagents on PWR materials and components.

4. To decontaminate the DIDO Water Loop (DWL) using preferred process.

Programme

Laboratory experimental studies at Winfrith.

Decontamination of DWL during 1979.

Facilities

Laboratory rigs at AEE Winfrith.

DIDO Water Loop at AERE Harwell.

[illegible]

	CLASSIFICATION: 5.3
TITLE (ORIGINAL LANGUAGE): GAS PHASE TRAPPING STUDIES	COUNTRY: UK
	SPONSOR: UKAEA
TITLE (ENGLISH LANGUAGE):	ORGANISATION: WNL Windscale
	PROJECT LEADER: J S Hillary
INITIATED: 1978 / COMPLETED:	SCIENTISTS:
STATUS: / LAST UPDATING: May 1979	

DESCRIPTION

1. GENERAL AIM: Improvement and standardisation of aerosol trapping, with particular reference to normal emissions from reactors.

2. PARTICULAR OBJECTIVES: Study trapping of methyl iodide on charcoals including effects of carrier gas face velocity and composition, and charcoal granule size and bed geometry.

3. EXPERIMENTAL FACILITIES AND PROGRAMME: Useful preliminary results are being obtained from scoping tests.

	CLASSIFICATION: 5.3
TITLE (ORIGINAL LANGUAGE): GAS PHASE TRAPPING STUDIES	COUNTRY:
	SPONSOR: UK
TITLE (ENGLISH LANGUAGE):	ORGANISATION: WNL Windscale
	PROJECT LEADER: J J Hillary
INITIATED: 1972 / COMPLETED:	SCIENTISTS:
STATUS: / LAST UPDATING: May 1979	

DESCRIPTION:

1. General Aim

Improvement and standardisation of aerosol trappings, 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

Methyl iodide trapping data have been tabulated on commercially produced batches of British impregnated charcoals: some variability in performance together with a tendency to lower values of trapping index with later date of manufacture was noted. Results on small batches (Lab prepared) confirmed a strong effect of particle size. Smaller granules giving improved performance. The data on commercial material are presently being analysed statistically including comparison with manufacturer's data on process parameter measurements.

5. Reference Documents

UKAEA document TRG Report 2906(W) 'The Performance of Commercially-Prepared Impregnated Charcoals for the Trapping of Methyl Iodide', J J Hillary and L R Taylor (November 1977).

	CLASSIFICATION: 5.3
TITLE (ORIGINAL LANGUAGE): GAS PHASE TRAPPING STUDIES	COUNTRY:
	SPONSOR: UK
TITLE (ENGLISH LANGUAGE):	ORGANISATION: WNL WINDSCALE
	PROJECT LEADER: J.J HILLARY
INITIATED: 1972 / COMPLETED:	SCIENTISTS:
STATUS: / LAST UPDATING: MAY 1979	

DESCRIPTION:

1. General Aim

Improvement and stadardisation 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 has been set up for controlled static ageing tests on a number of charcoal samples in a variety of storage gas conditions.

4. Project Status

Apparatus has been commissioned and experiments have been under way for over twelve months, useful preliminary results have been obtained.

	CLASSIFICATION: 5.3
TITLE (ORIGINAL LANGUAGE): In-pile studies of PWR primary circuit chemistry (C 3.1)	COUNTRY: UNITED KINGDOM SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Harwell) PROJECT LEADER: D J Ferrett (Winfrith) J B Sayers (Harwell)
INITIATED: March 1978 | COMPLETED:	SCIENTISTS:
STATUS: In progress | LAST UPDATING: March 1979	

Background

If man-rem totals incurred during PWR operation, refuelling and maintenance are to be reduced, a better understanding of factors influencing activity build-up is required from studies of PWR primary circuit chemistry.

Objectives

To investigate activated corrosion product behaviour, transport and deposition phenomena under in-pile PWR primary circuit conditions.

Programme

When the present loop refurbishing has been carried out a by-pass section will be designed to accommodate on-line filters (magnetic filters), and a more realistic Inconel source, in the form of a heat exchanger, will be provided. A removable pipe section will be substituted for some of the deposition coupons.

The initial experimental programme in the DIDO Water Loop is expected to commence at the end of 1979, following a period of out-of-reactor recommissioning, during which the loop modifications will be tested and base-line data on the chemical and radiochemical characteristics of circulating corrosion products will be obtained. The experimental programme will be devoted to mechanistic studies of corrosion product behaviour, aimed at improving understanding of the man-rem problem.

The first phase of experimental work will be devoted to the determination of the first characteristics of circulating corrosion products and deposited corrosion products (concentrations of soluble and insoluble material, particle sizes, and distribution of active species under typical PWR conditions). Deposition coupons will be used for the examination of deposited corrosion products, and some of these coupons will be available for decontamination studies.

The second phase of experimentation will attempt to determine the effect of variation of water chemistry control on the behaviour of corrosion products. Parameters to be varied are lithium hydroxide and boric acid concentration, and the fraction of flow taken through the purification circuit.

C 3.1 contd

Facilities

DIDO Water Loop at Harwell.

Reference Documents

Berichtszeitraum/Period 1. 1. 78 - 31. 12. 78	Klassifikation/Classification 5.4	Kennzeichen/Project Number RS 239
Vorhaben/Project Title Aktivitätsüberwachung der Dosisbelastung in der Umgebung von Kernenergieanlagen Dose Rate Proportional Measurement of the Radioactive Gaseous Effluents from Nuclear Power Plants.		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 131, Frankfurt
Arbeitsbeginn/Initiated 1. 10. 76	Arbeitsende/Completed 30. 6. 79	Leiter des Vorhabens/Project Leader Dr. Grosse-Schulte
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 78	Bewilligte Mittel/Funds

1. General Aim

In order to facilitate more precise evaluations of the environmental dose rate in the vicinity of an nuclear facility, the methods of measuring radioactive noble gas discharge rates must be improved: A system capable of automatically analysing the type of nuclide and the quantities released will be coupled to a system yielding the meteorological distribution factors. These two sets of data will be used for the on-line calculation of the actual environmental burden. The on-line measurement and analysis of the airborne particulate will be included at a later point of time.

2. Particular Objectives

- improvement on on-line nuclide-identification systems for stack surveillance (noble gases, aerosols)
- development of computer programs for real-time (or at least at short intervalls) calculation of the actual environmental burden
- development and improvement of the on-line calculation of distribution parameters from metereological factors (wind characteristics, temp. profile etc.)
- experimental verification of the means and methods conceived (prototype system at a nuclear power plant).

3. Research Program

3.1 Nuclide identification (& release rate measurement) programs - establish a test routines

3.2 Meteorological Data

3.3 Establish programs for the computation of environmental dose rates on the basis of data received from the nuclide identification monitor and from the meteorological instrumentation.

3.4 Investigate the boundary conditions given by licensing requirements and plant conditions

3.5 Investigate the possibility of interconnection the effluent monitors from several plants at one site

3.6 Test the programs and systems by means of setting up one prototype arrangement at a plant

3.7 Recalibrate the prototype system by means of test measurements (actual or mock releases from the stack).

4. Test Facilities

For Biblis A, a measurement device has been provided for specific nuclide-related monitoring or radioactive noble gases. The apparatus has been tested in Gundremmingen for approx. 4 months and has been operating in Biblis since Jan. 1976.

5. Progress to Date

To 3.1 The Aerosol measurement device with a GeLi-detector was thoroughly tested. It was coupled to equipment borrowed from the BGA. An evaluation program on the preliminary analysis of the acquired data will be adapted to Aerosol-monitor requirements.

After start-up and calibration of the Aerosol-monitor with a radiation source, the monitor was tested in the FRN reactor in München-Neuherberg.
With the participation of Mr. Wilhelm, KFK and Messrs. Herfurth an objective was established for further development of Iodine monitor in regard to an improved discrimination of the Iodine isotope J^{131}.

To 3.2 Investigation of the boundary conditions associated with the meteorological calculations were made.

To 3.3 In discussions with KfK, it was attempted to establish the suitability of the programs proposed by Karlsruhe for the on-line-evaluation of propagation factors.

The computer program UBANO is a method for the determination of dose with long-time period factors. Because the necessary computer times for an on-line program are too long, a two-step method is being prepared with the use of UBANO. A data set will be determined from which, through interpolation in a second procedures, the appropriate actual values of the dose will be calculated.

To 3.7 The device for nuclide-specific acquisition of Aerosol releases was completed and shipped to BGA/Neuherberg (Mü) for first tests.

6. Results

To 3.1 In the first examinations the GeLi-detector of the Aerosol measurement device showed a severe microphony. The horizontally placed detector converted acoustic signals (sound vibrations) into electrical impulses, which could lead to interference with the measurement results. Based on these results, the detector was returned to the manufacturer where it was placed in a different position. The device is not in a proper condition and ready to be used.
First spectra have been developed with the Aerosol-monitor.

To 3.3 Current computer methods for evaluation of the dose corresponding to gamma-radiation of the inert gas release to the atmosphere are designed for the calculation of plotted point and/or to allow for the determination of point of maximum dose. Under the scope of the above R+D project - starting with the stack - a net of plotted points is chosen, for which the momentary dose is determined and

stored for simulation.
The large number of points is not compatible with a computer time of approx. 10 minutes per point. Therefore it was agreed to prepare normalized results in matrix form and to determine the actual values by an interpolation method. This accelerated method appears to make feasible the use of a net of plotted points.

The KfK programs are still not being used in free (uncorrected) on-line-operation. The complementary evaluation of the results by hand limits the use in automated facilities.
The mechanism of the Aerosol-measurement devices for individual nuclide determination was assembled.

To 3.7 The mechanical and electrical tests of the on-line filter device and the mechanical adjustment of the GeLi-detector were successfully completed with a horizontally located cooling finger.

7. Next Steps

To 3.1 The Aerosol Monitor is supposed to be tested in a power reactor.

To 3.2 The precision requirements for the instrumentation of the weather mast are to be discussed with external officials (weather bureau?).

To 3.3 Modification of the UBANO program to be continued. Subsequently comparison with other programs will be performed.

To 3.7 The Aerosol monitor will be tested and calibrated with aerosols and specimens in cooperation with the BGA/ Institute for Radiation Hygiene.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.1978	5.4	06.03.01/01A (PNS 4811)

Vorhaben/Project Title	
Untersuchung des physikalischen und chemischen Verhaltens biologisch besonders wirksamer Radionuklide in der Umwelt Investigation of the Physical and Chemical Environmental Behaviour of Radionuclides Characterized by a Particular Biological Effectiveness	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Kernforschungszentrum Karlsruhe Projekt Nukleare Sicherheit AS

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.1974	31.12.78	DI H. Schüttelkopf
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	31.12.1978	

1. General Aim

Determination of the longterm exposure of the environment of the reprocessing plant by longlived ^{129}I.

2. Particular Objectives

The behaviour of ^{129}I in the environment of the Karlsruhe reprocessing plant.

3. Research Program

3.1 Development of analytical methods for the determination of ^{129}I.

3.2 Measurement of the concentration distribution of ^{129}I in the Karlsruhe reprocessing plant and of the ^{129}I release from the plant.

3.3 Measurement of the ^{129}I in the environment in the Karlsruhe reprocessing plant.

3.4 Measurement of the stable iodine in the environment.

4. Experimental Facilities, Computer Programs

The measurement of ^{129}I and ^{127}I is performed using neutronactivation; for this purpose the irradiation facilities of the Karlsruhe research reactor FR2 are applied.

5. Progress to date

3.3: In 1978 milk samples were analyzed which had been taken in the environment of WAK.

3.4: $CH_3{}^{127}J$ in the environmental air were measured in Karlsruhe.

6. Results

3.1: Very sensitive analytical methods for ^{129}I were developed for different sample materials.

3.2: The behaviour of ^{129}I in the Karlsruhe reprocessing plant and its release was completely investigated.

3.3: A longterm risk for the population caused by ^{129}I is not expected. An existing environmental contamination with ^{129}I is reduced by a half-life of 0,3 y. The exposure of the population via the pasture-cow-milk path until now was overestimated at least by a factor 45.

3.4: In the environmental air of Kiel, Stade, Karlsruhe, Gundremmingen and Munich elemental iodine and iodine aerosols were measured. In the environmental air of Karlsruhe $CH_3{}^{127}I$ was determined.

7. Next steps

The research program is finalized.

8. Relations with other projects

9. References

Results were presented at the 1978 PNS-Colloquium. They will be published during 1979.

10. Degree of Availablity of the Reports

Unrestricted distribution.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.7.1978 - 31.12.1978	5.4	06.03.01/02 (PNS 4812)

Vorhaben/Project Title	
Untersuchung des physikalischen und chemischen Verhaltens biologisch besonders wirksamer Radionuklide in der Umwelt - Pu, Am, Cm. Investigation of the Physical and Chemical Environmental Behaviour of Radionuclides Characterized by a Particular Biological Effectiveness - Pu, Am, Cm.	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Kernforschungszentrum Karlsruhe Projekt Nukleare Sicherheit / AS

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.7.1978		DI H. Schüttelkopf

Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	31.12.1978	

1. General Aim

Determination of the longterm exposure of the environmental population after an accidental release of Pu, Am und Cm.

2. Particular Objectives

The behaviour of Pu, Am und Cm in the environment of the Karlsruhe reprocessing plant.

3. Research Program

3.1 Development of analytical methods for the determination of Pu, Am and Cm.

3.2 Determination of Pu, Am and Cm releases from the Karlsruhe reprocessing plant.

3.3 Concentrations of actinides in the environment of the Karlsruhe reprocessing plant.

3.4 Determination of transfer processes, especially soil /plant and plant/animal.

3.5 Experiments to increase the mobility of actinides in soil.

3.6 Field experiments to increase the mobility of actinides in soil.

4. Experimental Facilities, Computer Programs

5. Progress to date

3.1: A very sensitive analytical method for the determination of Pu was developed. First experiments to achieve automatical separation of Pu, Am and Cm by means of HPLC were performed.

3.2: The releases of plutonium with gaseous and liquid effluents of the Karlsruhe reprocessing plant were measured monthly.

3.3: The determination of plutonium in numerous environmental samples was performed.

6. Results

3.1: Using the developed analytical method for Pu, a technician is able to perform 4 analyses per day. Chemical yield: 70 - 80 %. Lower detection limit: <0.1 fCi/g.

3.2: The monthly determined releases of plutonium in the gaseous effluents of Karlsruhe reprocessing plant range over several orders of magnitude. The released Pu-238 and Pu-239+240 remains below the releases of gross alpha activity. The plutonium concentrations in liquid effluents range in the average between 0,1 - 1 pCi Pu/l.

3.3: Concentrations of Pu were measured in: soil, plants, sediments, air, animals and water.

7. Next steps

3.1 - 3.3 will be continued 1979.

8. Relations with other projects

-

9. References

-

10. Degree of Availablity of the Reports

Unrestricted distribution.

Berichtszeitraum/Period 1.1. - 31.12.1978	Klassifikation/Classification 5.4	Kennzeichen/Project Number 06.03.02/01A (PNS 4821)
Vorhaben/Project Title Analytical and Experimental Investigation on the Atmospheric Dispersion of Radioactive Gases in the Near Distance Region (up to 15 km) at Source Heights < 100 m Theoretische und experimentelle Untersuchung der Ausbreitung radioaktiver Gase im lokalen Bereich (bis 15 km Entfernung) bei Emissionshöhen ≤ 100 m		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe GmbH Projekt Nucleare Sicherheit AS
Arbeitsbeginn/Initiated 1972	Arbeitsende/Completed 31.12.1978	Leiter des Vorhabens/Project Leader W.Hübschmann,H.Schüttelkopf
Stand der Arbeiten/Status completed	Berichtsdatum/Last Updating Dec. 1978	Bewilligte Mittel/Funds

1. General Aim

Improvement of the knowledge about the atmospheric dispersion of radioactive substances in the micro-scale (up to 15 km distance) at source heights up to 100 m.

2. Particular Objectives

Assessment of horizontal and vertical diffusion parameters according to the diffusion category and the source height and of the influence of surface roughness on these parameters.

3. Research Program

3.1 Tracer diffusion experiments at the different stability categories; chemical tracer gases are emitted at heights up to 100 m from the meteorologic tower.

4. Experimental Facilities

Sampling stations are operated automatically to collect air-tracer samples. The tracer is evaporated at the meteorological tower. The tracer concentrations are measured by a gas-chromatograph.

5. Progress to Date

The tracer experiment series at normal diffusion situations is completed. The diffusion parameters evaluated from the single experiments have been combined into a parameter set for emission heights up to 130 m. The curve families have been smoothened and centered in order to refer to the most frequent turbulence intensity in each diffusion category.

6. Results

The final diffusion parameter set is completed. A final comprehensive report on the detailed experimental results is being prepared.

7. Next Steps

The project is completed.

8. Relation with Other Projects: PNS 4822, PNS 4823

9. References

K. Nester, P. Thomas;
Vergleich und Zusammenfassung der in Jülich und Karlsruhe ermittelten Ausbreitungsparameter. Staub, Reinhaltung der Luft (in Vorbereitung)

10. Availability of the Reports

Reports are available through Kernforschungszentrum Karlsruhe GmbH, Karlsruhe, Zentralbücherei

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1 - 31.12.1978	5.4	06.03.02/03A (PNS 4823)
Vorhaben/Project Title Investigation on the Atmospheric Dispersion of Radioactive Gases in the Near Distance Region (up to 15 km), at Source Heights above 100 m Untersuchung der Ausbreitung radioaktiver Gase im lokalen Bereich (≤ 15 km Entfernung), Emissionshöhe über 100 m		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe GmbH Projekt Nucleare Sicherheit AS
Arbeitsbeginn/Initiated 1978	Arbeitsende/Completed 1980	Leiter des Vorhabens/Project Leader W.Hübschmann,H.Schüttelkopf
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Improvement of the knowledge about the atmospheric dispersion of radioactive substances in the micro-scale (up to 15 km distance) at source heights above 100 m.

2. Particular Objectives

Short range atmospheric diffusion, diffusion models for accidental releases, influence of surface roughness.

3. Research Program

3.1 Tracer diffusion experiments at the different stability categories; chemical tracer gases are emitted at 160 m and 195 m above ground.

3.2 Turbulence studies over areas of different roughness.

4. Experimental Facilities

4.1 A 200 m high meteorological tower is operated in the Karlsruhe Nuclear Research Center in order to collect comprehensive meteorological information in the lower atmospheric layer. Wind velocity and direction profiles as well as dry and dew point temperature profiles are measured across the tower height.

4.2 Sampling stations are operated automatically to collect air-tracer samples. The tracer is evaporated at the meteorological tower. Tracer concentrations are measured by a gas-chromatograph.

4.3 Field measurements are performed over areas of different surface structure, using a 15 m mast.

5. Progress to Date

9 diffusion experiments have been performed in 1978. The chemical tracers have been emitted at heights of 160 m and 195 m.

A vector vane is operated at a 15 m high mast in an area of small surface roughness (grade II). The measured turbulence parameters are compared to those generated in the KfK (roughness grade III).

-2-

6. Results

On the basis of the tracer experiments a preliminary curve family of the diffusion parameters σ_y and σ_z for emission heights > 130 m has been compiled. Further experiments are necessary to confirm this curve family.

7. Next Steps

The tracer experiments and the field measurements are continued in 1979.

8. Relation with Other Projects

PNS 4820 and PNS 4822

9. References

H. Kiefer, W. Koelzer;
Jahresbericht der Abteilung Sicherheit 1978, KfK 2775

10. Availability of the Reports

Reports are available through Kernforschungszentrum Karlsruhe GmbH, Karlsruhe, Zentralbücherei

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	5.4	06.03.02/02A (PNS 4822)

Vorhaben/Project Title	
Investigation on the Atmospheric Dispersion of Radioactive Gases **in the Mesoscale (more than 15 km distance)** Untersuchung der Ausbreitung radioaktiver Gase im regionalen Bereich (>15 km Entfernung)	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Kernforschungszentrum Karlsruhe GmbH Projekt Nucleare Sicherheit AS

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1978	1981	W.Hübschmann, H.Schüttelkopf
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

Improvement of the knowledge about the atmospheric dispersion of radioactive substances in the meso-scale (> 15 km distance).

2. Particular Objectives

Assessment of medium and long range atmospheric transport and diffusion models and parameters.

3. Research Program

Tetroons are started and tracked by radar.

4. Experimental Facilities

Tetroons are Helium-filled balloons of constant volume. The tetroons will be equipped by transponders in order to improve the trackability.

5. Progress to Date

A series of tetroon flights has been performed in the Rhine Valley close to the KfK. These flights showed the limitations of radar tracking without transponder. It is planned, therefore, to equip the tetroons by a transponder (light-weight receiver-sender unit). A small series of transponders is developed and tested in a stationary arrangement.

6. Results

The tetroon tracks have been evaluated in respect to atmospheric parameters. It is shown that the time constant of the vertical oscillations as well as the horizontal diffusion parameter σ_y, derived from several consecutive tetroon flights, fall into the expected range.

7. Next Steps

Transponders will be tested further and go into series production. Tetroon flights will be continued in the Rhine Valley.

8. Relation with Other Projects

PNS 4820, PNS 4823

9. References

H. Kiefer, W. Koelzer;

Jahresbericht der Abteilung Sicherheit 1978, KfK 2775

10. Availability of the Reports

Reports are available through Kernforschungszentrum Karlsruhe GmbH, Karlsruhe, Zentralbücherei

Berichtszeitraum/Period 1.1.78-31.12.78	Klassifikation/Classification 5.4	Kennzeichen/Project Number 06.02.01/01A (PNS 4511)
Vorhaben/Project Title Störfallablaufanalyse für die große Wiederaufarbeitungsanlage (Explosion) Incident Analysis for the Large Reprocessing Plant (Explosion)		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Projekt Nukleare Sicherheit Kernforschungszentrum Karlsruhe / IRE
Arbeitsbeginn/Initiated 1978	Arbeitsende/Completed 1981	Leiter des Vorhabens/Project Leader Dr.Caldarola/Dr. Schleisiek
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating Dec. 1978	Bewilligte Mittel/Funds

1. General Aim

Investigation of the safety and environmental protection of reprocessing plants.

2. Particular Objectives

Incident analysis for the large reprocessing plant (explosions)

3. Research Program

3.1 Preliminary analysis
3.2 Incident identification
3.3 Quantitative analysis of the development of explosive processes including the emissions from the buildings of the plant

4. Experimental Facilities, Computer Codes

3.1 -
3.2 Modification and application of computer codes for fault tree analysis
3.3 Application of computer codes developped in the framework of PNS 4125, if any.

5. Past Work

3.1 To identify the materials and conditions which may lead to explosive processes in reprocessing plants the relevant theories have been studied further.
3.2 A new analytical computer code to analyse systems made of multistate components and containing components which are statistically dependent has been established and tested to a degree allowing the successfull treatment of some simple examples.

6. Results

3.1 Concerning explosions in reprocessing plants it was stated that explosive mixtures are especially formed by hydrogen and paraffinic hydrocarbons. The compilation of the properties of these materials relevant for explosions has been started.
3.2 The new computer code for fault tree analysis was applied to the following problems:

1. Comparison of three different fault trees for a system having been established by different authors. The result was that the three fault trees are equivalent to the same Boolean structure function.
2. Evaluation of a system with statistically dependent components which are usually met in practice. The first is common mode failure. The second is the case of components characterized by a failure rate whose value depends

upon the occurrence of particular events. The third is the case of two components in which the repair of one effects the operation of the other.

7. Next Step

The compilation of characteristic data of explosive processes will be enlarged to enable applications to reprocessing plants.
The testing of the first version of the new computer code for fault tree analysis will be completed. A second version will be established to calculate the failure intensity of a system of multistate components. The methods for computerized fault tree construction will be further investigated.

8. Relations to other Projects

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9. References

L. Caldarola: Fault Tree Analysis with Multistate Systems with Multistate Components. ANS Topical Meeting and Probabilistic Analysis of Nucl. Reactor Safety, Los Angeles (Calif.), May 8-10, 1978.

L. Caldarola, H. Wenzelburger, A. Wickenhäuser, P. Schwab: Störfallablaufanalyse für die große Wiederaufarbeitungsanlage (Explosionen). In PNS-Halbjahresbericht 1978/1.
KFK-2700 S. 4500/1-4500/2.

10. Availability of the References

Literaturabteilung der Kernforschungszentrum Karlsruhe GmbH

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	5.4	06.02.01/02A (PNS 4513)
Vorhaben/Project Title Incident Analysis for the Large Reprocessing Plant (Extraction) Störfallablaufanalyse für die große Wiederaufarbeitungsanlage (Extraktion)		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KfK/PNS IDT
Arbeitsbeginn/Initiated 1.1.1978	Arbeitsende/Completed 1981	Leiter des Vorhabens/Project Leader Dr. R. Avenhaus
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31.12.1978	Bewilligte Mittel/Funds

1. General Aim

For the case of the WAK selected incidents in the extraction area are analyzed quantitatively which may lead to mechanical-hydraulic plant operations disturbances and/or fires.

2. Special Aim

The Applicability of Monte-Carlo-methods to failure tree analysis is analyzed. Cause consequence analyses of important incidents in the extraction area are performed. Event trees are established as a first step towards probabilistic analyses.

3. Research Program

3.1 Methodological investigations with respect to the applicability of Monte-Carlo-methods to failure tree analysis are completed.

3.2 Cause consequence analyses of important incidents in the extraction area are performed, and possible consequences for the whole system are estimated.

3.3 Event trees for possibly important incidents are established.

3.4 Probabilistic failure tree analyses will be performed depending on available component data.

4. Computer codes

Code for the evaluation of failure trees with S_k-systems.

5. Performed work

In 1978 task 3.1 was completed in a preliminary way: Theoretical considerations for shortening the computer time with the help of importance sampling were completed for a concrete model (periodically inspected parallel system). Using these results a code was developed with the help of which one can evaluate failure trees with periodically inspected components. The efficiency of this code was verified with the help of selected failure trees with differing structures.

Furthermore, task 3.2 was started in collaboration with WAK representatives: ignition sources, ignitable substances, possible fires and the possible propagation of these fires within and beyond the cell were analyzed. A detailed analyses of a special incident was started.

6. Results

With the help of a concrete example it was demonstrated in which way the importance sampling method has to be applied and how effectively one can shorten the computer time. In view of the large computer time needed in case of big systems, the value of this method is given by the fact that in especially important cases one can evaluate the failure probability of subsystems independently of analytical methods based on approximations.

Concrete results with respect to task 3.2 were not yet obtained.

7. Future work

With the completion of three reports the work under task 3.1 will be preliminarily finished; in the future the application of the methods and codes will be emphasized.

With respect to task 3.2 failure trees and event trees will be established for the special incident under consideration; the possibility of the propagation of kerosine fires within and beyond the original cell will be analyzed quantitatively.

8. Relation to other work

The computer time, which was needed for the evaluation of a specific example with the help of the Monte-Carlo-method, was compared with that time, which was needed for the evaluation of the same example with the help of an analytical method.

9. Literature

H. Wenzelburger, H. Frick
Failure Probability of a Periodically Inspected Parallel System
Fall 1978 (unpublished)

H. Wenzelburger
Abschätzung kleiner Ausfall-Wahrscheinlichkeiten von Systemen mit Importance Sampling
Fall 1978 (unpublished)

H. Wenzelburger
Optimales Importance Sampling für die Ausfall-Wahrscheinlichkeit von periodisch inspizierbaren Parallelsystemen
Fall 1978 (unpublished)

Ch. Schneider
Fehlerbaumanalyse von periodisch inspizierbaren Systemen mit Hilfe von Monte-Carlo-Methoden
Dissertation Universität Karlsruhe 1978, KfK-Report 2628

Ch. Schneider
Programmbeschreibung für die Programm BAUM und MINCUT
March 1978 (unpublished)

Ch. Schneider
Rechenzeitvergleich zwischen analytischen und Monte-Carlo-Programmen am Beispiel des Fehlerbaumes der Aufgabenstellung vom Wettrechnen
March 1978 (unpublished)

L. Caldarola, A. Wickenhäuser
The Karlsruhe Computer Program for the Evaluation of the Availability and Reliability of Complex Repairable Systems
Nuclear Engineering and Design, 43 (1977), p. 463-470

10. Availability of Reports

Literaturabteilung Kernforschungszentrum Karlsruhe GmbH.

Unrestricted.

Classification: 5.4

Title: Konsekvenser af frigørelser af fissions produkter til atmosfæren	Country: DENMARK Sponsor: Risø National Lab.
Title Consequences of Releases of Fission Products to the Atmosphere	Organization: Risø National Laboratory
Initiated date: 1972 Completed date: Status: Progressing Last updating: May 1979	Scientists: O. Walmod-Larsen S. Thykier-Nielsen P. Hedemann Jensen

1. General aim

Calculation of consequences of releases of fission products to the atmosphere under various environmental conditions.

2. Particular objectives

Development of models for calculation of

a. Doses to individuals:
External gamma doses from airborne radioactive material.
Internal doses due to inhalation of radioactive material.
External gamma doses from radioactive material deposited on the ground.
Beta doses to the skin from airborne radioactive material and from radioactive material deposited on the ground.

b. Concequences of doses based on given dose-consequence relations.

c. Doses to individuals and population under specified meteorological conditions.

d. Probability distribution of doses to individuals and population for a given probability distribution of meteorological parameters.

e. Isodose curves: Shape, area and number of people receiving doses witin specific limits.

f. Number of concequences (i.e. number of people having e.g. early illness, cancer) for a given release.

Both normal and accidential releases are considered.
Furthermore the parameters in the models are studies:
Duration of release, atmospheric stability, plume rise, ground roughness etc.

3. Experimental facility and programme

None.

4. Project status

A computer model, PLUCON2, based on the gaussian plume model and fulfilling the objectives a.-f. mentioned in section 2, has been developed. PLUCON2 can be used for calculation of doses and consequences in near-zone, i, e. the area witin 50 km from the release point.
A limited comparison between PLUCON2 and the models used in the other Nordic countries, Finland, Norway and Sweden has been made. On the basis of the calculation results from the models, it was concluded that there are no essential differences between the Nordic dose models. The results of the comparison is published in reference 1.
PLUCON2 has been used for a calculation of doses from hypotetical accidents at a nuclear power plant [2].

5. Next steps

Development of a model, PUFCON, for calculation of consequences of accidental releases in situations where the meteorological conditions varies with time. This model is based on a puff-model which is under development in the micro-meteorological group at Risø.
It is expected that this model is completed in 1982.

6. References

1. Comparison of Nordic Dose Models, S. Thykier-Nielsen, Risø-M-1972.

2. Calculation of the Individual and Population Doses on Danisk Territory Resulting from Hypothetical Core-melt Accidents at the Barsebäck Reactor, P. Hedemann Jensen, E. Lundtang Petersen, S. Thykier-Nielsen and F. Heikel Vinther, Risø Report No. 356.

3. Modeller til beregning af eksterne gammadoser og inhalationsdoser fra frigørelser til atmosfæren af radioaktive stoffer, S. Thykier-Nielsen, Risø-M-1725.
   (Description of the models for calculation of external gammadoses and inhalation doses).

4. Sammenligning af matematiske modeller til beregning af eksterne gammadoser hidrørende fra radioaktivitetsfrigørelser til atmosfæren, P. Hedemann Jensen, Risø-M-1726.
   (Comparison of different models for calculation of external gammadoses from a plume).

7. Degree of availability

Available on an exchange basis.
The computer programmes are written in Burroughs ALGOL for a B6700-computer.

148-2 - 01 158-2 - 01 / 4110-10		5-4
TITRE Modélisation des transferts de radioactivité **à l'intérieur d'une installation nucléaire,** de la dispersion atmosphérique des rejets et des calculs de dose : code ALICE		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) Modelization of radioactivity transfers within a nuclear plant, atmospheric dispersion of releases and doses calculations : ALICE computer code.		Organisme Exécuteur CEA/DSN/SESRS Responsable
Date de démarrage 01/09/75	Etat actuel en cours	Scientifiques
Date d'achèvement 31/12/80	Dernière mise à jour 02/01/79	

1. OBJECTIF GENERAL

Etude pour toute installation nucléaire, de la dispersion des produits radioactifs depuis le point initial d'émission dans l'installation jusqu'à l'environnement et calcul, à l'intérieur et à l'extérieur de l'installation, des activités, débits de dose et équivalents de dose en fonction du temps.

Ceci doit conduire à l'évaluation réaliste des conséquences radiologiques du fonctionnement normal **et des divers accidents envisageables, compte tenu** des caractéristiques de l'installation et du site.

2. OBJECTIFS PARTICULIERS

2.1. Modélisation des phénomènes influant sur le transfert des produits actifs à l'intérieur de l'installation (piégeage, filtration, aspersion, dépôts d'aérosols).

2.2. Modélisation du transfert atmosphérique avec prise en compte du dépôt sec, du lavage, de l'existence de couches d'inversion.

2.3. Modélisation des effets biologiques sur l'homme : irradiation externe par la panache et les dépôts, contamination interne par inhalation ; doses individuelles et collectives.

3. INSTALLATIONS EXPERIMENTALES 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 de transfert atmosphérique à bouffées séquentielles.

4. ETAT DE L'ETUDE

4.1. Avancement à ce jour

La partie du programme (ALICE interne) traitant les transferts et piégeages à l'intérieur de l'installation est opérationnelle. On peut ainsi disposer, en fonction du temps et pour une séquence accidentelle dont les cinétiques d'émission sont connues :

- des concentrations volumiques dans tous les compartiments,
- des contaminations surfaciques des sols et murs,
- des activités piégées sur les filtres,
- des doses à l'intérieur de l'installation,
- des doses, dans l'environnement immédiat, dues au rayonnement à travers les parois des bâtiments,
- de la cinétique, par isotope, des rejets dans l'environnement, ce qui constitue le terme source pour la partie de programme ALICE-externe (dispersion atmosphérique et doses dans l'environnement) déjà opérationnelle précédemment.

4.2. Résultats essentiels

Applications d'ALICE-externe au calcul des doses dans l'environnement pour plusieurs séquences accidentelles hors dimensionnement pour les PWR.

5. PROCHAINES ETAPES

- Mise au point fine de la partie ALICE-interne et utilisation sur des séquences de complexité croissante.
- Examen des possibilités de couplage d'ALICE-interne avec des codes spécialisés, tels les codes calculant l'évolution des aérosols en suspension.
- Amélioration de la partie ALICE-externe pour le calcul des effets à long terme ; en particulier introduction du lavage et de la remise en suspension du dépôt.
- Etude du raccordement possible des 2 parties du programme au niveau du modèle à bouffées séquentielles de transfert atmosphérique ; ceci implique pour ce dernier une écriture informatique adaptée au problème et optimisée.

6. RELATIONS AVEC D'AUTRES ETUDES

.../...

- Etudes des transferts atmosphériques. Fiche 123-1-01.

- Etudes des transferts hydrogéologiques. Fiche 123-3-01.

- Accidents de réacteurs PWR. Analyse des divers scénarios possibles ; conséquences quant aux transferts de la radioactivité à l'intérieur de la centrale et aux rejets hors du confinement. Fiche 148-1-02.

- Calcul à l'aide du code ALICE des conséquences radiologiques de rejets accidentels dans l'environnement d'une installation nucléaire. Applications. Fiche 148-2-02.

- Feux de sodium : modélisation et codes de calcul du comportement des aérosols sodés. Fiche 156-1-01.

- Accidents sur les réacteurs à neutrons rapides. Analyse des divers scénarios possibles ; conséquences quant aux transferts de la radioactivité à l'intérieur de la centrale et aux rejets hors du confinement. Fiche 158-1-01.

7. DOCUMENTS DE REFERENCE

Notes NOVATOME et DSN à paraître.

8. DEGRE DE DISPONIBILITE

Diffusion limitée.

123-1-01 223-1-01 / 4171-10		5.4.
TITRE Etude des transferts atmosphériques		Pays FRANCE
		Organisme Directeur CEA/DSN
TITLE (Anglais) Studies on atmospheric transfers.		Organisme Exécuteur CEA/DSN/SESRS
		Responsable J.P. MAIGNÉ DSN/SESRS/FAR
Date de démarrage 01/01/72	Etat actuel En cours	Scientifiques G. DEVILLE-CAVELIN B. CRABOL
Date d'achèvement 31/12/81	Dernière mise à jour 02/01/79	

1. OBJECTIF GENERAL

Cette étude a pour objectif général l'élaboration de codes de calcul qualifiés afin de prévoir quantitativement (concentrations moyennes et maximales, concentrations intégrées dans le temps, etc ...) le transfert par l'atmosphère vers l'environnement d'un polluant minoritaire résultant de rejets permanents ou accidentels des installations industrielles, nucléaires notamment.

2. OBJECTIFS PARTICULIERS

2.1. Objectifs théoriques

a/ Développement et perfectionnement du modèle de transfert atmosphérique utilisé au DSN par la prise en compte de nouveaux éléments de la dispersion atmosphérique :

- variation temporelle du vent (pour mémoire),
- variations spatiales du vent,
- surélévation des nuages ou panaches,
- cisaillement vertical du vent et variations de la stabilité.

b/ Développement et perfectionnement des méthodes d'utilisation des statistiques météorologiques afin d'affiner les prévisions effectuées à l'aide des codes de calcul par une meilleure prise en compte :

- des durées de rejets envisagés (accidents),

.../...

- des spécificités du site considéré sur le plan de la dispersion atmosphérique (en particulier, étude des persistances de situations météorologiques données pendant une durée déterminée).

c/ Comparaison des résultats des applications du modèle DSN avec ceux que l'on obtient, dans les mêmes conditions, avec d'autres modèles français ou étrangers.

2.2. Objectifs expérimentaux

a/ Qualification du modèle DSN à partir d'expériences de simulations des transferts atmosphériques in situ (traceurs) ou sur maquette en veine hydraulique (en particulier, détermination des limites éventuelles de validité pour les paramètres de diffusion actuellement utilisés, à savoir les écarts-types σ_x, σ_y, σ_z de la SANDIA-CORPORATION proposés en 1966).

b/ Adaptation éventuelle du modèle DSN pour des situations météorologiques et/ou orographiques complexes (vents faibles, brises côtières, terrain construit, tache thermique des villes, relief) afin d'étendre son domaine d'application (en particulier ajustement éventuel des écarts-types).

c/ Développement et perfectionnement des deux techniques expérimentales précédentes (traceurs sur le terrain et maquette en veine hydraulique).

d/ Etude de faisabilité concernant la technique nouvelle de télédétection acoustique (sodar) pour l'acquisition de certains paramètres météorologiques.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Les moyens expérimentaux comprennent :

- Une station météorologique complète située à PIERRELATTE et équipée d'un pylône de 100 m qui permet d'acquérir et de mettre en forme les données nécessaires aux études purement météorologiques (voir 2.1.b.). Cette station permet également d'assurer la couverture météorologique nécessaire lors de certaines campagnes de simulation in situ par traceurs.

- Deux mâts télescopiques de 30 m qui permettent de constituer deux stations météorologiques simplifiées mais mobiles. Ces stations sont utilisées pour assurer la couverture météorologique des campagnes de simulation in situ et pour étudier les particularités météorologiques locales d'un site.

- Le matériel de prélèvement et d'analyse des traceurs (SF_6 et Fréon 13 B 1).

Ce matériel comprend :

- Les dispositifs d'émission des traceurs à débit réglable.
- Les dispositifs de prélèvement d'échantillons gazeux.
- Le matériel d'analyse des prélèvements (chromatographes en phase gazeuse et intégrateur).

.../...

- Une veine hydraulique installée à EVIAN et associée à une chaîne de mesures des paramètres caractéristiques des phénomènes turbulents (profils de vitesse, intensités de turbulence, spectres énergétiques turbulents, hauteur de couche limite, hauteur de rugosité). Cette installation sert de support aux études de simulation sur maquette de la dispersion dans l'hypothèse d'une atmosphère neutre.
- Une seconde veine hydraulique, de dimensions plus réduites, implantée au DTCE/STT, est utilisée pour une étude de faisabilité de la simulation d'une atmosphère stable.
- Un dispositif pilote sodar-fluxmètre en location en temps partagé à partir de 1979.

4. ETAT DE L'ETUDE

4.1. Avancement à ce jour

Bilan actuel et propositions pratiques pour le traitement des transferts atmosphériques et des données météorologiques associées dans le cadre des dossiers de sûreté /1/.

- Objectif particulier 2.1.a. :

Elaboration du code de calcul ICAIR 2, nouvelle version du code ICAIR prenant en compte la variation spatiale du vent. Le code est opérationnel. La notice d'utilisation est en cours de rédaction.

- Objectif particulier 2.1.b. :

a/ Propositions quant aux mesures météorologiques à effectuer sur un site nucléaire /2/.

b/ Elaboration du code de calcul COTRAM 1, nouvelle version du code COTRAM, qui permet un calcul plus exact des concentrations moyennes annuelles sur un site à partir des probabilités d'occurrence de différentes situations météorologiques. La notice d'utilisation est en cours de rédaction.

- Objectif particulier 2.1.c. :

Une première comparaison des prévisions des transferts atmosphériques effectuées à l'aide du modèle DSN, d'un modèle du type PASQUILL et des abaques LE QUINIO a été effectuée. Les résultats ont été confrontés aux résultats expérimentaux provenant d'essais in situ (traceurs). Ce travail a fait l'objet de deux communications /3/ /4/.

- Objectif particulier 2.2.a :

. En raison de difficultés survenues au niveau de l'interprétation des quatre premières campagnes de simulation in situ par traceurs, pour des conditions de vents faibles, une cinquième campagne a été effectuée sur le site de CADARACHE, avec une couverture météorologique renforcée. L'exploitation des résultats est en cours à l'aide du code de calcul ICAIR 2, mis au point à cette occasion.

.../...

. L'étude préliminaire de la faisabilité et des limites d'application de la simulation des transferts en atmosphère neutre sur maquette en veine hydraulique est terminée. Cette étude a été l'objet d'une thèse de troisième cycle /5/.

- Objectif particulier 2.2.b. :

Non engagé.

- Objectif particulier 2.2.c. :

La construction du prototype de l'appareil de prélèvement à 10 voies avec commutation automatique d'une voie à l'autre et durée des prélèvements réglable est achevée. Les premiers tests de l'appareil ont déjà commencé.

4.2. Résultats essentiels

- Code ICAIR 2 (Objectif 2.1.a.)

- Code COTRAM 1 (Objectif 2.1.b.).

- Détermination des limites d'application de la simulation sur maquette en veine hydraulique des transferts en atmosphère neutre (Objectif 2.2.a.).

- Réalisation de l'appareil de prélèvement de traceurs à 10 voies (Objectif 2.2.c.).

5. PROCHAINES ETAPES

- Objectif particulier 2.1.a. :

Prise en compte de la surélévation des nuages et panaches (1979).

- Objectif particulier 2.1.b. :

Recherche d'une méthodologie pour déterminer, sur un site donné, des séquences types d'une durée donnée de situations météorologiques, en affectant à chacune de ces séquences une probabilité d'occurrence (étude liée à celle des persistances de situations météorologiques données) (1980).

- Objectif particulier 2.1.c. :

Poursuite de la comparaison du modèle DSN avec les autres modèles français ou étrangers. Possibilité d'une participation à l'étude de comparaison des modèles projetée par la C.C.E.

- Objectifs particuliers 2.2.a. et 2.2.b. :

. interprétation des expériences "vents faibles" à l'aide du code ICAIR 2.

.../...

. Réalisation d'expériences de simulation en veine hydraulique sur la maquette de CADARACHE pour les conditions météorologiques obtenues dans les expériences in situ "vents faibles". Comparaison des résultats.

. Réalisation d'expériences sur maquette et in situ (traceurs) pour un terrain construit :

- Faisabilité d'une étude sur maquette pour une centrale nucléaire avec réfrigérants atmosphériques.

- Objectif particulier 2.2.c. :

Etude de la faisabilité de la simulation en veine hydraulique d'une atmosphère stable (DTCE/STT).

Poursuite des tests de l'appareil de prélèvement à 10 voies et construction en petite série (1979).

- Objectif particulier 2.2.d. :

Suivi des tests EdF sur le matériel de détection acoustique, puis location en temps partagé d'un appareil pour des tests complémentaires, par comparaison avec les données fournies par une station météorologique du C.E.A.

6. RELATIONS AVEC D'AUTRES ETUDES

Calcul à l'aide du code ALICE des conséquences radiologiques de rejets accidentels dans l'environnement d'une centrale nucléaire. Applications. Fiche 148-2-02.

7. DOCUMENTS DE REFERENCE

/1/ Rapport DSN N° 193.
Traitement des transferts atmosphériques et données météorologiques associées.
C. DEVILLERS.

/2/ Acquisition et traitement des données météorologiques sur les sites d'installations nucléaires.
G. DEVILLE-CAVELIN.
Séminaire sur la dispersion en milieu physique naturel S.F.R.P.-
CADARACHE/ MARS 1978.

/3/ Prévision quantitative des transferts de polluants par l'atmosphère.
J.P. MAIGNE.
Séminaire sur la dispersion en milieu physique naturel S.F.R.P.
CADARACHE/ MARS 1978.

/4/ Study of pollutant dispersion in water and air.
R. PORTE et J.P. MAIGNE.
Conférence ENS-ANS "The nuclear power reactor safety" - BRUXELLES - 16-19 octobre 1978.

/5/ Contribution à l'étude de la simulation en laboratoire des transferts de masse en atmosphère neutre.Thèse de 3è cycle - B. CRABOL.

8. DEGRE DE DISPONIBILITE

Le document [1] est à diffusion limitée.

Les documents [2], [3], [4] et [5] sont disponibles.

123-2-01 / 223-2-01 /4175-20		5-4
TITRE Etude des transferts de polluants dans les eaux de surface et dans la mer.		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) Studies on pollutant transfers in surface waters and in the sea.		Organisme Exécuteur CEA/DSN/SESRS Responsable J. DUCO
Date de démarrage 1/1/77	Etat actuel En cours	Scientifiques C. BOHET (ECOPOL)
Date d'achèvement 31/12/1979	Dernière mise à jour 02/01/1979	

1. OBJECTIF GENERAL

L'étude des transferts océaniques et hydrologiques doit permettre d'effectuer des prévisions quantitatives de la dilution dans l'eau de mer, des fleuves et des lacs des effluents susceptibles d'y être rejetés de manière continue ou accidentelle par des installations nucléaires.

2. OBJECTIFS PARTICULIERS

2.1. Dispersion des rejets dans le champ proche (par exemple, à moins de 300m du point de rejet).
Dans cette zone,où la dilution dépend fortement des conditions locales de rejet, l'objectif est d'expertiser les modèles qui peuvent servir pour le recalage de certains modèles de dispersion utilisés pour le champ lointain.

2.2. Mise au point de codes de calcul opérationnels pour l'évaluation de la dilution des effluents dans le champ lointain, en mer ou en rivière.
La dilution dépend alors presque exclusivement des propriétés du milieu récepteur ; ceci est en général réalisé à 500m du point de rejet.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

4. ETAT DE L'ETUDE

4.1. Avancement à ce jour

Objectif 2-1

L'expertise de la méthode de calcul en champ proche de EdF/LNH est terminée.

.../...

Objectif 2-2

- L'expertise de la méthode de calcul d'EdF/LNH des courants marins le long des côtes soumises à des marées importantes (Manche en particulier) est terminée.

- La mise au point du code ISOPOME pour le calcul de la dilution d'un polluant dans le champ lointain dans le cas de mers peu profondes soumises aux marées est achevée. Ce code est une application du modèle de diffusion à émissions séquentielles mis au point au DSN ; les écarts-types des distributions gaussiennes horizontale et verticale proviennent d'une compilation effectuée par OKUBO ; les trajectoires des centres de masse des diverses "bouffées" émises sont des données provenant soit de mesures (lâcher et suivi de flotteurs), soit de calculs des courants avec la méthode d'EdF/LNH.

- L'expertise du modèle de calcul du champ de concentration développé par l'EdF/LNH est terminée et une comparaison des résultats des modèles DSN et EdF a été faite dans le cas de la centrale de PALUEL.

- Deux modèles pour le calcul de la dilution en rivière de rejets ont été mis au point ; le premier, fondé sur le modèle à émissions séquentielles, permet de traiter le cas le plus général, tandis que le second, qui suppose une vitesse d'entraînement grande par rapport à la vitesse de diffusion longitudinale, autorise une solution analytique (modèle à disque).

4.2. Résultats essentiels

Objectif 2-1

La méthode de calcul en champ proche d'EdF est une méthode aux différences finies permettant une résolution tridimensionnelle des équations locales moyennées dans le temps, où des hypothèses ont été introduites pour alléger les calculs.
Il en résulte que le domaine pratique d'utilisation du modèle est limité aux cas où le rejet est fortement entraîné par un courant traversier, ce qui n'est souvent pas le cas.

Objectif 2-2

- La méthode de calcul des courants d'EdF/LNH repose dans le cas de la Manche sur trois modèles "gigognes" : le modèle général de grande emprise, le modèle régional et le modèle local, chaque modèle fournissant les conditions aux limites pour l'emploi du modèle suivant. Les calculs prévisionnels avec ce modèle de courants sont en bon accord avec les mesures in situ en ce qui concerne les courants instantanés et le marnage ; néanmoins un calage supplémentaire à partir de suivis de flotteurs est nécessaire pour atteindre les courants moyens de dérive, qui ont pourtant une contribution essentielle pour la dispersion des rejets.

.../...

- En ce qui concerne la dilution en mer dans le champ lointain (mers à marées) le modèle d'EdF/LNH a l'avantage de prendre en compte les particularités du site, notamment la non uniformité locale des courants ; néanmoins la définition du coefficient de dispersion utilisé ne permet pas un calage expérimental direct. Le code ISOPOME est d'un emploi plus souple et s'appuie sur des coefficients expérimentaux extraits de la littérature ; toutefois ceux-ci n'ont pas été complètement qualifiés le long des côtes. Quoi qu'il en soit une comparaison des deux méthodes pour le site de PALUEL a donné des courbes enveloppes voisines.

- Pour ce qui est enfin de la dilution en rivière l'emploi du modèle simplifié à disque est recommandé chaque fois que la vitesse du fleuve est grande devant la vitesse de diffusion longitudinale et que le débit du rejet est constant ou lentement variable. Une difficulté demeure au niveau de la connaissance de la loi de variation en fonction du temps de l'écart-type transversal de la distribution gaussienne.

5. PROCHAINES ETAPES

Objectif 2-1

Pas de nouveau développement en vue compte tenu des besoins prévisibles de l'analyse de sûreté.

Objectif 2-2

- Terminer la compilation bibliographique en cours sur la dispersion en champ lointain en fleuve (recherche d'écarts-types expérimentaux) et achèvement du code fondé sur le modèle à émissions séquentielles.

- Maintenir opérationnels et réactualiser autant que nécessaire les moyens de calcul du DSN pour la dilution dans le champ lointain (mers à marée, rivières).

- Effectuer un bilan des besoins de l'analyse de sûreté afin de définir d'éventuelles études nouvelles (par exemple opportunité d'engager la mise au point d'un modèle de dispersion dans les lacs ou les mers sans marée).

123-3 - 01/4173-10 223-3 - 01		
TITRE Etude des transferts hydrogéologiques		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) Studies of hydrogeological dispersion		Organisme Exécuteur CEA/DSN/SESRS Responsable J-Ch. PEYRUS DSN/SESRS/FAR
Date de démarrage 01/01/1976	Etat actuel en cours	Scientifiques Ch. MADOZ-ESCANDE
Date d'achèvement 31/12/ 1982	Dernière mise à jour 02/01/79	

1. OBJECTIF GENERAL

L'étude des transferts hydrogéologiques doit permettre de fournir des éléments techniques pour la prévision quantitative des transferts de pollution sous l'angle de la sûreté des installations nucléaires : prévision des temps de transferts jusqu'aux exutoires naturels ou artificiels et des concentrations à attendre à ces mêmes exutoires.

2. OBJECTIFS PARTICULIERS

2.1. Etude d'un modèle mathématique tridimensionnel simple de transferts hydrogéologiques (solution gaussienne dans l'hypothèse de lois de sorption "idéales").

2.2. Etude d'un modèle mathématique numérique tridimensionnel de transferts hydrogéologiques tenant compte de manière plus réaliste des lois physiques d'adsorption (nouvel objectif).

2.3. Mise au point de méthodes expérimentales permettant d'obtenir les données nécessaires à la vérification des modèles correspondant aux objectifs 2-1 et 2-2.

2.4. Recherche fondée sur des essais "in situ" des paramètres et des coefficients d'une série de terrains "types" pour l'établissement d'abaques de coefficients de diffusion en fonction des vitesses d'écoulement de la nappe du terrain aquifère.

.../...

2.5. Mise au point de méthodes "in situ" permettant d'établir les lois d'adsorption dans un terrain (double traçage).

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Les installations expérimentales sont mises en oeuvre pour effectuer les études correspondant aux objectifs particuliers 2-3, 2-4 et 2-5.

Ces installations sont constituées par :

- Un terrain équipé pour des essais de traçage au BARP, qui a été utilisé pour la mesure des transferts hydrogéologiques à l'échelle décamétrique en milieu sableux fin.

- Un terrain d'expérimentation "GRAVES" à VILLENAVE D'ORNON, qui est utilisé pour l'étude des transferts hydrogéologiques par la méthode bipuits (thèse de M. SAUTY - BRGM).

- Un second terrain d'expérimentation "GRAVES", en cours d'équipement, qui sera utilisé pour des essais de transferts hydrogéologiques à l'échelle décamétrique en milieu alluvionnaire grossier.

- Un terrain d'expérimentation dont le choix est à faire, qui sera utilisé pour des essais de transferts hydrogéologiques en milieu fracturé.

4. ETAT DE L'ETUDE

4.1. Avancement à ce jour

- La mise au point du modèle mathématique TRIDISOL, avec recherche automatique des coefficients est achevée (l'objectif 2-1 est donc atteint).

- Les expérimentations sur le site du BARP (milieu sableux fin) sont terminées. Les résultats sont en cours de dépouillement (objectifs 2-3 et 2-4).

- Les expérimentations sur le second site "GRAVES" débuteront au cours du 2è semestre 1979 après la phase d'étude hydrogéologique préliminaire du site qui est en cours (objectifs 2-4, 2-5).

- La recherche d'un terrain d'expérimentation en milieu fracturé est en cours (objectif 2-4).

4.2. Résultats essentiels

Une première interprétation des expérimentations sur maquette et sur le site du BARP montre qu'il est délicat de comparer, pour un même milieu, les résultats obtenus en laboratoire sur un prélèvement à ceux obtenus "in situ".

5. PROCHAINES ETAPES

En priorité, dépouillement complet des essais effectués sur maquette et sur le chantier du BARP (publication des rapports d'interprétation avec TRIDISOL au cours du 1er trimestre 1979).

.../...

Ultérieurement :

- Etude de la faisabilité d'une méthodologie (méthode bipuits) permettant de simplifier les expérimentations de terrain.

- Mise au point d'une méthodologie pour la détermination "in situ", en milieu fracturé, des coefficients de diffusion.

- Etude de faisabilité et mise au point d'une sonde permettant de mesurer le deutérium en forage, en vue d'utiliser la technique du double traçage.

	CLASSIFICATION: 5.4
TITLE (ORIGINAL LANGUAGE): RICERCHE SUL TRASFERIMENTO DI ALCUNI ELEMENTI STABILI DALL'AMBIENTE ALL'UOMO E SUL LORO METABOLISMO IN UOMO	COUNTRY: ITALY
	SPONSOR: CNEN, IAEA
TITLE (ENGLISH LANGUAGE): RESEARCH ON THE TRACE ELEMENT PATHWAYS FROM THE ENVIRONMENT TO MAN AND ON THEIR METABOLIC BALANCE IN MAN.	ORGANISATION: CNEN
	PROJECT LEADER: G.F. CLEMENTE
INITIATED: Jan. 1969 / COMPLETED:	SCIENTISTS: G.F. CLEMENTE, G. INGRAO
STATUS: in progress / LAST UPDATING: July 1979	

1. General Aim:

In order to increase our basic knowledge about the functions and the effects of the trace elements on human life it is extremely important to study their geographical world-wide distribution; the knowledge of such a distribution is in fact very helpful to distinguish between the normal levels of trace elements and those levels which are caused by some local either source of pollution or/and deficiency. Furthermore the estimation of radiation dose to the human body requires a certain amount of data about daily intakes and metabolic balances of the stable elements which serve as carriers of the corresponding radioactive elements. Since the uptake and retention in man of many radionuclides cannot be determined accurately, data on the corresponding stable elements are often useful for constructing a retention model for the reference man. Particularly long-lived nuclides in fallout from weapon testing or the natural radionuclides may be expected to approximate a state of equilibrium in the environment, and the stable element metabolism is appropriate.

2. Particular Objectives:

The research progam is aimed to define the levels of some trace element in Italy and to assess their main pathways from environment to man and their metabolic balance in the Italian population. Various groups of Italian population living in different areas of Italy have been followed up for one week to determine the daily dietary intake, daily total excretion, blood concentration and hair concentration of the following trace elements: Ag, Co, Cr, Cs, Eu, Fe, Hg, Ni, Rb, Sb, Sc, Se, Sn and Zn. Furthermore the toxic effects of some trace metals will be also studied in exposed subjects, in order to compare the effects due to the radioactive pollutants to those due to the trace element.

3. Experimental Facilities and Programme:

Instrumental neutron activation analysis has been employed. The irradiation facility is the 1 MW Triga Reactor of the CSN Casaccia.

39.507

TITLE (ENGLISH LANGUAGE): RESEARCH ON THE TRACE ELEMENT PATHWAYS FROM THE ENVIRONMENT TO MAN AND ON THEIR METABOLIC BALANCE IN MAN	CLASSIFICATION: 5.4

Ge (Li) true coaxial detectors of large volume together with computerized multichannel analyzer have been applied to the gamma spectrometry analysis of the samples after irradiation.

4. Project Status:

   1. Progress to Date: Six population groups, mainly located in the middle and northern part of Italy, have been examined and about 60 subjects have been followed up until now. The metabolism of mercury vapor and its cytogenetic effects have been studied in a group of exposed subjects.

   2. Essential Results: The Italian data have been compared to the reference man data reported by ICRP 23 thus showing big differences for some trace elements between the Italian metabolic data and those reported by the ICRP. Furthermore some preliminary data referring to long-term human exposure to elemental mercury vapor seem to indicate that significant cytogenetic effects are induced at exposure levels which do not show other neurological sign and/or symptom..

5. Next Steps: Other population groups mainly living in the southern part of Italy will be studied in the future.

6. Reference Documents:

6.1 Clemente G.F., Cigna Rossi L., Santaroni G.P. (1978). "Studies on the distribution and related health effects of the trace elements in Italy". In course of publication in the Proc. of the Conf. on Trace Substances in Environmental Health, Univ. of Missouri, Columbia.

6.2 Clemente G.F., Cigna Rossi L., Santaroni G.P.(1978). "Trace element composition of hair in the Italian population". Int. Symp. on Nuclear Activation in the Life Sciences, pp 527-543, IAEA, Vienna.

6.3 Clemente G.F., Cigna Rossi L., Santaroni G.P. (1978). "Studies on the trace element distribution in the diets and population of Italy". In course of publication in Reviews on Environmental Health.

6.4 Mariani A., Clemente G.F., Santaroni G.P. (1979). "Mercury levels in food and its intake in high risk population groups". Proc. of the GEN Symposium, Budapest, 8-10 November 1978 (in course of publication).

6.5 Clemente G.F., Cigna Rossi L., Santaroni G.P. (1979). "Nickel in foods and dietory intake of nickel". Biogeochemistry of Nickel, book to be published by J. Wiley and Sons.

7. Degree of Availability: Free.

G.F. Clemente, CNEN, CSN Casaccia, C.P. 2400, I-00100 Roma.

	CLASSIFICATION: 5.4 5.5
TITLE (ORIGINAL LANGUAGE): RICERCHE SUI RADIONUCLIDI NELL'AMBIENTE	COUNTRY: ITALY SPONSOR:
TITLE (ENGLISH LANGUAGE): RESEARCHES ON RADIONUCLIDES IN THE ENVIRONMENT	ORGANISATION: CNEN PROJECT LEADER: F.GIORCELLI

INITIATED: Jan. 1961	COMPLETED:	SCIENTISTS: Laboratorio Contaminazione Continentale
STATUS: In progress	LAST UPDATING: Jun. 1979	

1) General Aim: Surveillance on the radioactive contamination of the environment, on a country-wide scale.

2) Particular objectives
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.

3) Experimental facilities and Programme:
3.1 Experimental facilities
   a) Laboratory of Chemistry
   b) Low-level beta counters
   c) Gamma spectrometry, Ge(Li) detectors
   d) Atomic absorption flame spectrophotometry

3.2 Programme
Systematic measurements of radioactive contamination in several kinds of environmental samples, as shown in Table 1.

45.897

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
RESEARCHES ON RADIONUCLIDES IN THE ENVIRONMENT -2-	5.4 5.5

Table 1

Item	Number of sampling points	Radionuclides measured	Measurements frequency
Fallout	1	Sr^{90}, gamma spect.	Monthly
River water	14	Sr^{90}	Quarterly
Sea water	4	Sr^{90}, Cs^{137}	Twice a year
Irrigation water	2	Sr^{90}	Quarterly
Milk	11	Sr^{90}, Cs^{137}	Quarterly
Vegetables (1)	1	Sr^{90}, gamma spectr.	Quarterly
Bovine meat	2	Cs^{137}	Quarterly
Fish (2)	3	Gamma spectrometry	Quarterly

(1) Brassica oleracea, Lactuca sativa, Lycopersicum esculentum, Malus domestica, Solanum tuberosum.

(2) Clupea pilchardus, Engraulis encrasicholus, Scomber scomber, Mytilus galloprovincialis

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Together with Sr^{90} and Cs^{137} measurements, determination of the isometabolic stable elements Ca and K are also performed.

4) Project Status:
   4.1 Progress to date
   Within few months will be ready the data referring to 1978
   4.2 The present levels of radioactive contamination on country-wide scale, due to fall-out from atmospheric nuclear tests, are low. Normally, only small concentrations of long lived nuclides (Sr^{90}, Cs^{137}) are detectable in environmental samples. Occasionally, in the periods following atmospheric nuclear tests, low concentratior of short lived nuclides (Ba^{140}, I^{131}, Zr^{95}, Nb^{95}, Ce^{141}, Ce^{14}, Ru^{103}, etc.) are measured.

5) Next steps
   The work will be continued as in previous years.

6) Relation to Other Projects and Codes:
   The activity here described is the contribution of the Continental Contamination Laboratory, CNEN, to the "Reti Nazionali per la Sorveglianza della Radioattività Ambientale" (National Networks for the Surveillance of Environmental Radioactivity) co-ordinated by CNEN under auspices of the Italian Health Ministry, with the co-operation of the following Organizations:

TITLE (ENGLISH LANGUAGE): RESEARCHES ON RADIONUCLIDES IN THE ENVIRONMENT -3-	CLASSIFICATION: 5.4 5.5

CCR-EURATOM
(Centro Comune Ricerche Euratom)
Servizio di Protezione
ISPRA (Varese)

CISE
(Centro Informazioni Studi Esperienze)
SEGRATE (Milano)

CNEN
(Comitato Nazionale Energia Nucleare)
Viale Regina Margherita, 125
ROMA

MDA-SERV.METEO-GNMRA
(Ministero della Difesa Aeronautica
Servizio Meteorologico
Gruppo Naz.Misure Radioattività dell'Aria)
P.le Luigi Sturzo, 31
ROMA

Istituto di Scienze dell'Alimentazione
dell'Università di Pavia
Via Taramelli, 1
PAVIA

7) Reference Documents:
The results obtained by all the Agencies co-operating to the "Reti Nazionali per la Sorveglianza della Radioattività Ambientale" are reported in "Data on Environmental Radioactivity Collected in Italy", ed. CNEN, Roma

8) Degree of Availability
The series of reports quoted before is available from CNEN (Direzione Centrale per la Sicurezza Nucleare e la Protezione Sanitaria), Viale Regina Margherita 125, 00198 Roma.

	CLASSIFICATION: 5.4 5.5
TITLE (ORIGINAL LANGUAGE): STUDI PER LA VALUTAZIONE DELL'IMPATTO DI IMPIANTI NUCLEARI SULL'AMBIENTE MARINO	COUNTRY: Italy SPONSOR: CNEN-CEE
TITLE (ENGLISH LANGUAGE): STUDIES FOR THE EVALUATION OF THE IMPACT OF NUCLEAR PLANTS ON THE MARINE ENVIRONMENT	ORGANISATION: CNEN-CEE PROJECT LEADER: Pietro SCOPPA
INITIATED: 1957 / COMPLETED:	SCIENTISTS:
STATUS: in progress / LAST UPDATING: June 1979	

1. General Aim:
According to the increasing utilization of nuclear energy as a means of electricity generation, the marine environment will play an important role as a disposal area for large quantities of low-level radioactive liquid wastes. Furthermore, it will receive thermal discharges and biocides used for fouling control. Therefore, extensive studies are directed to the evaluation of the impact of nuclear plants discharges under health protection and ecotoxicological standpoints.
2. Particular Objectives:
2a - A better knowledge on the interactions between radioactive, thermal and chemical discharges, taking into consideration their effects on marine organisms.
2b - Modelling of marine systems: development of computer programs and application of models to specific areas.
3. Experimental Facilities and Programme:
The basic instrumentation to perform research in the fields of physical oceanography, chemistry, botany, zoology, microbiology, radioecology, etc., is available.
The programme is composed by several research lines, such as:
3a - Interactions between chlorine and seawater components.
3b - Influence of water temperature on the transfer of radionuclides.
3c - Combined effects of biocides and increased water temperature on marine organisms.
3d - Levels of radionuclides in seawater, sediments, and organisms.
3e - Studies on coastal currents and transport.
3f - Evaluation of radiation protection data collected in typical areas.
3g - Long-term transfer of radionuclides in marine ecosystems.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
STUDIES FOR THE EVALUATION OF THE IMPACT OF NUCLEAR PLANTS ON THE MARINE ENVIRONMENT -2-	5.4 5.5

4. Project Status (progress to date and essential results):

4a - A mathematical model to calculate equilibrium distribution of oxidant species present in chlorinated seawater has been implemented.

4b - Studies on the release of ^{51}Cr by fish at 10 and 20 C are in progress.

4c - The resistance of several marine organisms to increased water temperature has been determined under laboratory conditions.

4d - Levels of beta- and gamma-emitting radionuclides have been measured in a large number of samples (water, sediments, organisms) collected along Italian coasts.

4e - Collection and elaboration of data needed for the evaluation of currents and transport in typical sites are in progress.

4f - A model for the predictive description of the behaviour of radionuclides in the La Maddalena Archipelago has been completed.

4g - Preliminary studies on accumulation and release of radionuclides in/from marine sediments have been directed to the improvement of sampling methodology and analytical techniques for the determination of trace elements.

5. Next Steps:

Future work will be in accordance with the programme described under 3.

6. Relation to other Projects and Codes: none.

7. Reference Documents:

ANDREOLI, G. et al. - Environmental Survey in the La Maddalena Archipelago: Data Reports of the July 1975 to April 1977 Campaigns.
RT/BIO(78)11, RT/BIO(78)29, RT/BIO(78)32. Other three reports of the series RT/BIO on the same subject in press.

BERNHARD, M., ZATTERA, A. - La distribuzione di alcuni polluenti nell'ambiente marino con tentativi di stima delle conseguenze.
ARCH. Oceanogr. Limnol., 18, Suppl. 3 (1976) 83-111.
RT/BIO(78)14.

BONIFORTI, R. - Metal Analysis in Aquatic Sediments. A review.
Thalassia Jugoslavica (in press).

TITLE (ENGLISH LANGUAGE): STUDIES FOR THE EVALUATION OF THE IMPACT OF NUCLEAR PLANTS ON THE MARINE ENVIRONMENT -3-	CLASSIFICATION: 5.4 5.5

BONIFORTI, R. - Alcune considerazioni sulla contaminazione del Mediterraneo.
PRO/RTI(1978)LSAM 1

BONIFORTI, R. - TOUSSAINT, C.J. - Application of the X-ray fluorescence spectrometry to the study of the marine environment.
Int. J. Environ. Anal. Chem. (1978) (in press)

CEPPODOMO, I., GALLI, C., RAMPI, L., ZATTERA, A. - La Distribuzione del fitoplancton nell'Arcipelago di La Maddalena. Luglio 1975. Febbraio 76. Giugno-Luglio 76.
RT/BIO(78)44. RT/BIO(78)48. RT/BIO(78)49.

GALLI, C., ZATTERA, A. - Processi e modelli di diffusione nella baia di S. Stefano: esperienze del luglio 1975.
RT/BIO (78) 30.

GALLI, C., ZATTERA, A. - Processi e modelli di diffusione nella baia di S. Stefano: esperienze del luglio 1976.
RT/BIO (78) 37.

GALLI, C., ZATTERA, A. - Temperature as pollutant. Long term effects and thermal shock on some marine phytoplankters.
RT/BIO (78) 31.

GALLI, C., ZATTERA, A. - Accumulation of Cs by some marine phytoplankters.
Proceedings of the XXVI Congress of the ICSEM, Antalya (1978), 24-28 (in press).

PIRO, A., ROSSI, G., and PAPUCCI, C.- Chemical and physicochemical measurements in the La Maddalena Archipelago.
RT/CHI(78)1.

SCHULTE, E.H., SECONDINI, A., FIORE, V. - La distribuzione dello zoobenthos nell'Arcipelago di La Maddalena.
RT/BIO(78)28.

SCHULTE, E.H. - Studies on the distribution of zoobenthos on the Northern Sardinian Coast.
Proceedings of the XXVI Congress of the ICSEM, Antalya, (1978) (in press).

TITLE (ENGLISH LANGUAGE): STUDIES FOR THE EVALUATION OF THE IMPACT OF NUCLEAR PLANTS ON THE MARINE ENVIRONMENT -4-	CLASSIFICATION: 5.4.5.5

SCOPPA, P., BOUDOUIN, M.F. - Applications of various mathematical models to data from the accumulations of radionuclides by aquatic organisms.
Bull. H.P.-2o Calculator Log. No. 1307 (1978).

C.N.E.N.-C.N.R. - Misure di temperatura e salinità con sonda automatica nell'Arcipelago di La Maddalena.
RT/FI(78) (in press).

C.N.E.N.-C.N.R. - Data report: Current and temperature measurements in the Archipelago of La Maddalena.
RT/FI(78)14.

C.N.E.N.- C.N.R. - Risultati degli esperimenti con traccianti svolti nell'arcipelago di La Maddalena per la determinazione del coefficiente di diffusione turbolenta locale.
RT/FI(78)12.

8. Degree of Availability: free, CNEN- Laboratorio per lo Studio dell'Ambiente Marino
I-19030 FIASCHERINO (La Spezia) Italy

		CLASSIFICATION: 5.4 - 5.5 - 5.6
TITLE (ORIGINAL LANGUAGE): Valutazione quantitativa del rilascio di sostanze radioattive naturali nell'ambiente		COUNTRY: ITALY SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Quantitative evaluation of the release of natural radioactive substances into the environment		ORGANISATION: CNEN PROJECT LEADER: M. Dall'Aglio
INITIATED: January 1974	COMPLETED:	SCIENTISTS: R. Gragnani, C. Orlandi, G. Paganin
STATUS: in progress	LAST UPDATING: July 1979	

1. General Aim

Study of the environmental impact of nuclear plants referred to the release of radioactive elements.

2. Particular Objectives

In the surroundings of some nuclear plants, study of the distribution and circulation of natural isotopes of radioactive elements which can be released by the plants, before the start of the industrial activity.

3. Experimental Facilities and Programme

A well equipped geochemical standard laboratory plus alfa and gamma spectrometry, neutron analysis equipment. Fluorimetric instrumentation for analysis. Radiological surveys are in program in Italian nuclear research centers and Uranium mine areas.

4. Project Status

Radioecological surveys have been carried out for the CNEN Trisaia Research Center and for the "Fabbricazioni Nucleari" Fuel Fabrication Plant, Bosco Marengo (Alessandria). Preparatory studies have been performed for the Novazza mine.

5. Next Steps

To extend the radioecological survey to all Italian nuclear plants and mines.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Quantitative evaluation of the release of natural radioactive substances into the environment	5.4 - 5.5 - 5.6

6. Reference Documents

Dall'Aglio M., De Cassan B., Ghiara E., Gragnani R. "Studio sulla distribuzione e sul comportamento dell'uranio e del fluoro nella zona interessata dagli scarichi dell'impianto Fabbricazioni Nucleari, Bosco Marengo (Alessandria)". Soc. Italiana di Mineralogia e Petrologia, Rendiconti, Vol. XXXII(1), pag. 437-459, 1976.

7. Degree of Availability

Free available by M. Dall'Aglio, CNEN, CSN Casaccia, C.P. 2400, I-00100 Roma.

N.V. KEMA	CLASSIFICATION : 5.4
TITLE : Gevolgen voor de omgeving van ongevallen bij kernenergiecentrales	COUNTRY: THE NETHERLANDS
TITLE (ENGLISH LANGUAGE): Environmental effects of nuclear power plants accidents	SPONSOR : KEMA ORGANIZATION : KEMA PROJECTLEADER : B.Th. Eendebak
INITIATED : - LAST UPDATING : 1978	SCIENTISTS : B.Th. Eendebak
STATUS : - COMPLETED : 1977	

General aim

To analyse the risks of light water reactors on specific sites in the Netherlands.

Particular objectives

To study the effects of nuclear accidents as a function of site, population density, wheather conditions, etc.

Experimental facilities

Not applicable.

Project status

Computer code "MAKRO" is available.

Next steps

Not applicable.

Relation to other projects

This project was started by an order of the Minister of Economic Affairs to make a risk analysis of the fuel cycle in the Netherlands. This study was finished in June 1975.
See also the projects "Calculation of the quantities of radioactivity released as a result of a serious reactor accident" and "Failure analysis by application of event and fault trees".

Reference documents

Not available yet.

Degree of availability

Through the organization KEMA.

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX

	CLASSIFICATION: 5.4, 5.6
TITLE (ORIGINAL LANGUAGE): THE DEVELOPMENT OF A COMPUTER CODE TO CALCULATE THE CONSEQUENCES OF A RELEASE OF RADIOACTIVE MATERIAL TO THE ATMOSPHERE	COUNTRY: UK
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD
	PROJECT LEADER: G D KAISER
INITIATED: 1974 / COMPLETED: 1977	SCIENTISTS:
STATUS: / LAST UPDATING: MAY 1979	

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 areas 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 organs of the body due to the inhalation of radioactive material.

b) whole-body gamma dose to external radiation from the passing cloud and from deposited fission products.

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 over all weather conditions, consequence/probability curves are plotted where the consequences can be a. thyroid cancer b. early deaths due to lung dose, bone dose and GI-Tract dose, c. cancers of organs other than thyroid, d. total man-rem, e. areas dangerously contaminated by deposited γ emitting fission products, deposited 137 Cs or deposited actinides at various times after the release, f. the number of people within those area and g. the area within which milk consumption must be barred.

The model used takes account of the effect of the duration of release on the atmospheric dispersion. The effect of surface roughness in included as are simple prescriptions for handling the effect of building wakes, of inversion lids and of plume rise.

3. PROJECT STATUS

The program in its present state is able to produce all the results described in Section 2.

	CLASSIFICATION: 5.4
TITLE (ORIGINAL LANGUAGE): THE ATMOSPHERIC DISPERSION OF RADIOACTIVE MATERIAL IN THE EVENT OF AN ACCIDENT TO A NUCLEAR INSTALLATION	COUNTRY: UK SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD, UKAEA PROJECT LEADER: G D KAISER
INITIATED: 1974 / COMPLETED:	SCIENTISTS: P C COOPER, C HARTER
STATUS: CONTINUING / LAST UPDATING: EARLY 1978	

GENERAL AIM

To provide a watertight method for evaluating the consequences of accidental releases of radioactive material to the environment.

PARTICULAR OBJECTIVES

The continuing development of the computer code TIRION.

PROGRESS TO DATE

TIRION is now a flexible tool with which a rapid assessment of the order of magnitude of the consequences of an accidental release of radioactive material to the environment can be made.

NEXT STEPS

1. Interfacing TIRION with codes such as FRASC, AEROSIM and FISPIN which describe processes within and/or leakage from a containment.

2. Modifications to the Meteorological model including time varying effects.

3. Automation of input and output.

4. Miscellaneous problems.

REFERENCES

SRD R62; SRD R63; SRD R85; SRD R120; SRD R134

DEGREE OF AVAILABILITY

Freely

		CLASSIFICATION: 5.4
TITLE (ORIGINAL LANGUAGE): EVALUATION OF CONSEQUENCES OF F.P. RELEASES (GSD1.1)		COUNTRY: UK
		SPONSOR: UKAEA
TITLE (ENGLISH LANGUAGE):		ORGANISATION: SRD/UKAEA
		PROJECT LEADER: RNH McMillan
INITIATED: December 1978	COMPLETED:	SCIENTISTS/ENGINEER MA King
STATUS:	LAST UPDATING: May 1979	

BACKGROUND

It is fundamental that the requirements of the European Directive on radiation protection standards are followed with regard to radiation exposure to persons on site and to members of the general public.

A thorough appreciation of the behaviour of postulated activity releases is necessary and must include treatment of factors both intrinsic and extrinsic to the design. The evaluation methods must therefore be capable of treating a wide range of site-dependent conditions such as demography, hydrology, meteorology, etc. in addition to generic design-dependent features. The programme is therefore addressed to the further development of established models which have been used in safety evaluation of various reactors (including specialised PWR) and other nuclear facilities.

OBJECTIVES

1. To report the current status on the development of the code TIRION, which predicts the environmental consequences of fission product release to atmosphere. The report will include statements on the extent of validation of the analytical models used in the programme.

2. To examine the results of sensitivity studies on the code TIRION, relevant to PWR conditions, and to carry out further sensitivity studies if necessary. Report results.

3. To report the current status of development of the core fission product inventory code FISPIN, its applicability to PWRs and its level of validation.

4. To review available models for the assessment of volatile fission product behaviour within containment including evaluation of the effect of post-accident engineered safeguards such as containment spray. To recommend adoption of appropriate models or, alternatively, the development of new methods.

FACILITIES

FISPIN: RISLEY 472 and 2980 AEEW 470 AERE 370
TIRION: RISLEY 472 AERE 361

REFERENCE DOCUMENT

SRD R 134, TIRION 4, A Computer Program for Use in Nuclear Safety Studies, Fryer and Kaiser, November 1978.

	CLASSIFICATION: 5.4 5.5
TITLE (ORIGINAL LANGUAGE): RICERCHE SUI RADIONUCLIDI NELL'AMBIENTE	COUNTRY: ITALY
	SPONSOR:
TITLE (ENGLISH LANGUAGE): RESEARCHES ON RADIONUCLIDES IN THE ENVIRONMENT	ORGANISATION: CNEN
	PROJECT LEADER: F.GIORCELLI
INITIATED: Jan. 1961 / COMPLETED:	SCIENTISTS: Laboratorio Contaminazione Continentale
STATUS: In progress / LAST UPDATING: Jun. 1979	

<table>
<tr><td colspan="2"></td><td>CLASSIFICATION:
5.4 5.5</td></tr>
<tr><td colspan="2">TITLE (ORIGINAL LANGUAGE):
STUDI PER LA VALUTAZIONE DELL'IMPATTO DI IMPIANTI NUCLEARI SULL'AMBIENTE MARINO</td><td>COUNTRY:
Italy
SPONSOR:
CNEN-CEE</td></tr>
<tr><td colspan="2">TITLE (ENGLISH LANGUAGE):
STUDIES FOR THE EVALUATION OF THE IMPACT OF NUCLEAR PLANTS ON THE MARINE ENVIRONMENT</td><td>ORGANISATION:
CNEN-CEE
PROJECT LEADER:
Pietro SCOPPA</td></tr>
<tr><td>INITIATED: 1957</td><td>COMPLETED:</td><td rowspan="2">SCIENTISTS:</td></tr>
<tr><td>STATUS: in progress</td><td>LAST UPDATING: June 1979</td></tr>
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	CLASSIFICATION: 5.4 - 5.5 - 5.6
TITLE (ORIGINAL LANGUAGE): Valutazione quantitativa del rilascio di sostanze radioattive naturali nell'ambiente	COUNTRY: ITALY SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Quantitative evaluation of the release of natural radioactive substances into the environment	ORGANISATION: CNEN PROJECT LEADER: M. Dall'Aglio

INITIATED: January 1974	COMPLETED:	SCIENTISTS: R. Gragnani, C. Orlandi, G. Paganin
STATUS: in progress	LAST UPDATING: July 1979	

N.V. KEMA	CLASSIFICATION : 5.5
TITLE : Bepaling van het aantal lekke splijtstofstaven en de kernpositie tijdens het reactorbedrijf	COUNTRY: THE NETHERLANDS
TITLE (ENGLISH LANGUAGE): Determination of the number of leaking fuel rods on the core position during operation	SPONSOR : KEMA ORGANIZATION : KEMA PROJECTLEADER : J. Hoekstra
INITIATED : - / LAST UPDATING : 1978	SCIENTISTS : J. Hoekstra
STATUS : - / COMPLETED : 1977	

General aim

To reduce the wet-sipping time.

Particular objectives

- An increase of the off-gas activity combined with a one step control-rod movement indicates the position of ruptured fuel.
- The release of certain fission-products during reactor start-up is a measure for the number failed fuel rods.

Experimental facilities

Dodewaard nuclear power plant.

Project status

Still in progress.

Next steps

Not applicable.

Relation to other projects

None.

Reference documents

None.

Degree of availability

Through the organization KEMA.

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX

	CLASSIFICATION: 5.4 - 5.5 - 5.6
TITLE (ORIGINAL LANGUAGE): Valutazione quantitativa del rilascio di sostanze radioattive naturali nell'ambiente	COUNTRY: ITALY SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Quantitative evaluation of the release of natural radioactive substances into the environment	ORGANISATION: CNEN PROJECT LEADER: M. Dall'Aglio
INITIATED: January 1974 | COMPLETED: STATUS: in progress | LAST UPDATING: July 1979	SCIENTISTS: R. Gragnani, C. Orlandi, G. Paganin

	CLASSIFICATION: [illegible]
TITLE (ORIGINAL LANGUAGE): THE DEVELOPMENT OF A COMPUTER CODE TO CALCULATE THE CONSEQUENCES OF A RELEASE OF RADIOACTIVE MATERIAL TO THE ATMOSPHERE	**COUNTRY:** UK **SPONSOR:**
TITLE (ENGLISH LANGUAGE):	**ORGANISATION:** SRD **PROJECT LEADER:** G D KAISER

INITIATED: 1974	**COMPLETED:** 1977	**SCIENTISTS:**
STATUS:	**LAST UPDATING:** MAY 1979	

6. FAULTS AND ACCIDENT COMBINATIONS

Classification 6

Title Common cause failure	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
	Project leader J.R. Taylor
Initiated 1974 Completed 1st January 1979	

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 Risø National Laboratory ORGANIZATION Risø National Laboratory
	Project leaders J.R. Taylor S. Bologna R. D'Agostino
Initiated 1973	

General aim

Research on the topic of computer reliability was originally taken up at two separate institutions - Risø and CNEN, Cassacia. On the specific topic of developing software for deriving systematic testing data for control programs, the two institutes joined forces. The joint project dates from 1977.

The present work is aimed at producing an interactive program, which will produce sets of data capable of testing all paths or all branches in a computer program.

Project status

Progress to date (April 1977) includes completion of the basic support routines and program language analyser for the program.

Relation with other projects

The project is part of a larger program of work at CNEN.

Reference documents

One report has been published.

J.R. Taylor, Proving Correctness of a Real Time Operating System, 3rd European Real Time Conference, Budapest 1973.

Availability

Copies of reports are available on request.

	CLASSIFICATION: 6
TITLE (ORIGINAL LANGUAGE): Driftsforstyrrelser på A-værker	COUNTRY: Denmark SPONSOR: Risø National Laboratory
TITLE (ENGLISH LANGUAGE): Systematic Analysis of Nuclear Power Plant Incidents	ORGANISATION: Risø National Laboratory PROJECT LEADER: H. Larsen
INITIATED: 1977 — COMPLETED:	SCIENTISTS: H. Larsen, H.E. Kongsø, E. Nonbøl
STATUS: In progress — LAST UPDATING:	

1. General Aim

Systematic analysis of incidents on nuclear power plants with the purpose of identifying trouble areas and event sequences of importance to safety. As a first step incidents on nuclear power plants have been analysed on the basis of Nuclear Power Experience Documents (NPE). NPE compiles and reports on the operating experience of Light Water Reactors, with emphasis on operating problems. A classification system has been set up, comprising a total number of 18 classification-criteria like: Docket No., Time of Commissioning, Primary Component, Secondary Component, Primary Fault·Machanism etc. Each incident is registered on punchcards. A computer program to analyse the data has been developed. An analysis comprising all reports concerning PWR's in 1977 has been performed.

2. Particular objectives

3. Experimental facilities and programme

4. Project status

In Progress.

5. Next steps

Future work on this subject will focus on a systematic analysis of safety related incidents based on Licensee Event Reports from various countries.

6. Relation with other projects

7. Reference documents

Annual Progress Report, Department of Reactor Technology, Risø 1979.

K. Alstrup, H.E. Kongsø and H. Larsen, Analysis of Incidents in Pressurized-Water Reactors. Second National Reliability Conference, Birmingham, 28-30 March 1979.