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

NUCLEAR SCIENCE AND TECHNOLOGY

European Community Water Reactor Safety Research Projects

VOLUME I

DECEMBER 1983

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European Community Water Reactor Safety Research Projects

VOLUME I

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DOCUMENT EXCLUSIVEMENT INTERNE

INTRODUCTION

This is the eigth 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 of 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
   Fed. Rep. of 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.

VOLUMES I AND II - WATER REACTORS

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DENMARK</td></tr>
<tr><td>SPONSOR : Risø National Laboratory</td></tr>
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NORCOOL. A Model for Analysis of a BWR under LOCA Conditions.</td><td>ORGANISATION : Risø National Laboratory</td></tr>
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O. Rathmann</td></tr>
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September 1976</td><td>COMPLETED :</td><td rowspan="2">SCIENTISTS :
O. Rathmann
P. Astrup
N. Bech
P. Hansen</td></tr>
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Under development</td><td>LAST UPDATING :</td></tr>
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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

4.1. Progress to data

NORCOOL-I is completed.
NORCOOL-II is under development.

4.2. Essential results

NORCOOL-I has been compared with 2 Göta experiments with good agreement.

NORCOOL-II has been tested regarding the basic hydrodynamics and heat conduction. Furthermore, it has been compared to transient natural circulation experiments with good agreement, and with single pin bottom quenching experiments, also with relatively good agreement.

5. Next steps

NORCOOL-II will be run-in ussing experimental test cases, e.g. - loss of coolant experiments, and quenching experiments performed at Risø.

6. Relation to other projects

The NORHAV projects includes:

a) The core heat-up programme RHC, Risø.
b) A one-dimensional blow down computer program NORA for reactor systems developed at IFE, 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 developed at Risø.
d) A 64-rod (electrically heated) core heat-up Göta experiment by Studsvik Energiteknik, Sweden.

Furthermore, NORCOOL-II is used in connection with the European Economic Community
Indirect Action Programme on the Safety of Thermal Water Reactors, Project No. 6A, Contract No. 022 SRDK
"Study of rewetting and quench phenomena by single pin out-of-pile experiments, with special emphasis on the effort of pin composition".

7. Reference documents

1. 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.

2. O. Rathmann, Ö. Rosdahl and P. Astrup: NORCOOL-II: An Advanced Computer Code for Thermohydraulic LOCA Analysis of a BWR. Part I: Model Description. NORHAV-D-79.

3. O. Rathmann, NORCOOL-II. Input-Output Manual. NORHAV-D-94. Risø National Laboratory. April 1982.

4. O. Rathmann, NORCOOL-II. Part III: Numerical Methods and Program Description of the Heat Component Part. NORHAV-D-96. Risø National Laboratory. August 1982.

5. O. Rathmann, Global Solution for NORCOOL-II. NORHAV-D-97. Risø National Laboratory. September 1982.

8. Degree of availability

NORCOOL-I is implemented in the US-NRC program system WRAP-EM at Savannah River Laboratory and has been verified, but not finally reported. NORCOOL-II has not been developed to a degree justifying release.

145-1-01	1
TITRE CALCULS DES ACCIDENTS DE REFROIDISSEMENT DES REACTEURS A EAU : PROGRAMME GENERAL COORDONNE DE MISE AU POINT DES CODES.	*Pays* FRANCE *Organisme directeur* CEA/DSN
TITRE en anglais EVALUATION OF PWR COOLANT ACCIDENT : COORDINATED GENERAL PROGRAM OF CODE DEVELOPMENT.	*Organisme éxécuteur* CEA/DSN-SRS/SEAREL CEA/DRE *Responsables* M. REOCREUX SEAREL/CEN-FAR
Date de démarrage 01.01.71 — *Etat actuel* en cours *Date d'achèvement* — *Dernière mise à jour* 01.1983	*Scientifiques* A. FORGE, M. MEZZA R. POCHARD, P.PROBST R. GONZALEZ, P.TREFOURET MA. VIVIANDE

1. OBJECTIF GENERAL :

Mise au point de codes en vue du calcul des accidents de perte de caloporteur dans les réacteurs à eau pressurisée.

2. OBJECTIFS PARTICULIERS :

2.1 - Opérationnalité

- Mise sur machine des codes disponibles.
- Adaptation informatique

2.2 - Qualification-validation.Suivi des programmes expérimentaux

La qualification-validation a pour but de vérifier les capacités des codes à prévoir les phénomènes séparés se produisant au cours des accidents étudiés.

Cette qualification-validation est faite sur les expériences analytiques ou à effets séparés réalisées en France (OMEGA, ERSEC, MOBY DICK, CANON,...) ou à l'étranger.

Cette qualification-validation nécessite un suivi de ces programmes.

2.3 - Evaluation-vérification

L'évaluation-vérification des codes a pour but de vérifier les capacités des codes à coupler les différents phénomènes séparés et pour certains d'entre-eux à effectuer une première transposition d'échelle avant la transposition au réacteur.

L'évaluation-vérification doit être réalisée par des analyses des essais effectués sur les grandes expériences globales (LOFT, SEMISCALE, LOBI, PKL, ...).

2.4 - Calculs réacteurs

Ces calculs comprennent :

- une phase de mise au point où il s'agit de rendre opérationnel les outils de calcul pour les différents types d'accidents (grosse brèche, petite brèche, transitoires type TMI,..)

- des études de sensibilité mettant en évidence les points où des études supplémentaires de qualification-validation sont requises.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES :

Ces études s'appuient sur l'ensemble des expériences suivantes :

- Expériences analytiques et à effets séparés

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 dans fichier 1980
MARVIKEN-CFT	Ecoulement diphasique critique à grande échelle	participation au programme international. voir fiche 145-1-08 dans fichier 1980
OMEGA	Ecoulement et transfert de chaleur en dépressurisation	voir fiche 145-1-04
ERSEC	Ecoulement et transfert de chaleur en renoyage. Modèle de front de trempe	voir fiche 145-1-05 dans fichier 1980
EPIS	Injection de secours	voir fiche 145-1-07 dans fichier 1980
EVA, EPOPEE	Pompe en diphasique	EVA programme FRA-CEA EPOPEE programme EDF
EDGAR	Thermomécanique de la gaine du combustible	voir fiche 145-2-02
PHEBUS	Thermomécanique du combustible	voir fiche 170-1-04

- Expériences globales

SEMISCALE	Dépressurisation-remplissage-renoyage (boucle chauffage électrique 1 MW)	Programme USNRC voir fiche 145-1-13
LOFT	Dépressurisation-remplissage-renoyage (boucle chauffage nucléaire 50 MW)	Programme USNRC voir fiche 145-1-13
LOBI	Dépressurisation-remplissage (boucle chauffage électrique 5 MW)	Programme EURATOM voir fiche 145-1-14
P.K.L.	Remplissage-renoyage	Programme allemand voir fiche 145-1-13

4. ETAT D'AVANCEMENT :

4.1 - Avancement à ce jour

4.1.1 - Opérationnalité

- RELAP 4 MOD 6 : Code implanté sur machine CDC 7600 et CRAY 1 avec programmes de tracés de courbes adaptés au système français.

- TRAC : Implantation sur CDC 7600 des versions PIA, PD2 et PF1.Implantation sur CRAY 1 de la version PF1.

- RELAP 5 : Implantation successivement sur machines CDC sous système SCOPE 2 et NOS-BE des versions mod 0, mod 1 cycle 6 et mod 1 cycle 14. Aucune de ces versions n'a donné encore un fonctionnement totalement satisfaisant

- FRAPT 4 : Implantation sur CDC avec adaptation des programmes de tracé. Couplage avec RELAP 4.

4.1.2 - Qualification-validation

- Ecoulement diphasique en dépressurisation. Interprétation d'essais MOBY DICK et CANON avec le code TRAC PIA puis TRAC PD2. Reprise de cette interprétation avec la version TRAC PF1.

- Transfert de chaleur en dépressurisation (OMEGA) (voir fiche 145-1-11). La qualification sur ce point a été faite avec les codes RELAP 4 Mod 6 et TRAC PD2. Elle a porté sur des essais tube et grappe.

- Renoyage (ERSEC) (voir fiche 145-1-11). Interprétation d'essais tube et grappe avec RELAP 4 Mod 6 et TRAC PD2.

4.1.3 - Evaluation-vérification

Les expériences sur lesquelles ont porté les tâches d'évaluation-vérification sont les expériences SEMISCALE, LOFT, PKL, LOBI.

SEMISCALE (voir fiche 145-1-13)

- Essai S06-3 (problème standard N° 8)	RELAP 4 Mod 5(1971)
- Essai S06-3 (phase renoyage)	RELAP 4 Mod 6(1980)
- Essai S02-6 (petite brèche)	RELAP 4 Mod 6(1979)

LOFT (voir fiche 145-1-13)

- Essai L1-4 (problème standard N° 5)	RELAP 4 Mod 3
- Essai L3-1 (problème standard N° 9)	RELAP 4 Mod 6 FRARELAP (1980)
- Essai L3-6 (problème standard N° 11)	RELAP 4 Mod 6 FRARELAP (1981)
- Essai L2-5 (problème standard N° 12)	RELAP 4 Mod 6(1982)
- Essai L2-3 (dépressurisation)	RELAP 4 Mod 6(1982)
- Essai L3-6 (reprise)	RELAP 4 Mod 6(1982)
- Essai L3-5	RELAP 4 Mod 6(1982)
- Essai L1-5	TRAC PF1 (1982)

P.K.L. (voir fiche 145-1-13)

- Essai K9 (problème standard N° 10)	RELAP 4 Mod 6(1980)
- Reprise essai K9	RELAP 4 Mod 6(1981)
- Essais K5a, K5.3a, K7a, K5.1b, K5.4a	RELAP 4 Mod 6(1982)

LOBI (voir fiche 145-1-14)

- Réalisation calculs préliminaires	RELAP 4 Mod 5 Mod 6 (1980)
- Réalisation du problème LOBI-PREX	RELAP 4 Mod 6(1980)
- Réalisation de calculs prévisionnels	
• Essai B101, B101-M (grosse brèche)	RELAP 4 Mod 6(1980)
• Essai SDSL01 (petite brèche)	RELAP 4 Mod 6 FRARELAP (1980)
• Essai SDSL02 (petite brèche)	FRARELAP (1980)
• Essai B222 (grosse brèche)	RELAP 4 Mod 6(1982)

- Réalisation de calculs d'interprétation

• Essai B 101 (grosse brèche)	RELAP 4 Mod 6 (1981)
• Essai A1-66 (grosse brèche)	RELAP 4 Mod 6 (1981)
• Essai SDSL01	TRAC PIA (1980)
• Essai SDSL03	FRARELAP (1982)

- Participation aux task forces dans le cadre du groupe de travail LOBI-B

4.1.4 - Calculs réacteurs

- Calculs FESSENHEIM grosse brèche -RELAP 4 Mod 5 et Mod 6 (1978, 1979)

- Calcul CP1 grosse brèche - RELAP 4 Mod 6 (1980)

- Etudes de sensibilité sur le calcul CP1 grosse brèche (structures, température dôme, inertie hydraulique coeur, canal chaud, brèches dissymétriques) (1980, 1981).

- Calcul CP1 physique grosse brèche - RELAP 4 Mod 6(1981,1982)

- Etudes de sensibilité complémentaires sur calcul CP1 grosse brèche avec RELAP 4 Mod 6 (1982)

- Calcul CP1 renoyage - RELAP 4 Mod 6 (corrélations FLECHT)

- Calcul CP1 renoyage - RELAP 4 Mod 6 (physique) en cours 1982

- Calcul CP1 grosse brèche dépressurisation-remplissage TRAC PD2

- Calcul CP1 brèche de 7,5cm de diamètre - RELAP 4 Mod 6. Comparaison avec calcul FRARELAP (1980, 1981).

- Calcul CP1 brèche de 5cm de diamètre sans ISHP - RELAP 4 Mod 6 (1981)

- Calcul CP1 brèche de 34,4cm de diamètre avec les options grosse et petite brèches (1981)

- Calcul CP1 petite brèche sans ISHP conduisant à la fusion (1982)

- Calcul CP1 rupture RRA - RELAP 4 Mod 6 (1982)

- Calcul CP1 rupture tubes GV avec ou sans rupture circuit vapeur - RELAP 4 Mod 6 (1982, en cours)

- Calcul 1300 brèche 5cm - RELAP 4 Mod 6 (1982, en cours)

4.2 - Principaux résultats

Grosses brèches - Dépressurisation

Dans l'ensemble des actions de qualification et de vérification le code RELAP 4 Mod 6 présente une fiabilité certaine en dépressurisation grosse brèche. Les résultats sont en général satisfaisants sur l'ensemble des paramètres. Néanmoins, un examen détaillé des résultats laisse apparaître des lacunes sur certaines grandeurs, ce qui semble indiquer l'existence de compensations d'erreurs pour les paramètres les mieux prévus.

Les difficultés de prévision suite aux changements de configuration d'expériences (LOBI par exemple) confirment cette indication et montrent la nécessité de disposer d'une physique plus évoluée. Le code TRAC PD2 qui possède une telle physique voit en fait ses résultats faussés par une vitesse de dépressurisation trop importante. Les premiers résultats avec TRAC PF1 montrent une certaine amélioration mais ne sont, en ce qui nous concerne, pas encore assez nombreux pour conclure.

Grosses brèches - Remplissage

De nombreuses difficultés numériques sont rencontrées pratiquement avec tous les codes, nécessitant souvent de procéder à des artifices comme sur la température d'injection. Les modèles de condensation sont, soit totalement erronés (équilibre thermodynamique dans RELAP 4), soit apparemment mal ajustés (TRAC PD2). Les calculs réacteurs avec TRAC PD2 font apparaître pendant cette phase, des effets 3D importants dans la cuve. Des études supplémentaires sont nécessaires pour évaluer l'importance réelle de ces effets.

Grosses brèches - Renoyage

Des résultats satisfaisants sont obtenus avec les codes RELAP 4 Mod 6 et TRAC PD2 pour des expériences utilisant des simulateurs électriques en configuration grappe et pour des pressions voisines de 3 bars. Tout écart par rapport à ces conditions (configuration tube, pressions plus faibles (1 bar)ou plus fortes (6 bars)conduit à des désaccords très importants entre calculs et expériences. Les effets système propre à cette phase semblent relativement bien prédits par RELAP 4 Mod 6 (résultats PKL).

Petite brèche

Des résultats assez satisfaisants ont été obtenus avec RELAP 4 Mod 6 et FRARELAP. Ces résultats, pour RELAP 4 Mod 6, sont très sensibles au choix des options, ce qui demande pratiquement de déterminer ou de connaître à l'avance les phénomènes qui sont à prédire. Des difficultés et des écarts importants sont dans certains cas rencontrés dans la description des phases d'injection des accumulateurs.

Aucune expérience propre n'a encore été obtenue avec TRAC PF1 dont les modèles devraient avoir des potentialités supérieures.

5. PROCHAINES ETAPES :

5.1 - Opérationnalité

FRAPT 4 : Implantation du code sur CRAY 1

5.2 - Qualification-validation

- RELAP 4 Mod 6
  - Fin de l'analyse des essais grappe 2 ème campagne
- TRAC PF1
  - Fin des calculs MOBY DICK et CANON
  - Reprise des calculs OMEGA et ERSEC effectués précédemment avec TRAC PD2
  - Calcul des autres essais figurant dans la liste du dossier de qualification de CATHARE

5.3 - Evaluation-vérification

LOFT

- Essai L2-5 (problème standard N° 12) renoyage avec	RELAP 4 Mod 6
- Essai L2-3	TRAC PF1
- Essai L2-5	TRAC PF1
- Essai L3-6	TRAC PF1
- Essai L3-5 (étude complémentaire)	RELAP 4 Mod 6

P K L

- Essais "petite brèche"	TRAC PF 1

LOBI

- Révision des matrices d'essais B en particulier des essais petite brèche
- Définition de la matrice d'essais des transitoires spéciaux
- Suite de l'interprétation de l'essai SDSL03
- Interprétation des essais B grosse brèche avec l'espace annulaire réduit
- Calculs prévisionnels des essais A2 et B à réaliser sur LOBI
- Préparation et réalisation du calcul du problème standard N° 18 avec le code TRAC PF1.

5.4 - Calculs réacteurs

- Fin calcul renoyage CP1 (RELAP 4 Mod 6)
- Calcul CP1 grosse brèche avec une schématisation détaillée (TRAC PF1)
- Poursuite des calculs CP1 RTGV et RTGV + RTV (RELAP 4 Mod 6)
- Reprise de ces calculs avec TRAC PF1
- Calcul CP1 petite brèche avec TRAC PF1
- Calcul CP1 ATWS perte d'eau alimentaire avec TRAC PF1
- Calcul 1300 petite brèche avec RELAP 4 Mod 6 (fin)
- Calcul 1300 petite brèche avec TRAC PF1

6. RELATIONS AVEC D'AUTRES ETUDES :

Fiches 145-1-3 à 145-1-8	Thermohydraulique du LOCA programmes expérimentaux
Fiches 145-1-11 à 145-1-15	Qualification-validation, évaluation-vérification. Calculs réacteurs (RELAP-TRAC)
Fiches 142-3	Conséquences sur l'enceinte de confinement
Fiche 170-1-04	PHEBUS .

7 . DOCUMENTS DE REFERENCE :

- Résumé des travaux effectués par la Section d'Etudes des Accidents des Réacteurs à Eau Légère jusqu'au 1er Juin 1979.
  Note technique SEAREL 79/14 (Novembre 1979)

- Résumé des travaux effectués par la Section d'Etudes des Accidents des Réacteurs à Eau Légère du 1er Juin 1979 au 31 Décembre 1980.
  Note technique SEAREL 81/44

- Résumé des travaux effectués par la Section d'Etudes des Accidents des Réacteurs à Eau Légère du 1er Janvier 1981 au 31 Décembre 1981.
  Note technique SEAREL à paraître

- Autres documents : voir fiches citées en références.

145-1-02		1
TITRE DEVELOPPEMENT DU CODE AVANCE CATHARE POUR L'ETUDE DE L'ACCIDENT DE PERTE DE REFRIGERANT (APRP) DANS LES REACTEURS A EAU PRESSURISEE.		*Pays* FRANCE *Organisme directeur* CEA-IPSN EDF/SEPTEN
TITRE en anglais DEVELOPMENT OF AN ADVANCED CODE FOR THE STUDY OF THE LOSS OF COOLANT ACCIDENT IN A PRESSURIZED WATER REACTOR (CATHARE).		*Organisme éxécuteur* CEA-EDF FRAMATOME EQUIPES MIXTES (GRENOBLE) *Responsables* M. BOULOT EDF M. PELCE CEA
Date de démarrage 01.01.77	*Etat actuel* en cours	*Scientifiques* MM. COURTAUD, NIGON (CEA-ST1 M. MEGNIN /FRAMATOME M. REOCREUX/CEA-DSN M. SUREAU /EDF-SEPTEN M. HOUDAYER/EQUIPE MIXTE M. BRUN /EQUIPE MIXTE
Date d'achèvement 31.03.83	*Dernière mise à jour* 05.83	

1 - OBJECTIF GENERAL :

Conception et réalisation d'un code avancé (CATHARE) permettant la transposition au réacteur des essais analytiques et globaux, français et étrangers, pour l'analyse des APRP en support et/ou remplacement des codes de première génération.

Ce développement est la continuation des travaux sur CLYSTERE et POSEIDON.

Une équipe commune EDF/CEA/FRAMATOME a été constituée le 1/9/1979 à GRENOBLE (CEN/G) pour la réalisation des modules.

Une seconde équipe commune a été mise en place le 1/9/1980 pour la réalisation de l'assemblage.

2 - OBJECTIFS PARTICULIERS :

2.1. - Ecriture de modèles physiques

Ces modèles décrivent l'ensemble des phénomènes intervenant au cours des APRP.

- Ecoulement 1D-2φ

• Avec - échanges thermiques en parois, frottements
  - échanges entre phases (masse, énergie, frottements)

• Pour tous les types d'écoulement (dispersés, stratifiés, contre-courant, flooding, front de taux de vide, etc...)

- Pour tous les éléments axiaux; entre autres :

  - approche de brèche (calcul de débits critiques)

  - rattrapage des déséquilibres dûs aux injections de secours

  - écoulements et transferts :
    dans le coeur en dépressurisation
    dans le coeur en renoyage
    dans le primaire des G.V.

- Conduction thermique radiale dans les tuyaux et structures couplée au modèle d'écoulement, avec en plus pour le combustible :

  - thermomécanique des crayons (oxydation, gonflement, rupture)

  - conduction 2D et échange au voisinage du front de trempe (code PSCHITT)

- Volume OD-2φ (entrainement, désentrainement, niveau, fall back piquages sur volumes)

- Pompe OD et Pompe 1D-2φ

- Downcomer 2D-2φ

- Jonction 2φ pour injection de secours et piquage pressuriseur avec séparation de phase éventuelle.

- Singularités

- Neutronique

2.2. - Validation des modèles physiques

Tous les modèles sont établis et validés par les expériences françaises (voir liste au paragraphe 3) ou étrangères.

Les modèles font la synthèse entre les différents essais.

2.3. - Analyse numérique

Ces études ont pour but d'optimiser la précision, la fiabilité et la rapidité des calculs et la souplesse du code.

Elles comportent :

- La mise au point d'une méthode numérique efficace pour les modules (discrétisation du module 1D–2φ essentiellement)
- Le test de la méthode numérique d'assemblage
- La recherche d'algorithmes pour l'optimisation au cours du transitoire des pas de temps et d'espace et des tests de convergence.

2.4. - Réalisation des modules

Les modules recensés sont les suivants :

- module de base (axial) avec ses différentes options :
  - a) version semi–implicite et implicite
  - b) avec ou sans front de trempe
  - c) avec combustible ou parois simples, ou sans parois
  - d) option générateur de vapeur (secondaire ponctuel)
- module volume OD avec ses différentes options (ou versions) : plenum supérieur, inférieur, couvercle, fond de cuve, pressuriseur, boîte à eau GV, accumulateur, etc...
- module "Té"
- module pompe nodale avec son moteur
- module pompe 1D 2φ avec son moteur
- downcomer 2D
- module GV 2x1D primaire et secondaire axiaux (éventuellement)
- modules divers :
  - modules conditions limites (pour circuits ouverts)
  - module singularité
  - modules dérivés des singularités et/ou des conditions limites (brèche, vanne, soupapes, clapets, jonctions diverses)
- module neutronique et puissance résiduelle
- modules asservissements et contrôle commande

2.5. - Conception et réalisation de l'assemblage

- Conception et réalisation du système informatique :

  - langage, Entrées-Sorties
  - gestion mémoire
  - gestion des enchainements

Ce système conduit à un code modulaire souple.

Les assemblages sont ou seront réalisés pour les calculs réacteurs et pour interpréter les expériences globales : OMEGA, ERSEC Système, PKL, SEMISCALE Renoyage, LOFT, SEMISCALE, LOBI,...

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)
OMEGA	Ecoulement et transfert de chaleur en dépressurisation	(voir fiche 145-1-04)
ERSEC	Ecoulement et transfert de chaleur en renoyage. Modèle de front de trempe	
PERICLES	Renoyage incluant le coeur et les effets de système	(voir fiche 145-1-09)
EPIS	Injection de secours	
EVA, EPOPEE	Pompe en diphasique	(EVA programme FRA-CEA) (EPOPEE programme EDF)
PATRICIA I et II	Comportement des générateurs de vapeur en conditions accidentelles	(voir fiche 145-1-10)
ECTHOR	Stratification dans les boucles	
PIERO	Fonds de cuve	(voir fiche 145-1-17)
EDGAR	Thermomécanique de la gaine du combustible	(voir fiche 145-2-02)

PHEBUS	Thermomécanique du combustible	(voir fiche 170-1-04)
REBECA	Ecoulement diphasique air-eau-vapeur entre les casemates	(voir fiche 145-4-02)
ECOTRA	Condensation dans les enceintes	(voir fiche 142-3-01)

4 - ETAT D'AVANCEMENT :

4.1. - Avancement à ce jour

4.1.1. - Système informatique :

L'assemblage a été défini et validé.

4.1.2. - Méthode numérique :

L'analyse numérique finalement retenue pour le modèle d'écoulement 1D 2φ est une discrétisation à mailles décalées/cellule donneuse (méthode ICE) avec deux versions compatibles dans l'assemblage (semi-implicite et implicite). Actuellement, seule la version totalement implicite est utilisée parce que s'étant avérée plus avantageuse. Les autres modules ont une discrétisation implicite.

4.1.3. - Réalisation des modules :

Module de base (écoulements axiaux)

Ce module comprend une grille complète de lois constitutives, à savoir :

- des lois de transfert de masse, d'impulsion et d'énergie entre phases.
- des corrélations d'échange de chaleur en dépressurisation et en renoyage.
- des lois de frottement à la paroi.

Autres modules

Les modules suivants ont été réalisés et testés :

- parois
- combustible
- GV
- pompe OD
- volumes
- té
- neutronique
- accumulateur

4.1.4. - Mise au point-optimisation :

- Une version complète du code (CATHARE 0) interne aux équipes mixtes a été réalisée.
- Avec la version CATHARE 0 deux calculs réacteurs ont été entrepris :
  - un calcul grosse brèche (ADR) comprenant les phases de dépressurisation, remplissage et renoyage.
  - un calcul petite brèche (brèche de 3").

4.2. - Principaux résultats

- Un jeu unique et cohérent de lois de transfert a été finalisé dans la révision 2 des grilles.
- Les reconstitutions partielles avec cette révision 2 des essais MOBY DICK, SUPER MOBY DICK, CANON, SUPER CANON, MARVIKEN, DADINE, ECTHOR, REBECA, OMEGA, ERSEC, donnent des résultats qualitativement corrects.
- La réalisation des deux calculs réacteurs actuellement en cours a permis d'apporter un certain nombre de corrections et d'améliorations du code notamment sur :
  - les modules pompe et té
  - les lois aux jonctions entre modules.

5 - PROCHAINES ETAPES :

Les travaux de développement CATHARE sont orientés dans quatre directions :

5.1. - Mise au point-optimisation

La fin de la réalisation des deux calculs réacteurs doit conduire à la première version opérationnelle de CATHARE (CATHARE 1).

Cette version est prévue pour fin 1983.

5.2. - Amélioration des lois de transfert

Le dépouillement du programme expérimental doit conduire à de nouvelles révisions des grilles de lois de transfert qui seront incorporées à CATHARE 2.

5.3. - Qualification et vérification de CATHARE 1

- Un contrôle systématique de CATHARE 1 version 1 sur un ensemble représentatif d'essais du programme expérimental analytique doit conduire à un dossier de qualification du code. Ce dossier devrait être réalisé à la livraison du code.

- CATHARE 1 dans sa version figée doit ensuite être vérifié sur un ensemble représentatif d'essais sur boucle système (LOFT, LOBI, PKL).

5.4. - Développements CATHARE 2

Ces développements doivent se faire selon deux directions :

- amélioration des méthodes numériques
- Ecriture de nouveaux modules : downcomer bidimensionnel, pompe axiale, GV axial, introduction des incondensables.

6 - RELATIONS AVEC D'AUTRES ETUDES :

- Calculs des accidents de refroidissement des réacteurs à eau. Programme général coordonné de mise au point des codes. (fiche 145-1-01).

- Programmes expérimentaux (voir § 3).

7 - DOCUMENTS DE REFERENCE :

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Rapport DSN 167 e - Presentation at the 5th Water Reactor Safety Research Information Meeting. Washington 1977

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Calculation of single tube reflooding with the HEXECO code
Analysis of ERSEC experiment.
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OECD - 3rd Transient Two-Phase Flow Specialists Meeting
Pasadena - March 1981

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A Synthesis of qualification of the advanced safety code CATHARE, based on analytical experiments
European Two-Phase Flow Group Meeting
June 2-4, 1982

J.C. ROUSSEAU, G. HOUDAYER
Advanced Safety Code CATHARE
A Synthesis of qualification on analytical experiments
International Topical Meeting on Nuclear Reactor Thermohydraulics
Santa-Barbara Jan 11-14, 1983

145-1-13			1
TITRE EVALUATION - VERIFICATION RELAP-TRAC CALCULS SEMISCALE, LOFT, PKL.			*Pays* FRANCE *Organisme directeur* CEA- IPSN - CEA/DSN
TITRE en anglais EVALUATION - VERIFICATION OF RELAP AND TRAC CODES. SEMISCALE, LOFT, PKL, CALCULATIONS.			*Organisme éxécuteur* CEA/DSN/SEAREL (FAR) CEA/DRE/STRE (CAD) /SERMA (SAC) *Responsables* M. REOCREUX /SEAREL M. GINIER /STRE
Date de démarrage 1978		*Etat actuel* en cours	*Scientifiques* M. LERIDON /STRE M. THOMAS /SERMA M. POCHARD /SEAREL
Date d'achèvement		*Dernière mise à jour* 1/ 1983	

1. OBJECTIF GENERAL :

Réaliser les tâches d'évaluation-vérification des codes de calculs disponibles, actuellement RELAP et TRAC, sur les expériences étrangères SEMISCALE, LOFT (INEL-IDAHO FALLS- USA) et PKL (KWU - ERLANGEN- RFA).

2. OBJECTIFS PARTICULIERS :

2.1 - Réalisation de calculs dans le cadre des problèmes standards OCDE concernant ces expériences.

2.2 - Réalisation de calculs prévisionnels et d'interprétation des expériences SEMISCALE, LOFT, PKL.

2.3 - Tirer des calculs précédents des enseignements pour les calculs réacteurs.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME:

Les installations utilisées dans cette action d'évaluation-vérification sont les suivantes :

SEMISCALE MOD 1

Cette installation située sur le Centre de l' INEL à IDAHO FALLS (USA) est destinée à étudier les phénomènes physiques de la dépressurisation, du remplissage et du renoyage. Elle simule, dans sa configuration MOD 1, un réacteur 4 boucles au moyen d'une boucle et demie, c'est à dire une boucle avec éléments actifs (représentant les boucles non rompues) et une boucle avec éléments passifs (boucle rompue).
Le coeur est représenté par 40 barreaux à chauffage électrique de 1,68 m de haut.

LOFT

Situé également sur le Centre de l'INEL (IDAHO FALLS - USA), LOFT est un réacteur nucléaire de 55 MW. une boucle active simule 3 boucles de PWR et la boucle rompue est passive. Le coeur de 1,68 m de haut, contient 1300 crayons combustibles.

PKL

Situé sur le Centre KWU à ERLANGEN (RFA), cette installation est destinée à étudier les phénomènes physiques du remplissage et du renoyage au cours d'un LOCA. Elle simule un réacteur 4 boucles à l'échelle 1/134 pour les volumes et la puissance; 340 barreaux de 3,9 m de haut, à chauffage électrique, permettent de représenter la puissance résiduelle du coeur. La boucle est conçue pour permettre des injections branche chaude, branche froide et directement dans la zone annulaire.
L'installation a été modifiée pour permettre l'étude des accidents petites brèches : adjonction d'un pressuriseur et d'un dôme vapeur G.V., modification de l'instrumentation.
Les essais petites brèches sont sous forme soit de transitoires, soit d'une succession d'états permanents avec une masse d'eau primaire décroissante.

4. ETAT D'AVANCEMENT

4.1 - Avancement à ce jour :

- Rappel : problème standard N° 5 LOFT L 1-4 (code RELAP 4 Mod 3).

Ont été réalisés courant 79 :

- Problème standard N° 8 SEMISCALE S 06-3 (code RELAP 4 Mod 5).
- Problème standard N° 4 SEMISCALE S 02-6 (code RELAP 4 Mod 6).

Ont été réalisés courant 80 :

- Problème standard N° 9 LOFT L 3.1 (codes RELAP 4 Mod 6, FRARELAP)
- Essai de renoyage SEMISCALE S 06-3
- Problème standard N° 10 PKL K 9 (code RELAP 4 Mod 6)

Ont été réalisés courant 81 :

- Problème standard N° 11 LOFT L 3.6 (codes RELAP 4 Mod 6, FRARELAP)

Ont été réalisés courant 82 :

- Problème standard N° 12 LOFT L 2.5 (code RELAP 4 Mod 6)

- Calcul des essais LOFT :
  - L 2.3 dépressurisation (code RELAP 4 Mod 6)
  - reprise L 3.6 (code RELAP 4 Mod 6)
  - L 3.5 (code RELAP 4 Mod 6)
  - L 1.5 (code TRAC PF 1)

- Calcul des essais PKL :
  - reprise essai K 9 (Problème standard N° 10)
  - K 5a, K 5.3a, K 7a, K 5.1b, K 5.4a (code RELAP 4 Mod 6)

4.2 - Principaux résultats :

4.2.1 - Etude des problèmes standards :

Problème standard N° 8 : Ce problème correspond à l'essai S 06-3 du programme SEMISCALE Mod 1.
L'essai S 06-3 est une expérience de dépressurisation par une rupture guillotine branche froide suivie de remplissage et de renoyage. La puissance linéique maximum est de 394 W/cm. Cet essai devait servir de répétition et de préparation de l'essai LOFT L 2.3.
La modélisation de l'expérience a été faite au moyen de 40 volumes, 44 jonctions et 14 zones d'échange de chaleur. Le coeur a été représenté par un canal hydraulique unique comportant 2 types de barreaux électriques : chauds et moyens. Le débit à la brèche a été calculé par l'équation des moments modifiée en sous-critique et par le modèle homogène équilibré en régime saturé, la transition entre les 2 modèles se faisant pour une qualité de 2 %.Des coefficients de décharge de 0,865 en sous-saturé et 0,7 en saturé ont été utilisés. Des modèles de montée de bulles ont été utilisés dansle plenum inférieur, le secondaire du générateur de vapeur, le pressuriseur et l'accumulateur.
Les résultats ont montré que la baisse de pression dans le système était trop rapide; ce phénomène peut être partiellement expliqué par une prédiction de flux critique trop précoce (un des points faibles de RELAP). La température maximum de gaine a été correctement calculée (en temps et en amplitude), cependant des anomalies, probablement dûes au découpage, ont été observées dans le bas du canal.

Essai de renoyage S 06-3

Les données utilisées pour le calcul de la phase de dépressurisation ont été reprises pour être adaptées à la phase renoyage.

Deux types de difficultés ont été rencontrés :

1) taille mémoire insuffisante dûe à l'utilisation de deux canaux en parallèle avec maillage glissant pour la représentation du coeur. Cette difficulté a été surmontée par une nouvelle segmentation du programme.

2) RELAP 4 Mod 6 n'étant pas capable de calculer la phase remplissage, les conditions initiales de la phase renoyage ont été fixées d'après les résultats expérimentaux à l'instant où l'eau commence à rentrer dans le coeur. A cet instant la pression dans la boucle est encore nettement supérieure à la pression de l'enceinte (P boucle = 4,5 bars, P enceinte = 2,9 bars). Cette initialisation a conduit à des oscillations de débit non amorties qui à 5 secondes ont bloqué le calcul. Cette expérience nous a permis de nous rendre compte de l'incapacité de RELAP 4 Mod 6 à démarrer un calcul renoyage dans des conditions purement expérimentales; une adaptation est nécessaire pour partir d'un état tout à fait stable.

<u>Problème standard N° 4</u> : Cet ancien problème standard (1976), correspondant à la seule expérience petite brèche disponible à la mi-79, a été calculé avec RELAP 4 Mod 6.
L'essai SEMISCALE S 02-6 est une expérience brèche intermédiaire (6 %) sur la branche froide, correspondant environ à une brèche de 7" sur un réacteur. L'accident se déroule sur une période de 400 s.
Avec un découpage en 31 volumes et 40 jonctions, deux calculs ont été réalisés :

1er calcul :

<u>Hypothèse</u> - Utilisation du modèle de Wilson dans les volumes : coeur, primaire des générateurs de vapeur, branche en U, espace annulaire.

<u>Résultats</u> - . Dénoyage trop important du coeur
. Température de gaine trop élevée
. Présence d'un bouchon liquide au bas des épingles du G.V.
. Importance des échanges thermiques avec les structures.

2ème calcul :

<u>Hypothèses modifiées</u> :
. Modèle homogène dans la partie ascendante des tubes de G.V.
. Glissement dans les jonctions des volumes correspondants.

Résultats - . Absence de dénoyage du coeur (expérimentalement un dénoyage partiel localisé a été constaté)
. Température de gaine dans le domaine expérimental.
. Importance des échanges thermiques avec les structures.

Les résultats des calculs montrent l'importance du choix des modèles sur les résultats des futurs calculs réacteurs petite brèche.

Problème standard N° 9 : L'essai LOFT L 3.1 est une expérience de dépressurisation petite brèche dont la taille équivaut à une brèche de 10 cm de diamètre sur un réacteur.
Un calcul a été réalisé avec le code RELAP 4 Mod 6 en choisissant un découpage en 31 volumes et 40 jonctions. Le modèle de Wilson a été utilisé dans l'ensemble plenum inférieur, coeur, plenum supérieur, les parties descendantes des G.V. et la branche intermédiaire. Les autres volumes sont homogènes.

Les résultats ont montré un bon accord concernant :

- la loi de décroissance de la pression
- le temps de vidange du pressuriseur
- le temps d'injection de l'accumulateur (640 s, contre 633,6 s).

A partir de ce moment, le calcul a dû être poursuivi en imposant le débit expérimental injecté.

Dans le cadre de la fiche d'étude et de développement N° 117 entre FRAMATOME et C.E.A., un calcul a été réalisé par FRAMATOME avec le code FRARELAP.
Les résultats trouvés sont voisins. Le calcul de la phase d'injection des accumulateurs a pu être mené à bien en calculant le débit injecté.

Problème standard N° 10 : Il correspond à l'essai K 9 de la série IB du programme PKL. L'essai K 9 est une expérience de remplissage-renoyage avec injection dans les branches froides et l'espace annulaire seulement. Le profil radial de puissance est plat.
La modélisation pour RELAP 4 Mod 6 comprend 29 volumes, 31 jonctions et 25 zones d'échange de chaleur. Le coeur est représenté par un volume unique contenant 12 zones d'échanges; un maillage thermique local plus fin se déplace avec le front de trempe. Les différentes options utilisées sont celles recommandées par EG & G. Trois calculs ont été effectués:

1 er Calcul : L'injection d'eau sous saturée en trois points distincts conduit à de fortes oscillations numériques qui obligent à injecter de l'eau à saturation pendant les premiers instants du calcul. Les résultats montrent de ce fait des températures de retournement et des temps de trempe supérieurs aux valeurs expérimentales.

2 ème Calcul :L'injection d'eau sous saturée, en un seul point (haut de l'espace annulaire) améliore sensiblement l'accord des résultats avec l'expérience, pour les températures de retournement comme pour les temps de trempe.

3 ème Calcul :Ce calcul utilise la corrélation globale FLECHT du code RELAP 4 FLOOD. La phase de remplissage ne pouvant être calculée à l'aide de cette version, le calcul commence à l'instant où le niveau d'eau affleure le bas de la zone chauffante. Les premiers résultats montrant d'importantes différences avec l'expérience, le calcul a été repris en tenant compte de la sous saturation à l'entrée du coeur et en corrigeant la puissance linéique du coeur. Les résultats obtenus sont alors beaucoup plus proches de ceux de l'expérience tout en restant conservatifs.

Problème standard N° 11 : L'essai LOFT L 3.6 est une expérience de dépressurisation petite brèche (2,5 % en branche froide de la boucle intacte) avec pompes en fonctionnement. L'option du fluide homogène a donc été retenue pour la réalisation d'un calcul RELAP 4 Mod 6.
Les résultats de calcul montrent un bon accord avec l'expérience (pression et température du fluide). Un léger désaccord subsiste sur le bilan final de masse. Dans le cadre d'une révision de la fiche 117 un calcul a été réalisé par FRAMATOME avec une version de FRARELAP possédant un modèle de glissement plus général. Les résultats trouvés sont également en bon accord avec l'expérience.
Une étude complémentaire avec la modélisation et les principales options du premier calcul, a été réalisée (code RELAP 4 Mod 6). Ses particularités sont les suivantes :

- Les caractéristiques double phase des pompes sont tirées de l'essai L 3.6 lui-même.
- Utilisation d'un nouveau modèle de G.V. avec séparateur
- Prise en compte du scénario réel de l'expérience (en particulier fermeture vanne vapeur en 7 secondes, fuite de vapeur, ouverture intempestive à 100 secondes).

Résultats Une meilleure représentation de la séquence des évènements est ainsi obtenue, ce qui améliore les résultats. Deux calculs ont été réalisés pour retrouver le débit brèche, surestimé durant les 200 premières secondes :
- utilisation d'un coefficient de contraction à la brèche
- particularisation du volume amont brèche , en mettant un coefficient de perte de charge adéquat.

Seule cette 2 ème méthode permet d'approcher correctement le débit brèche expérimental, sans trop modifier la pente de dépressurisation pendant les 500 premières secondes de l'accident Mais la dépressurisation devient beaucoup trop lente vers la fin du transitoire.

Problème standard N° 12 : L'essai LOFT L 2.5 est une expérience de perte de réfrigérant par grosse brèche (200 %) en branche froide, avec arrêt des pompes et retard de l'ISHP.
La schématisation du circuit comprend 55 volumes, 64 jonctions et 32 structures. Ses particularités sont les suivantes :

- le coeur est divisé en 3 volumes
- un seul canal moyen dans le coeur (pas de canal chaud)
- le combustible est représenté par 9 structures.

Le modèle de débit critique est le modèle de débit critique homogène équilibré (HEM).
Seul le calcul de la phase dépressurisation a été réalisé.

Résultats Un bon accord est obtenu avec l'expérience pour la courbe de dépressurisation. L'injection accumulateur se produit au même instant, (17 secondes) que dans l'expérience.
On surestime par contre de 0 à 12 secondes la densité du fluide de la branche froide de la boucle rompue.
Enfin, il n'est pas apparu de remouillage expérimental comme dans L 2.3 dans la partie centrale de la 1ère région du coeur.
Le calcul du comportement moyen du coeur ne fait pas apparaître de remouillage prématuré.

4.2.2 - Autres études : LOFT :

Essai LOFT L 1.5 : Cet essai est une expérience de perte de réfrigérant par brèche double de 100 % en branche froide de la boucle rompue. Il s'agit du premier essai avec coeur nucléaire (grappes hautes), mais avec des conditions initiales isothermes (555K). Les pompes restent en fonctionnement jusqu'à 70 S.
Même schématisation du circuit avec RELAP 4 Mod 6 que pour le calcul L 2.5 mais avec les particularités suivantes :

- Le " downcomer" de la cuve est moins finement découpé
- Trois structures (une par volume coeur) représentent le combustible

Résultats : Le calcul de cet essai,préliminaire à ceux des essais L 2.3 et L 2.5, a permis de valider, et la schématisation du circuit, et les options choisies pour les futurs calculs grosse brèche LOFT (L 2.3 et L 2.5).
Un bon accord général avec l'expérience a pu être observé dans le domaine hydraulique et thermique.
Une schématisation grosse brèche pour les calculs TRAC PF 1 a été réalisée (fiche 150 CEA-FRA).
Elle comprend 144 mailles dont 44 mailles dans la cuve.
Une pompe double représente les 2 pompes.
Le modèle de débit critique disponible dans TRAC a été utilisé à la brèche.

Résultats : Le transitoire de pression est correctement prédit et le débit brèche est bien reconstitué.
Cependant des difficultés de modélisation de la cuve subsistent. Elles concernent la schématisation du by-pass du downcomer et la modélisation des échanges paroi fluide qui s'y produisent. Ces difficultés empêchent le bon déroulement du calcul après l'initiation de l'injection d'eau de secours.
Il faut enfin noter que le coût de ce calcul est très inférieur au coût du calcul réalisé avec RELAP 4 Mod 6.

Essai LOFT L 2.3 : Cet essai est une expérience de perte de réfrigérant par brèche double (100 %) en branche froide de la boucle rompue, avec pompes en fonctionnement et ISHP normale.
La puissance linéaire maximum est la même que celle de l'essai LOFT L 2.5.
La schématisation du circuit avec le code RELAP 4 Mod 6 et les principales options sont identiques à celles du calcul de l'essai LOFT L 2.5.

Résultats : Un bon accord avec l'expérience est obtenu pour les températures de fluide et les pressions. L'accord est cependant moins bon en ce qui concerne les températures de gaine dont le premier pic est sous estimé. Cela peut s'expliquer par l'absence d'un canal chaud dans cette modélisation.
On retrouve cependant par le calcul, le remouillage précoce à 6 secondes, qui est apparu dans cet essai .

Essai LOFT L 3.5 : Cet essai est une expérience de dépressurisation petite brèche (2,5 % sur la branche froide de la boucle intacte) avec les pompes arrêtées au début de l'accident.
La comparaison des essais L 3.5 et L 3.6 doit permettre d'étudier l'influence des pompes sur la diminution de masse du circuit primaire.
La schématisation du circuit et les options du calcul sont identiques à celles du premier calcul L 3.6 (code RELAP 4 Mod 6).

Résultats Les résultats de ce calcul et de l'expérience sont très éloignés, en particulier le débit brèche est très largement surestimé pendant les 200 premières secondes. La difficulté provient du fait que la stratification du fluide qui se produit dans la branche froide où se situe la brèche, n'est pas modélisable à l'aide de RELAP 4.

4.2.3 - Autres études : PKL :

Calcul des essais PKL I A (brèche branche froide) avec injections branches froides seulement (code RELAP 4 Mod 6 option REFLOOD).

L'essai K 5a a été étudié avec les mêmes options que le deuxième calcul K 9 (problème standard N° 10.).

Par rapport à l'essai K 5a dont les principales caractéristiques sont :

- eau d'injection à 35 °C
- profil radial de puissance plat, facteur axial de point chaud 1.16
- pression primaire initiale 4 bar
- secondaires G.V. 55 bar, niveau entre 6 et 8 m

les différents essais calculés se distinguent par :

- des résistances des boucles plus faibles (K 5.3 a)
- un débit d'injection diminué (K 7 a)
- un facteur total de point chaud augmenté : 1.69 (K 5.1b)
- la suppression du tube vapeur du downcomer (K 5.4 a)

Pour tous ces essais étudiés avec la même schématisation et les mêmes options, les résultats calculés sont en bon accord avec l'expérience, que ce soit pour le profil de température, la température de retournement ou le temps de trempe.

Etude de sensibilité au coefficient d'entraînement

Des calculs avec des coefficients d'entraînement (fraction de liquide entraîné par la vapeur) de 0.65, 0.75 et 0.80 ont été effectués. Il apparaît que ce coefficient a peu d'influence sur l'évolution des températures de paroi.

5. PROCHAINES ETAPES :

5.1 - Etudes LOFT

Problème standard N° 12: LOFT L 2.5 : phase renoyage avec RELAP 4 Mod 6

. Calculs grosses brèches LOFT avec TRAC PF 1 :
  L 2.3
  L 2.5
. Calculs petites brèches LOFT avec TRAC PF 1 :
  L 3.6
. Calcul petite brèche LOFT avec RELAP 4 Mod 6
  L 3.5 (étude complémentaire)
. Calculs essais LOFT avec FRARELAP :
  L 3.7
  L 5.1

5.2 - Etudes PKL :

. Calcul d'essais petites brèches PKL, à l'aide de TRAC PF 1
- Essais sous forme de suites de permanents avec masse d'eau primaire décroissante (1 seule boucle reliée à la cuve)
- Essais transitoires (3 boucles)

6. RELATIONS AVEC D'AUTRES ETUDES :

Cette étude est en relation avec les études de qualification-validation des codes (fiche 145-1-11), les autres tâches d'évaluation-vérification (fiches 145-1-14 et 15)

7. DOCUMENTS DE REFERENCE :

- CSNI Standard Problem N° 5
  RELAP 4 (Mod) Input data and results
  A. FORGE

- Problème standard CSNI N° 8
  Analyse thermohydraulique d'un essai de perte de réfrigérant primaire sur la boucle SEMISCALE
  P. AUJOLLET Note technique DRE/STRE/LTA 79/191

- LOFT Nuclear experiment L 3.1; thermohydraulic analysis
  P. AUJOLLET DRE/STRE/LTA 80/278

- LOFT small break test L 3.1
  X.SIRETA FRAMATOME TP/CT/DC 478

- LOFT L 3.6 FRARELAP
  P. GUILLERMARD FRAMATOME EP/TT/DC 133

- LOFT nuclear experiment L 3.6; thermohydraulic analysis
  Y. MACHETEAU, D. MENESSIER DRE/SERMA T-1455

- Etude thermohydraulique de l'essai LOFT L 1.5
  Y. MACHETEAU DRE/SERMA T-536

- Etude thermohydraulique de l'essai LOFT L 2.3
  Y. MACHETEAU DRE/SERMA T-547

- LOFT nuclear experiment L 2.5
  Blind calculation on international standard problem (ISP) 13
  Y. MACHETEAU DRE/SERMA T-1521

- LOFT experience L 2.5
  Analyse thermohydraulique (problème standard ISP 13)
  Y. MACHETEAU DRE/SERMA T-548

- LOFT L 3.6
  Analyse thermohydraulique - Etude de sensibilité
  D. MENESSIER DEMT/SERMA T-558

- Analyse d'une expérience de renoyage PKL avec RELAP 4 Mod 6
  N. TELLIER Note technique SEAREL 80/ 22

- Nouvelle analyse d'une expérience de renoyage PKL au moyen du code RELAP 4 Mod 6
  I. SZABO, N. TELLIER DRE/STRE/LTA 81/381

- Calcul des essais grosse brèche PKL I avec RELAP 4 Mod 6
  Analyse des phases de remplissage et de renoyage dans le cas de l'injection d'eau de secours uniquement en branches froides
  I. SZABO NT DRE/STRE/LTA 82/486

- Calculs de l'essai PKL K 9 à l'aide de RELAP 4 FLOOD

  I. SZABO NT DRE/STRE/LTA à paraître

145-1-15			1
TITRE EVALUATION VERIFICATION PROGRAMME EDITH (PETITE BRECHE)			*Pays* FRANCE *Organisme directeur* CEA/ IPSN
TITRE en anglais EVALUATION VERIFICATION EDITH PROJECT (SMALL BREAK)			*Organisme éxécuteur* CEA/DRE/STRE TECHNICATOME *Responsables* M. GINIER - STRE M. FAJEAU - TA
Date de démarrage 06/ 1979	*Etat actuel* en cours		*Scientifiques* M. LERIDON - STRE M. CHAIX - TA
Date d'achèvement 30.12.1983	*Dernière mise à jour* 01/1983		

1. OBJECTIF GENERAL

Etude de la vidange d'une boucle pressurisée lors d'un accident petite brèche au pressuriseur.

2. OBJECTIFS PARTICULIERS

2.1 - Etude de l'influence du scénario de fonctionnement de la pompe

2.2 - Interprétation des résultats avec le code RELAP 4 Mod 6.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

Boucle EDITH (TA/CADARACHE).

4. ETAT D'AVANCEMENT

1) Avancement à ce jour

Objectif 2.1 : trois essais préliminaires ont été réalisés mi 80 dont un représentatif des phénomènes étudiés:
l'essai TMI 003

Des calculs de scénario de l'essai TMI 004 ont été réalisés.

Objectif 2.2 : des calculs d'interprétation de l'essai TMI 003 ont été réalisés avec RELAP 4 Mod 6 en 81.

2) Principaux résultats

L'essai TMI 003 a mis en évidence le remplissage du pressuriseur. La première interprétation de TMI 003 réalisée avec RELAP 4 Mod 5 a permis de montrer que :

- La taille de brèche de 10 mm était bien adaptée au problème étudié
- Une amélioration de la mesure de pression secondaire est nécessaire
- Une mesure du taux de vide à l'amont de la brèche est indispensable.

Diverses études de sensibilité ont été réalisées, pour le même essai, avec le code RELAP 4 Mod 6, montrant notamment, l'importance du by pass dans la cuve.

Actuellement la modélisation retenue est celle des volumes homogènes (hors pressuriseur) avec le modèle de MOODY à la brèche.
Une étude de sensibilité au temps d'ouverture de brèche et d'arrêt pompe a permis de déterminer le scénario de l'essai TMI 004.

L'essai n'a pu avoir lieu par suite de programmes prioritaires sur EDITH et de la visite décennale de l'installation.

5. PROCHAINES ETAPES

1. Déroulement du programme des essais courant 83 (limité désormais à un essai)

2. Interprétation de cet essai avec RELAP 4 Mod 6.

6. RELATIONS AVEC D'AUTRES ESSAIS

Cette étude rentre dans le cadre général des opérations d'évaluation vérification des codes dans le domaine des petites brèches.

7. DOCUMENTS DE REFERENCE

- Essais TMI sur EDITH
  P. BOUCHARD TA/CAD/SET 80/757

- Expérience de dépressurisation TMI 003 - Première comparaison avec le code RELAP 4 Mod 5
  T. TASTE DRE/STRE/LTA 80/314

145.1.17	1
TITRE NOUVELLES EXPERIENCES EFFETS SEPARES SUPPORT DU CODE CATHARE	*Pays* FRANCE
	Organisme directeur CEA - DSN EDF - SEPTEN
TITRE en anglais NEW SEPARATE EFFECTS TEST IN SUPPORT TO CATHARE CODE	*Organisme éxécuteur* CEA - DRE - STT
	Responsables M. COURTAUD STT

Date de démarrage	1981	*Etat actuel*	en cours	*Scientifiques* M. GULLY STT/SETRE
Date d'achèvement	1983	*Dernière mise à jour*	01.1983	

1. OBJECTIF GENERAL

Réalisation d'expériences à effets séparés destinées à la qualification de modèles ou de modules physiques développés pour le code CATHARE.

2. OBJECTIFS PARTICULIERS

2.1 - Etude de la décompression d'un canal vertical en vue de l'ajustement des lois d'interaction entre liquide et vapeur (expérience CANON VERTICAL).

2.2 - Etude sur maquette (expérience PIERO) de la séparation des phases dans le fond de cuve d'un réacteur PWR.
Etude des écoulements à contre courant dans l'espace annulaire.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

3.1 - Canon vertical

3.1.1. - Expérience

Cette expérience permet la décompression d'un canal vertical Ø 100 mm rempli d'eau chaude sous pression (300° C, 150 bars) et muni d'un orifice à sa partie supérieure. Le dispositif de mise à l'atmosphère est constitué d'une vanne rapide qu'il est possible de refermer au cours de la décompression.
Un soin particulier a été pris pour la mise en oeuvre de la mesure de grandeurs importantes telles la masse restante, le niveau, le taux de vide.

3.1.2. - Programme

Etude et réalisation du dispositif experimental	1981
Campagne d'essais :	1982

Etude de l'influence des deux paramètres

- diamètre de brèche Ø 3 à Ø 15 mm
- température initiale de l'eau dans le canal vertical 230 à 320° C

Compte rendu d'essais	1982

3.2 - PIERO :

3.2.1. - Expérience

L'expérience PIERO comprend une section d'essais simulant à l'échelle 1/4 l'écoulement dans le fond de cuve d'un réacteur en fin de décompression accidentelle. Par ailleurs, l'espace annulaire est également simulé. Les fluides utilisés sont l'eau et l'air à la pression atmosphérique.

3.2.2. - Programme

Etude et réalisation	1982
Campagne d'essais	1982/83
Analyse des résultats - Rapport	1983

4. ETAT D'AVANCEMENT

Avancement à ce jour

Le programme canon vertical est terminé.

La première campagne d'essai PIERO est achevée.

Principaux résultats

Objectif 2.1 : Les essais de décompression CANON VERTICAL ont montré l'existence de deux phases dans le transitoire. La première, inférieure à 10s est caractérisée par une ébullition en masse dans la totalité du canal, avec un débit important à la brèche. La seconde phase est caractérisée par un niveau établi dont la vitesse de descente est fonction de la température initiale et du diamètre de brèche.

Objectif 2.2 : Les essais de séparation des phases dans le fond de cuve ont été réalisés et sont en cours d'analyse.

5. PROCHAINES ETAPES

Objectif 2.1 : programme achevé

Objectif 2.2 : - La deuxième campagne d'essais comportant l'étude de contre-courants dans l'espace annulaire sera réalisée au premier trimestre 1983.

- L'analyse des résultats et la publication d'un rapport seront effectués à la fin du premier semestre 1983.

6. RELATIONS AVEC D'AUTRES ETUDES

Cette étude est en relation avec le programme général coordonné de mise au point des codes (fiche 145-1-01) et avec le développement du code avancé CATHARE pour l'étude de l'accident de perte de réfrigérant (APRP) (fiche 145-1-02).

7. DOCUMENTS DE REFERENCE

Expérience canon vertical
Décompression lente d'une capacité tubulaire verticale par une brèche en partie supérieure
PH. GULLY, JC. BLANC Septembre 1982 - Note TT SETRE 82 - 10

Expérience PIERO
Description
PH. GULLY - Note TT SETRE 82 - 15

146.2.07		1
TITRE SIMULATEUR ACCIDENT EN DOUBLE PHASE		*Pays* FRANCE *Organisme directeur* IPSN/EdF/THOMSON
TITRE en anglais TWO PHASE PLANT SIMULATOR		*Organisme éxécuteur* IPSN/EdF/THOMSON *Responsables* THOMSON
Date de démarrage 1982	*Etat actuel*	*Scientifiques*
Date d'achèvement	*Dernière mise à jour*	

1 - OBJECTIF GENERAL

Mise à disposition d'un outil permettant de simuler les transitoires et accidents conduisant à avoir de la double phase dans le circuit primaire.

2 - OBJECTIFS PARTICULIERS

Cet outil doit être développé à partir du code CATHARE, en lui apportant les modifications permettant en particulier de réaliser les transitoires en temps réel ou mieux.

Utilisation d'un calculateur propre et réalisation d'un panneau de commande.

3 - INSTALLATIONS EXPERIMENTALES

Néant.

4 - ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

Une étude de faisabilité à partir du code CATHARE a été réalisée et a conclu positivement moyennant quelques aménagements sur les pas en temps. La décision a été prise de continuer l'étude.

5 - PROCHAINES ETAPES

Réalisation du simulateur à partir de la version CATHARE 1.0. Commande d'un calculateur SEL.

6 - RELATION AVEC D'AUTRES ETUDES

Développement du code CATHARE.

7 - DOCUMENTS DE REFERENCE

Rapport de la seconde phase de l'étude de faisabilité du simulateur SIPA.

<table>
<tr><td colspan="2">174.1.01</td><td>1</td></tr>
<tr><td colspan="2" rowspan="2">TITRE

BOUCLE SYSTEME BETHSY</td><td>Pays FRANCE</td></tr>
<tr><td>Organisme directeur
IPSN/EdF/FRA</td></tr>
<tr><td colspan="2" rowspan="2">TITRE en anglais

BETHSY SYSTEM LOOP</td><td>Organisme éxécuteur
CEA/STT</td></tr>
<tr><td>Responsables
M. COURTAUD
CEA/STT</td></tr>
<tr><td>Date de démarrage 1982</td><td>Etat actuel en cours</td><td rowspan="2">Scientifiques</td></tr>
<tr><td>Date d'achèvement 1986</td><td>Dernière mise à jour 1. 1983</td></tr>
</table>

1 - OBJECTIF GENERAL

Dans le contexte général du développement des connaissances thermo-hydrauliques, la boucle système doit permettre une meilleure résolution des problèmes liés à la conduite du réacteur en situation post accidentelle et, en cas de besoin, doit apporter une aide dans la validation de conceptions nouvelles des systèmes de sûreté.

2 - OBJECTIFS PARTICULIERS ET PROGRAMME

La mise en oeuvre de la boucle système permettra d'apporter une contribution essentielle dans les domaines suivants :

- identification et compréhension de l'enchaînement des phénomènes,
- vérification des codes de calcul,
- validation globale des bases physiques des procédures,
- analyse des modifications de conception,
- maintien des compétences en thermohydraulique.

La réalisation doit commencer en 1983 et la mise en service en 1986.

3 - INSTALLATIONS EXPERIMENTALES

Boucle système installée au centre de GRENOBLE.

4 - ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

Décision de construction, avant-projet réalisé.

5 - PROCHAINES ETAPES

Construction de la boucle.

6 - RELATIONS AVEC D'AUTRES ETUDES

Etudes d'analyse de fonctionnement et de procédures.

Qualification de codes de calcul et simulateurs (en particulier CATHARE et SIPA).

7 - DOCUMENTS DE REFERENCE

	CLASSIFICATION : 1.
TITLE (ORIGINAL LANGUAGE) : SISTEMA DI SALVATAGGIO DEL NOCCIOLO (SSN)	COUNTRY : ITALY SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : CORE RESCUE SYSTEM (SSN)	ORGANISATION : ENEA - NUS CORP. PROJECT LEADER : Gianni PETRANGELI
INITIATED : January 1982 COMPLETED : Foreseen achievement: December '83 STATUS : In progress LAST UPDATING : /	SCIENTISTS : ENEA A. ANNUNZIATO, D. MAZZEI A. VALERI M. VIGNOLINI

1. General objective

Feasibility study of a new design Core Rescue System (SSN) aimed at reducing core melt probability of a nuclear power plant equipped with W type 312 PWR, possibly also in case of unforeseen sequences.

2. Particular objective

Verification of SSN effectiveness in reducing core melt probability by a factor (at least) ten.
Verification of thermohydraulic adequacy of SSN in assuring core cooling, for ten hours time, in presence of degraded accident sequences chosen as plant risk significant.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
	1.

3. Experimental facilities and research programme

The risk reduction potential of a proposed core rescue system (SSN) for a Westinghouse "312" 2785 MWt reference plant is to be estimated.

The probabilities of dominant core melt accident sequences for the reference plant without the SSN is to be estimated.
The reliability of the SSN under the various accident conditions important to core melt is to be evaluated.

The probability of dominant core melt accident sequences for the reference plant equipped with the SSN is to be evaluated.

System dynamic response analysis, using the RELAP 5 code, of the most dominant degraded core accident sequences is to be performed for the reference plant equipped with the SSN.
The level of performance to be demonstrated is an extension by, indicatively, ten hours, through SSN operation, of the time available before severe damage occurs.

A refined conceptual design of the system will be produced as a result of this work.

4. Project status

In 1982 a conceptualdesign of the SSN has been performed. The proposed system is mainly a new arrangement of subsystems based on well proven operating principles.

It performs an automatic primary circuit depressurization function in presence of very dangerous conditions of the core (core exit high temperature, vessel low level, failed scram), a low pressure borated water injection from ad hoc accumulators, a steam condensation and atmospheric dispersion of core decay heat.

Some preliminary thermohydraulics and risk reduction verifications have already been performed.
Complete independent thermohydraulic and probabilistic verifications are presently under way according to the program at point 3.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION: 1.

- ENEA-TERM 016 - Gennaio 1983
  "Calcolo del transitorio termoidraulico a seguito dell'apertura spuria della valvola SSN per lo studio dello shock termico nel vessel"

  D. Mazzei, M. Vignolini;

- ENEA-TERM 005 - Maggio 1983
  "Analisi termoidraulica del Sistema di Salvataggio del Nocciolo in caso di rottura di 6" in un reattore PWR"

  D. Mazzei - M. Vignolini.

- "Probabilistic Risk Analysis (PRA) on the Effectiveness of a Core Rescue System (SSN) for PWRs"

  G. Petrangeli, A. Valeri - ENEA-DISP/SER

- "A core Rescue System (SSN) for the implementation of preventive defense in depth safety criteria"
  Part 1 : System Description and Risk Reduction Analysis.
  CSNI - Specialist Meeting on Decay Heat Removal Systems-
  Würenlingen, Switzerland - 25-29/4/83;
  G. Petrangeli, A. Valeri - ENEA/DISP-SER

- "A Core Rescue System (SSN) for the implementation of preventive defense in depth safety criteria"
  Part 2: Thermohydraulic verifications.
  CSNI - Specialist Meeting on Decay Heat Removal Systems -
  Wurenlingen, Switzerland - 25-29/4/83

  A. Annunziato, D. Mazzei, M. Vignolini - ENEA/TERM

9. Degree of Availabiity

References can be requested to Mr. Gianni Petrangeli,
Director - Research and Development Sector -
ENEA, V.le Regina Margherita n. 125, 00198 Roma, Italy

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION: 1.

5. Results

Thermohydraulics and risk reduction computations already performed for a case in which the proposed SSN is assumed to be installed as additional system in a PWR plant, provide a reduction factor of the core melt probability higher than 10.

6. Next steps

A refined conceptual design of the system, basing on thermohydraulic and probabilistic computations, should be made.

The cost effectiveness of possible system engineering alternatives should be evaluated.

7. Relation to other projects

None.

8. References

- ENEA RT/DISP (82)1
  "Study on Safety Criteria and Systems for next PWRs"
  G. Petrangeli;
- ENEA-TERM 003 - Luglio 1982
  "Analisi termoidraulica del sistema di Salvataggio del Nocciolo (SSN) in caso di perdita di acqua di alimento in G.V. e mancanza di energia elettrica in un reattore PWR";
- ENEA-TERM 011 - Novembre 1982
  "Analisi termoidraulica del Sistema di Salvataggio del Nocciolo (SSN) in caso di piccola rottura e mancanza di energia elettrica in un reattore PWR"

  A. Annunziato, I. Mazzei, M. Vignolini;
- ENEA-TERM 015 Gennaio 1983
  "Studio di ottimizzazione delle caratteristiche dell'SSN e analisi termoidraulica in caso di perdita di acqua di alimento ai G.V. e mancato arresto rapido di un reattore PWR.

  D. Mazzei, M. Vignolini;

	CLASSIFICATION : 1
TITLE (ORIGINAL LANGUAGE) : Projekt FIX-II Värmeöverföring under nedblåsningsfasen vid LOCA	COUNTRY : Sweden SPONSOR : Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE) : Project FIX.II, Blow-down Heat Transfer during LOCA	ORGANISATION : STUDSVIK ENERGITEKNIK AB PROJECT LEADER : Rolf Persson
INITIATED : 1978-01-01 COMPLETED : 1983-11-30 (Phase 4) 1984-12-31 (Phase 5)	SCIENTISTS : Lars Nilsson P-Å Gustafsson
STATUS : In progress LAST UPDATING : 1983-06-06	

1 **General Aim:** Support and verification of analytical methods for the blow-down phase of LOCA's with application to Swedish BWR's.

2 **Particular objectives:** Measurement of time to dry out and post dryout HTC for pump trip experiments.

Determination of the dryout limit and heat transfer coefficients (HTC-s) during dryout and post dryout for large and small breaks. Three different configurations of breaks in one of the two main recirculation lines will be studied.

3 **Experimental Facilities and Programme:** The 6 MW FIX-loop with an electrically heated 36-rod bundle will be used.

Phase 1. Scaling calculations and experiment planning
" 2. Reconstruction of test facility, instrumentation and pretest calculations
" 3. Calibration and shake-down tests
" 4. First experimental period
" 5. Second experimental period

4 **Project Status:** Phase 1 through 3 have been completed. Experimental part of phase 4 has been carried through comprising large and intermediate size break tests. Experiments with dryout during flow transients after pump trip are in progress. Evaluation and documentation of phase 4. Planning of new experiments for phase 5. Overhauling of the FIX-loop. /contd

Blow-down Heat Transfer during LOCA-Project FIX-II

6	Relation to other Projects and codes:	FIX-I - Transient Dryout Experiment, (1974-1977). Code: RELAP5, Analysis of FIX experimental data.
7	Reference Documents:	Project description, AB Atomenergi, TPM-RL-1766 (in Swedish). FIX-II - Summary report of Phase 1, STUDSVIK/E4-79/22 FIX-II - Scaling Calculations for the Design of the FIX-loop, STUDSVIK/E4-79/107. FIX-II - Summary report of Phase 2, STUDSVIKS Arbetsrapport NR-83/238. FIX-II - Summary report of Phase 3, STUDSVIK Arbetsrapport NR-83/267 (In Swedish) FIX-II - Topical report of Phase 4; Results of static dryout measurements, STUDSVIK Arbetsrapport NR-83/274
8	Degree of Avail-ability:	In general no restriction upon published results except for some topical reports containing ASEA-ATOM proprietary data. Contact person: Project leader or C Gräslund, Nuclear Power Inspectorate, Box 27106, S-102 52 STOCKHOLM, SWEDEN

1.1.1. DYNAMIC EFFECTS OF DEPRESSURISATION

Berichtszeitraum/Period 1.1.82 - 31.12.82	Klassifikation/Classification 1.1.1	Kennzeichen/Project Number 150 312
Vorhaben/Project Title Begleitende theoretische Arbeiten zu den SWR-Versuchen des Forschungsvorhabens RS 16B bzw. 150.396 Pre and Post Test Analysis of the BWR-Series Experiments of Research Project RS 16B/ 150.396		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Battelle-Institut e.V. Frankfurt am Main
Arbeitsbeginn/Initiated 1.11.77	Arbeitsende/Completed 31.12.1982	Leiter des Vorhabens/Project Leader Dr. T. Kanzleiter
Stand der Arbeiten/Status completed	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1 General Aim

Analysis of the BWR pressure vessel blowdown experiments of Project 150.396 (previously RS 16B) by

- design calculations,
- pre-test calculations and
- immediate post-test calculations

using an existing computer code that is also used for reactor plant design, in order to

- continuously optimize the experimental procedure,
- investigate the merits and limits of the computer code for simulating BWR blowdown processes.

2 Particular Objectives

- Discharge process (steam line rupture, feedwater pipe rupture)
- Processes inside the pressure vessel (phase separation, mixture level behavior, non-equilibrium effects)
- Load on internals

3 Research Program

3.1 Design calculations with parameter variation for optimizing the experimental program, the experimental facility and the instrumentation.

3.2 Pre-test calculations on a best-estimate basis to fix the measuring ranges for the various measuring cascades and to determine the efficiency of the computer code (and the skill of the user) by later comparison with the experimental results.

3.3 Immediate post-test calculations to clarify discrepancies between pre-test calculations and experimental results, which are also aimed at improving performance or pre-test calculation of the subsequent experiments.

4 <u>Experimental Facilities, Computer Codes</u>

To perform the analyses, the RELAP4/MOD5 code made available by the NEA library (Ispra) at the request of USNRC was used and modified to a limited extent (e.g. by inserting a simple phase separation model).

5 <u>Progress to Date</u>

Ad 3.3 Completion of post-test calculations for all experiments, comparisons model calculations/experimental results, documentation of the results of the calculations in the Quick Look Reports for Project 150.396.

Ad 3.1 to 3.3 Drawing up the final report.

6 <u>Results</u>

Comparison of the results of the first steam line rupture tests with those of the relevant pre-test calculations showed substantial deviations, in particular as regards the load on internals. This gave rise to the following improvements of modeling:

- Very careful simulation of the hot-stand-by-like initial steady-state conditions in the experimental pressure vessel by previously making a separate steady-state model calculation taking into account phase slip (approximated pool boiling) and the associated reduction of the void fraction.
- Variation of the Moody discharge coefficient at the rupture site.

These modifications resulted in a markedly improved, but not yet fully satisfactory agreement of model calculation and experiment. The remaining differences,

- underestimation of the load on internals by the calculation, because boiling delay and flashing processes inside the pressure vessel are not being taken into account and
- deviations of calculation from experimental results as regards composition and mass flow of the escaping fluid, because the mixture movements (ejected fountains, flow around the steam drier) and the long-term phase separation inside the pressure vessel cannot be sufficiently modeled, in principle cannot be eliminated by an equilibrium calculation model of RELAP4 type and indicate the necessity

to use non-equilibrium models (taking into account thermal non-equilibrium and phase slip) for simulating BWR depressurization processes.

7 Next Steps

The project is completed.

8 Relation to other Projects

150.396, RS 475 and RS 478.

9 References

Burow, P.: RELAP4/MOD5-Modellrechnungen zu den SWR-Druckentlastungsversuchen, final report BIeV-R-63.547-1, Battelle-Institut, December 1982.

10 Degree of Availability of the Reports

Reports are available by special agreement through GRS-FB.

Berichtszeitraum/Period 01.01.82 - 31.12.82	Klassifikation/Classification 1.1.1	Kennzeichen/Project Number 150 376
Vorhaben/Project Title Vent-Valve Assembly, Functional Reliability And Pressure Relief Performance During The First Blowdown-Phase Überströmvorrichtung - Funktionssicherheit und Druckausgleichsverhalten in der ersten Blowdown-Phase		Land/Country Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Brown Boveri Reaktor GmbH
Arbeitsbeginn/Initiated 01.04.79	Arbeitsende/Completed 30.04.83	Leiter des Vorhabens/Project Leader Dr. B. Hofmann
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 82	Bewilligte Mittel/Funds

1. General Aim

The vent valve (vv) assemblies incorporated in BBR pressurized water reactors are major safety elements which guarantee cooling of the reactor core in the event of a large primary system break. They ensure pressure compensation between upper plenum and downcomer when depressurization waves reach these regions following a rupture in a "cold leg" of the reactor coolant loop. They also ensure pressure relief of the upper plenum when coolant evaporates after a primary system pressure drop caused by a leak. The mechanical load on the vent valve assemblies as well as the differential pressure load on the core barrel are governed primarily by the opening characteristics of the valves. The investigation of the opening characteristics of the vv provide essential data for establishing the actual safety margins in the design of the vv assemblies. The measurements are to be compared with analytical calculations.

2. Particular Objectives

The dynamic opening characteristics of an original vv are measured under realistic conditions in a special test setup. The hydraulic and mechanical loads under various depressurization conditions are compared with calculations using the PISCES-3DE, PISCES-2DELK and CRAFT codes.

3. Test Program

3.1 Preliminary theoretical study to define design and test parameters

3.2 Test-setup layout

3.3 Detailing of test matrix
3.4 Construction, erection and instrumentation of the test-facility
3.5 Experiments
3.6 Evaluation of measurements and comparison with design calculations and results of the preliminary study.

4. Experimental Facilities, Computer Codes

The design and the instrumentation of the test setup are described in detail in the technical reports mentioned under Re 9.
The following computer codes are used for pretest and design calculations of the vent valve assemblies and the experimental facilities.

ANSYS	Determination of mechanical stresses within the vv assemblies during an impact on the RPV-wall
PISCES-3DE	Calculation of the pressure relief conditions within the test facility, and the opening characteristic of the vent valve.

5. Progress to Date

During the period under review the test program was completed by running the main experiments. The experimental data have been plotted and interpreted for the final report. The following progress was archieved:

Re 3.4

- After attaching the rupture disc system to the blowdown pipe the completed experimental facility was dynamically tested by actuating a blowdown out of 50 bar system pressure (T = 120 C).

- The instrumented vent valve assembly was mounted to the flange in the separation wall between the inner and the outer test volume. The position detection devices were adjusted and calibrated.

- Pressure induced acceleration and impact were measured according to the test matrix starting with cold water tests. The mechanical loads increased with higher water temperatures and rising N_2-pressure in the pressurizers.

- The effects of undercooling were studied in an additional saturated water test at 235 C (p_{sat} = 30 bar).

- The highest loads resulted from depressurization of 50 bar initial system pressure. The differential pressures at the vv were comparable to those calculated for reactor conditions assuming a 2F breek of a "cold leg".

6. Results

- The experimental installations have been tested under heavy pressure relief conditions

- The effective inertia of the tested vv was three times as high as calculated for the steel component, e. g. the measured impact velocity was much lower than expected.

- The differential pressure acting on the vv is mainly governed by metastable thermodynamic effects in the blowdown pipe.

- No permanent deformation of the vv was observed.

7. Next Steps

- Analyses of the experimental data
- Discussion of the pressure undershoot observed in hot water blowdowns
- Completion of the final report

8. Relation to Other Projects

9. References

BBR F+E-Berichte Nr. 902-J32 F1(80) and Nr. 902-J32F2(81), Überströmvorrichtungen Funktionssicherheit in der 1. Blowdown-Phase; Technical Reports

10. Availability of Reports

Via GRS.

TPR 19A34

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	1.1.1	150 396

Vorhaben/Project Title	
Versuche zur Druckentlastung in einem Modell-Reaktordruckbehälter mit Einbauten für SWR-Bedingungen Experiments on the Pressure Relief in a Model Reactor Pressure Vessel with Internals at BWR Conditions	Land/Country Fördernde Institution/Sponsor: BMFT Auftragnehmer/Zuwendgsempf./Contractor: BATTELLE - INSTITUT E.V. Frankfurt am Main

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.08.1979	31.12.1982	Dr. T. Kanzleiter
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1 General Aim

The experimental blowdown program is aimed at integral large-scale experimental simulations of loss-of-coolant accidents (steam line break and feed water line break in a BWR, steam line break in a PWR steam generator). All experimental results are to be compared with the results of model calculations to show the applicability of the computer codes and, if possible, to improve them.

2 Particular Objectives

- Discharge process, mass flow and enthalpy flow at the rupture site
- Thermohydraulic phenomena inside the pressure vessel (pressure drop, thermodynamic non-equilibrium, flashing, behaviour of the water level, single- and two-phase flow phenomena)
- Loads acting on reactor pressure vessel internals.

3 Research Program

3.1 BWR steam line break experiments (Nos. DL1-DL6)

3.2 PWR steam generator steam line break experiments (Nos. FL1-FL4)

3.3 BWR feed water line break experiment (No. SL1)

4 Experimental Facilities, Computer Codes

Model reactor pressure vessel (11.3 m high, volume 5.3 m^3, electrical heater 600 kW) with BWR-type internals and 110 measuring systems.

5 Progress to Date

Ad 3.3 Performance of feed water line break experiment SL1

Ad 3.2 and 3.3 Completion of reports

6 Results

The eleven blowdown experiments performed (see 3) yielded a lot of new, reliable, accurate and well documented measuring data

- for the integral behaviour of the BWR model reactor pressure vessel during blowdown and also
- for some particular effects as boiling delay, flashing, phase separation under gravity, pressure wave behaviour in saturated water and at free water surfaces

The following conclusions can be drawn from the experimental results and the model calculations performed up to now:

- The blowdown process in the BWR model reactor pressure vessel, in particular in the case of steam line rupture, is much more complex and much more difficult to model than had been expected.
- The initial state prior to blowdown has a very large influence on the depressurisation process and therefore has to be determined and modelled very carefully.
- Although the overall process of depressurisation (e.g. the pressure history) can be roughly described by simple codes assuming equilibrium, substantial differences are observed in particular with respect to discharge process, mixture distribution in the pressure vessel and loads on the internals, which can only be partly eliminated with additional effort in modelling.
- In case of hot-stand-by and similar initial conditions (like those in the present experiments), equilibrium codes result in lower loads on internals than the measurements do. Satisfactory results can only be achieved by non-equilibrium codes.
- Blowdown experiments from full-load operating conditions cannot be performed in a large experimental facility as is used in the present experiments because the necessary heating capacities cannot be reached (here: 600 kW; correctly scaled down: 26.3 MW). This means that direct verification of computer codes cannot be realised for these operating conditions.

7 Next Steps

The project is completed

8 Relation to Other Projects

150 312, RS 475, RS 478

9 References

Battelle Frankfurt Reports (in German):

- Kanzleiter, T.: BWR Blowdown Experiments
  Final Report BF-R64.167-01 (December 1982)
- Zirnig, W.: Measuring Error Analysis for the BWR Blowdown
  Experiments. BF-R64.167-21-1 (December 1982)
- Quick Look Report DL6. BF-R64.167-30-6 (March 1982)
- Quick Look Report FL1. BF-R64.167-30-7 (Nov. 1982)
- Quick Look Report FL2. BF-R64.167-30-8 (Nov. 1982)
- Quick Look Report FL3. BF-R64.167-30-9-I and -II (Nov. + Dec. 82)
- Quick Look Report FL4. BF-R64.167-30-10 (Nov. 1982)
- Quick Look Report SL1. BF-R64.167-30-11-I and II (Oct. + Dec. 82)

10 Degree of Availability of the Reports

Reports are available by special agreement through GRS-FB.

Berichtszeitraum/Period 1.1.82 - 31.12.82	Klassifikation/Classification 1.1.1	Kennzeichen/Project Number 06.01.01/09A (PNS 4115)
Vorhaben/Project Title 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 dynamical loadings and deformations of reactor-pressure-vessel internals		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated 1.1.1978	Arbeitsende/Completed 31.12.1982	Leiter des Vorhabens/Project Leader Dr. A. Ludwig
Stand der Arbeiten/Status completed	Berichtsdatum/Last Updating Dez. 1982	Bewilligte Mittel/Funds

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 code HDRNA for analysis of the temperature distribution.

5. Progress to Date

A series of post-test calculations has been performed with FLUX/DRIX/ CYLDY3 for the HDR experiments V31/V31.1 in order to find the optimal parameters for the pre-computations of the more severe experiment V32 (which had been announced as the "German Standard Problem No. 5"). These pre-computations have been performed using two different methods of connecting the DRIX-model for the nozzle flow and the FLUX-model.

Furthermore, pre-computations have been done for the test V33, which started from the same thermodynamical initial conditions as V32, but the opening area of the blowdown pipe had been reduced to one quarter by an orifice plate.

After the tests V31.2 (a replication of V31/V31.1 with increased measurement density), V32 and V33 had been run in early 1982; their results have been evaluated and compared to the computational findings. From the comparisons, conclusions have been deduced regarding the validity and applicability of the computational models comprehended by the codes.

In the test V34, isothermal initial conditions have been established; the core barrel was not clamped at its upper flange, and the gaps between the snubbers at its lower edge and the pressure vessel wall had been reduced such that impacts could be expected to occur during the blowdown. In order to prepare this experiment, the low temperature measures of the gaps and the pre-test position of the core barrel have been determined in co-operation with PHDR. Besides this, parametrical studies concerning different initial gap sizes as well as a pre-calculation using the expected initial conditions have been performed with FLUX/DRIX/CYLDY3. After the test was run, post-calculations were performed with initial conditions fitted to the experimental findings, and the impact model enhanced to allow for plastic deformation.

6. Results

For the tests where the upper core barrel flange had been rigidly clamped to the pressure vessel (V31.x, V32 and V33), a generally good and often very good agreement between calculated and measured data was found. Parametric studies in the post-computations for V31 and detailed analysis of the pre-computations for V32 indicate:

- A refinement of the discretization in FLUX yields only marginal improvement; the "medium fine grid" used for V31 is sufficient.
- The coarse modelling of evaporation in the upper plenum after the pressure has dropped below the saturation pressure (in FLUX merely possible by adapting the speed of sound in this region) cannot describe adequately the thermodynamical processes. The effect on pressure differences and other strain entities at the core barrel, however, is small. Thus the evaporation in the upper plenum may be neglected in the model at all.
- DRIX shows pressure waves inside the blowdown nozzle at the beginning of the blowdown, which are less damped than the experimental waves. By these waves oscillations of the core barrel are induced, the amplitudes, however, being mostly far smaller than those of the main motion. Therefore the simulation of the nozzle flow by DRIX is suffi cient, especially as the quasi-stationary mean pressure is matched very well.
- A simple procedure for coupling the DRIX model to the FLUX model is satisfying.
- A detailed structural model of the pressure vessel is (even for HDR) of minor im-importance for determing loadings and strains of the core barrel, provided that the effect of the vessel flexibility upon the pressure wave propagation inside the downcomer is taken into account properly.

The computation for test V33 showed the reliability of the computational models which are used. Besides this, it confirms the already formerly stated finding that a reduction of the break opening cross section does not yield a linearly proportio-

nal reduction of the loadings of the pressure vessel internals.
In these tests various thermodynamical initial conditions as well as different fluid dynamical boundary conditions have been investigated. From the good agreement between measured and computed results in all these cases it is concluded that the combination of the codes FLUX, DRIX, and CYLDY3 as well as the computational models on which they are based, may be considered to be verified successfully, at least for configurations where the core barrel is clamped to the pressure vessel.
The evaluation of test V34 shows that the prescribed initial conditions (particularly the gap sizes) had not been matched by the experiment. Therefore the pre-calculation of kinematical entities (e.g. displacements) could not yield any agreement with the test results, whereas the pressure differences have been predicted perfectly. Furthermore, some effects of markable plastic deformations (presumably at the snubbers) could be settled for the periods of impacts between the core barrel and the pressure vessel wall. This had not been modelled in the pre-calculation. Some post-test calculations using the experimental initial conditions as well as a simple model of plastification yielded all together a rather good agreement. Besides this, the very good prediction of the fluid dynamics in V34 shows that fluid-structure interaction is hardly affected by the support conditions of the core barrel, but mainly by the "local" flexibilities of the structural components.

7. Next Steps

As the tests concerning pressure vessel internals have been finished and evaluated in phase I of the HDR safety program, work on this project will be closed by the end of 1982.

8. Relation with Other Projects

The project is closely related to the other RS 06.01.01 and RS 06.01.02 projects. Besides this, it is coordinated with all other projects of the HDR blowdown program.

9. References

U. Schumann: Impacts and Fluid-Structure Interactions in Pressurized Water Reactor Safety Analysis. Nucl. Engrg. Des. 69 (1982) 313-326.

A. Ludwig, U. Schumann: Fluid-Structure Analysis for the HDR Blowdown and Snapback Experiments with FLUX. Nucl. Engrg. Des. 70 (1982) 321-333.

U. Schumann: Experimental and Computed Results for Fluid-Structure Interactions with Impacts in the HDR Blowdown Experiment. To be published in Nucl. Engrg. Des. (1983).

10. Degree of Availability of the Reports

The above literature is published. Additional unpublished reports are available upon request.

Berichtszeitraum/Period 1.1.1982 - 31.12.1982	Klassifikation/Classification 1.1.1	Kennzeichen/Project Number 06.01.01/10A (PNS 4116)
Vorhaben/Project Title Experimental Data Acquisition and Processing of the Dynamic Behavior of the Pressure Vessel Test Internals in the HDR Blowdown Experiments Meßtechnische Erfassung und Auswertung des dynamischen Verhaltens der Versuchseinbauten im Reaktordruckbehälter (RDB) des HDR im Rahmen der HDR-Blowdown-Versuche		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated 1.1. 1975	Arbeitsende/Completed as item HDR	Leiter des Vorhabens/Project Leader J. Kadlec
Stand der Arbeiten/Status	Berichtsdatum/Last Updating Dec. 1982	Bewilligte Mittel/Funds

1. General Aim

Experimental verification of computer codes for predicting the dynamic response of reactor vessel internals.

2. Particular Objectives

Instrumentation with measuring sensors of the core barrel and reactor pressure vessel of the HDR experimental facility. Measurement of the danymic response of instrumented components within the framework of the HDR blowdown experiments.

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 breaks down into:

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 transducer data by laboratory basic test and dynamic calibration of the original measurement chains prior to their installation in the HDR reactor pressure vessel,

3.7 installation of the instrumentation in the HDR reactor pressure vessel including performance test and acceptance,

3.8 conduct of tests and acquisition of measuring data,

3.9 evaluation of measured data.

4. Experimental Facilities, Computer Codes

Facility for qualification of instrumentation. Statical and dynamical calibration facilities. Computer code EVA /̅ 1_7.

5. Work completed

Pre-test and post-test calibrations of accelerometers and displacement transducers. Extraction of eigenfrequencies, mode shapes and critical damping ratios of HDR-core barrel from the step relaxation response signals measured in the snapback test V59.05. Completion of report /̅ 1_7 summarizing the results of the snapback test series V59 and describing the computer code EVA.

6. Results obtained

The pre-test calibration of accelerometers and displacement transducers revealed minor deviations ($\leq \pm$ 3%) of the sensitivity from the supplier's data; nearly the same values (deviations < 1%) were obtained in the post-test calibrations performed on several selected samples. The modal-analytic evaluation of the snapback test V59.05 revealed good performance of the test procedure and demonstrated the feasibility of the experimental modal-analysis in the hostile environment (hot water 240^{o}C, 111 bar). Extracted eigenfrequencies revealed a small increase (approx. 3-5%) compared to the values at room temperature; this increase corresponds to theoretical expectations. Extracted critical damping ratios showed trends similar to the corresponding room temperature values. The intercomparison of the mode shapes extracted in both test series did not reveal remarkable deviations.

7. Plans for the near future

Completion of the modal analytic evaluation of the blowdown test series V31.2 - V33 and of the snapback test series V59.04 - V59.06. Elaboration of the final documentation.

8. Relation with other Projects

The data on the structural response of the reactor pressure vessel test internals under blowdown conditions, which are measured and evaluated in the experiments, serve as the input for code development under the PNS 4115 and 4126 tasks for prediction of the dynamic load of reactor internals, taking into consideration fluid-structure interaction.

9. Literature

/̅ 1_7 G. Eberle, J. Kadlec, Extraction of eigenfrequencies, mode shapes and critical damping ratios of HDR core barrel mockup from step relaxation response signals measured in the HDR snapback test series V59. KfK 3408,KfK Oct. 1982.

10. Availability of Reports

The reference cited above has been published.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.82-31.12.82	1.1.1	06.01.02/11A PNS4125
Vorhaben/Project Title Weiterentwicklung und Verifizierung von fluid-/ strukturdynamischen Codes zur Analyse der Beanspruchung von RDB-Einbauten beim Blowdown Development and Verification of Coupled Fluid-Structure Dynamics Codes for Analysis of Dynamic Stresses and Deformations of Reactor Vessel Internals during LOCA		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated 1.1.78	Arbeitsende/Completed 31.12.83	Leiter des Vorhabens/Project Leader Dr. Schlechtendahl
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 31.12.82	Bewilligte Mittel/Funds

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 after a postulated failure of the primary coolant circuit close to the reactor nozzle.

3. Research Program

Development of analytical methods and computer codes for fluid-structural dynamics and verification using the HDR-test-results.

4. Computer-Codes

During this project several computer codes have been developed to analyze multi-dimensional transient fluid flows and the dynamics of cylindrical shells with and without fluid-structure interaction. As a basic code for simulations of the whole process a combination of the following codes is being used:

DRIX2D for 2D transient non-equilibrium two-phase flow.

CYLDY3 for the shell dynamics of the core barrel and the pressure vessel.

FLUX for 3D fluid flow in the pressure vessel and fluid-structure coupling.

5. Work Performed

- Development of a new version (FLUX-5) of the FLUX code, which incorporates a homogenized model of the reactor core. Interactions of the core with the core support structure and the water are taken into account.
- Improvements of the structural model in FLUX, in particular with provision for flexible and flanges of the core barrel.
- Enhancement of the contact model which simulates the loose support of the core barrel at the upper flange and the potential impacts with the reactor vessel at the lower end. Plasticity at the points of contact have been included in the model.
- Conversion of the LANL-code KFIX from Control Data to IBM computers. The IBM

version has been made available to the HDR-project for further release.

- Simulation of a JAERI experiment (jet impingement and pipe whip test) with DRIX-2D. This work was performed in cooperation with a JAERI delegate to KfK/PNS.

6. Results Achieved

- A parametric study of PWR blowdown events in a PWR with the new FLUX-5 code indicates that the most significant interaction between the core barrel and the fuel elements is via the upper grid plate and the fuel pin spacers. Fluid-structure interaction has only a minor effect within the core. However, the variations of results for several model types are only in the order of the overall expected accuracy. The most simple model (that is a rigid mass ring at the lower end of the core barrel representing 50% of the fuel mass, which is in accordance with both the previous FLUX modelling and its simulation in the HDR experiment) gives results which deviate only by 25% from the more sophisticated model in a conservative way.
- The enhanced FLUX-4 model (flexible flanges and plastically behaving contact points) was used for post-calculations of the HDR experiment V34. The post-calculation has become necessary as the actual gap widths were obviously different from those used with predictive calculations. The overall agreement between calculation and experiment was satisfactory, even though some discrepancies at the very times of contact could not be resolved completely.
- The KFIX-code is now available in both Control Data and IBM versions.
- The calculations for the JAERI experiment provided another successful test for the DRIX code. The results indicate, however, that the experimental technique for rupturing the front disk of the pipe does not provide for instantaneous opening, but rather opens the full cross section in two steps with 25 msec delay.

7. Plans for Future Work

Work will continue mainly for the final development and application of the core model in FLUX-5. The nonlinear behaviour of the core barrel flange will be investigated in more detail. Additional calculations will be performed with KFIX for the experiments belonging to 06.01.03/16A with beginning utilization of the CYBER 205 vector computer at the University of Karlsruhe.

8. Relations to other projects

The project is closely related to the other RS 6.01.01 and 6.01.02 projects and the subproject EV 3000 of the HDR reactor safety program.

9. Literature

- PNS Jahresbericht 1981, KfK 3250
- U. Schumann; NED 69 (1982) p. 313-326.
- T. Belytschko, E.G. Schlechtendahl; NED 70 (1982) No. 3

10. Availability of the Literature

The above literature is published. Technical reports ("Primärberichte") are available upon request from PNS or KFK/ IRE.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1982-31.12.1982	1.1.1	06.01.02/12A (PNS 4126)

Vorhaben/Project Title	
Laboratory Experiments for Validation and Enhancement of Fluid/Structure Dynamics Codes Relevant to Initial Phase of LOCA Laborversuche zur Abstützung von fluid/strukturdynamischen Rechenprogrammen zur Beschreibung der Anfangsphase bei Kühlmittelverluststörfällen	Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe GmbH (KfK) Projekt Nukleare Sicherheit (PNS)

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.1978	31.12.1983	E. Wolf
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

Development and verification of codes treating the dynamic loading of RPV internals.

2. Particular Objectives

Planning, implementation and evaluation of laboratory tests serving as an experimental support of models and computer codes developed under PNS 06.01.02.

3. Research Program

Laboratory experiments with periodical excitation for dynamic coupling between the fluid and the flexible rod structures (fuel rods).

4. Test Facilities, Computer Codes

Thick-walled periodically displaced vessel with various flexible rod internals and liquid filling. Codes from PNS 06.01.02/03.

5. Work Performed

With the single cell and a different geometry of the internal pendulum tests were performed to investigate the influence of the gap on inertia coupling and damping. The fluid and the fluid density (vacuum, air, water) respectively, and system reset were subjected to variations.
A first version of the bundle geometry (16 rods, Ø 10) was fabricated and subjected to a functional test in the existing cylindrical external tube of the single cell.
A second version of the bundle geometry (49 rods, Ø 10), relying on the first version, with a rectangular external tube was designed and fabricated. Construction of a suitable measurement technology has started.

6. Results Obtained

The evaluation of the tests involving the single cell has shown that the good agreement of inertia coupling values between experiment and theory, already found earlier, is likewise maintained in case of gap reduction by the factor 2.5.
However, damping markedly increases.

Testing of the first version of the bundle geometry was successful and furnished valuable indications concerning single-rod and bundle supports which have been incorporated in the design of the second version.

First tests on the construction of a suitable measuring technology have been promising. It is intended to plot a graph of the movements of the single-rod ends.

7. Plans for Future Work

The measuring technology for the bundle geometry will be constructed. The tests with the two bundle geometries (16-rod bundle with circular boundary line and 49-rod bundle with rectangular boundary line) will be performed.

8. Relations with Other Projects

The project is closely linked to the projects PNS 06.01.01 and 06.01.02. It is furthermore coordinated with other subprojects of the HDR blowdown project.

9. Literatur

K.D. Tulke:

Konzeption eines Schwingungsexperiments zur Fluid/Struktur-Dynamik.

Diplomarbeit am Institut für Reaktortechnik der Universität (TH) Karlsruhe, Prof. Dr. D. Smidt, November 1980

10. Accessibility of Work

Kernforschungszentrum Karlsruhe GmbH

Institut für Reaktortechnik der Universität (TH) Karlsruhe, Prof. Dr. D. Smidt

Berichtszeitraum/Period 1.1.82 - 31.12.82	Klassifikation/Classification 1.1.1	Kennzeichen/Project Number 06.01.02/13A (PNS 4128)
Vorhaben/Project Title Mechanisches Verhalten der Kernmanteleinspannung und der Core-Stützen beim Blowdown Mechanical response of the core barrel clamping and the control rod guide tubes during blowdown		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated 1.1.81	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader Dr. R. Krieg
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 9.12.1982	Bewilligte Mittel/Funds

1. General Aim

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

2. Particular Objectives

Nonlinear deformations of the core barrel clamping in case of a postulated failure of the primary coolant circuit close to the inlet nozzle.
Fluid dynamic loading and stresses in the core support columns in case of a postulated failure of the primary coolant circuit close to the outlet nozzle.

3. Research Program

Application and verification of computer codes for fluid- and structural dynamics.

4. Computer Codes, Experimental Facilities

ADINA (Finite elemente program for structural mechanics, nonlinearities may be included)
SING1 (Incompress. potential flow, any three-dimensional geometry)
SING-S (SING1 with fluid-structure interaction)
WELLE 2 (Fluiddynamics, two-dimensional geometry)
Shallow water model for simulation of a slice taken from the upper plenum of the pressure vessel. Horizontal oscillations provided by a shaker.

5. and 6. Progress to Date and Results

For the core barrel clamping three different models have been developed: a simplified model describing the basic types of deformation, a two-dimensional finite element model and a three-dimensional finite element model. The simplified model is restricted to elastic deformations only; the finite element models, however, are able to describe plastic deformations. The results obtained are consistent. Of course, with the three-dimensional finite element model only coarse spatial resolutions are provided, since the number of degrees of freedom must not exceed a reasonable upper limit. The results show that for blowdown loading considerable plastic deformations of the core barrel clamping and a strong influence on the

- 2 -

overall core barrel response must be expected.
The analysis of the core support columns requires a three-dimensional description of the blowdown flow in the upper plenum between the columns. In addition, the close arrangement of the columns requires high spatial resolutions. Therefore the common solution methods are no longer acceptable. To overcome this problem the recently developed computer programs SINGl and SING-S which are based on a boundary integral method have been applied. Here the advantage is that not the three-dimensional fluid region but only the two-dimensional fluid boundaries must be discretized. The calculated core barrel deflections are essentially within the elastic region. Just for one small diameter column rather high plastic deformations are obtained. In order to check the different assumptions, also two-dimensional calculations for a fluid slice taken from the upper plenum have been carried out. Using the shallow water analogy these results could be compared with shallow water experiments. Again it has been confirmed that the fluid compressibility can be neglected for determination of the maximum column deflections. Furthermore it has been found that the two-dimensional calculations for the fluid slice taken from the upper plenum may be used to assess to total loading acting on each column.

7. Next Steps

Based on the findings about the core barrel clamping an improved blowdown analysis will be done for a typical PWR.
For the core support columns the investigations are finished.

8. Relations to other Projects

Coordination with the investigations of the fluid-structure interaction of core barrel and core in case of a blowdown.

9. Literature

R. Krieg, B. Dolensky, F. Eberle, G. Hailfinger, Core Support Columns in the Upper Plenum of a Pressurized Water Reactor under Blowdown Loading; Part I: Two-dimensional Analysis, Assessment of Essential Simplifications, Nucl. Eng. Des. 72 (1982)
R. Krieg, B. Dolensky, A. Granda, Core Support Columns in the Upper Plenum of a Pressurized Water Reactor under Blowdown Loading; Part II: Three-dimensional Analysis, Calculation of Deformations and Stresses, Nucl. Eng. Des. 72 (1982)

10. Availability of the Literature

The above literature is being published.

Berichtszeitraum/Period 1.1.82 - 31.12.82	Klassifikation/Classification 1.1.1	Kennzeichen/Project Number 06.01.03/16A (PNS 4140)
Vorhaben/Project Title Entwicklung von Kondensations- und Schlupfmodellen für Zweiphasen-Rechencodes Development of Condensation- and Slip Models of Two-Phase-Flow Computer Codes		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated 1.10.81	Arbeitsende/Completed 31.12.83	Leiter des Vorhabens/Project Leader M. Wadle
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating Dez. 82	Bewilligte Mittel/Funds

1. General Aim

Two-Phase-Flow Measurement

2. Particular Objectives

Development of condensation- and slip models of two-phase-flow computer codes and verification by nozzle experiments in the KfK-Two-Phase-Flow Instrumentation Test Facility.

3. Research Program

Development of constitutive relations for the computer codes, precalculations and performance of diffusor experiments in the KfK-Two-Phase-Flow Instrumentation Test Facility.

4. Experimental Facility, Computer Codes

KfK-Nozzle-Test-Section (KfK-Nozzle)

Computer codes DRIX, KFIX, DUESE

5. Progress to Date

The previously used test section (KfK-Nozzle) has been completely modified, and the instrumentation was improved in cooperation with the LIT and the IRB. Higher signal quality was achieved, and the 6-beam densitometer was extented to eight beams. Furthermore a traversable pitot-probe was constructed for measurement of the separation zone after the divergent nozzle.
According to precalculations with the DUESE-code, in which a flow pattern map was implemented, an experiment matrix has been established.
In the IRE the LASL-code KFIX is available in its HDR-version. This code was adapted to the 2D nozzle geometry and a friction model was implemented.
During summer 1982 about 110 experiments (2/3 water-vapor experiments, 1/3 water-air experiments) were performed. The experimental data were digitized, correlated and converted by means of calibration curves. Many experiments were calculated with the DUESE-code.

1.8.82 - 31.12.82 06.01.03/16A (PNS 4140)

6. Results

For more than 100 different experiments with varying inflow conditions, experimental pressure- and temperature-curves are available (water-vapor experiments: pressure level from 3 MPa to 10,5 MPa, volumetric vapor quality from 0 to 80 %; water-air experiments: pressure level from 0,3 MPa to 0,95 MPa, mass vapor quality from 0 to 7,8 %).

With a new condensation model and an additional pressure loss due to a sudden expansion, many experiments were calculated. The results show good agreement with the experimental data, depending on a sophisticated application of the friction factor f and the number of bubbles/droplets N. In some experiments an extremely steep pressure gradient develops in the divergent part of the nozzle, which cannot be calculated with the DUESE-code.

7. Next Steps

New experiments with tracer injection for slip measurement are prepared and scheduled for spring 1983. The results will be compared with the calculation and the constitutive relations will be improved.

8. Relations to other Projects

Close relations exist with the project 06.03.02, PNS 4125.

9. References

-

10. Degree of Availability of the Reports

-

140.2.01		1 - 1 - 1
TITRE EFFETS D'UNE DEPRESSURISATION SUR LES STRUCTURES INTERNES D'UN P.W.R.		*Pays* FRANCE *Organisme directeur* CEA/IPSN
TITRE en anglais MACHANICAL EFFECTS OF A LOCA ON P.W.R. INTERNALS		*Organisme éxécuteur* CEA/DEMT/SMTS *Responsables* P.J.GIBERT/DEMT-SMTS
Date de démarrage 1.1.1981	*Etat actuel* en cours	*Scientifiques* D.COSTES DAS/SAER
Date d'achèvement Décembre 86	*Dernière mise à jour* Mài 83	F.AXISA DEMT/SMTS M.GUILBAUD "

1) OBJECTIF GENERAL -

Etudier le comportement mécanique des structures internes des P.W.R. en situation accidentelle (LOCA) en tenant compte des effets de couplage fluide-structure.

2) OBJECTIFS PARTICULIERS -

a) Effectuer une approche linéaire du problème (valable dans les quelques premières dizaines de ms) en tenant compte correctement des effets tri-dimensionnels des ondes. Pour cela, on développe le couplage par fluide dans le programme d'analyse dynamique par sous-structures TRISTANA.

b) Analyse avec TRISTANA des impacts entre les différentes structures résultant de l'accident.

c) Développer et appliquer le programme PLEXUS pour traiter les autres non-linéarités (grands déplacements des structures et des fluides double phase)

3) INSTALLATIONS EXPERIMENTALES -

Il est prévu dans le cadre des accords entre le C.E.A et le projet H.D.R. allemand de faire l'interprétation de certains essais de dépressurisation effectués sur H.D.R.

D'autre part des relations avec E.D.F. et FRAMATOME dans le cadre de ces accords sont établies.

4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS -

- Avancement à ce jour :

a) Le programme TRISTANA a été adapté au problème du couplage fluide-structure. Des tests analytiques de validation de la liaison TEDEL-AQUAMODE par TRISTANA ont été effectués.

b) Une première application du code de calcul à la géométrie H.D.R. a été effectuée.

- Principaux résultats :

Le programme TRISTANA est actuellement opérationnel. Les tests préliminaires ont été satisfaisants.

L'accord entre les principaux modes de résonance calculés et mesurés sur H.D.R. est bon.

5) PROCHAINES ETAPES -

a) Poursuite de l'interprétation des essais H.D.R. Mise sur bande des résultats pour figurer dans le "Final report H.D.R.".

b) Calculs TRISTANA avec impacts.

c) Développement de PLEXUS.

6) RELATIONS AVEC D'AUTRES ETUDES -

Etudes liées au LOCA (145)

7) DOCUMENTS DE REFERENCE -

1) Note Technique EMT/SMTS/VIBR/82/118.

2) 7ème SMIRT : A substructure methode to compute the 3D fluid-structure interaction during blowdown (M. GUILBAUD - F. AXISA - RJ. GIBERT).

8) DEGRE DE DISPONIBILITE

Document 1 non disponible, sauf accord du CEA
Document 2 disponible .

	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 1980	

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 facilities	Small laboratory loops with visualized test sections.
4) Project status	A set of data and parameters have been analyzed.
5) Reports	Centi, Cumo, Farello, Lumini, "Preliminary remarks on jet impingement and jet flow configurations", Cagliari 10-14 Sept. 1979, XVIII International IAHR Congress.
6) 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: ENEA 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. General aim
   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. Particular objectives
   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. Experimental facilities
   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. Project status
   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.

45.897

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: ENEA 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	SCIENTISTS: 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 system 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)

1.1.2. THERMO-HYDRAULIC ASPECTS

	CLASSIFICATION: 1.1.2.
TITLE (ORIGINAL LANGUAGE): DEVELOPMENT OF TWO PHASE FLOW AND SAFETY INSTRUMENTATION	COUNTRY: BELGIUM
	SPONSOR: CEN/SCK
TITLE (ENGLISH LANGUAGE):	ORGANISATION: CEN/SCK
	PROJECT LEADER: M. DECRETON
INITIATED: 1975 / COMPLETED:	SCIENTISTS:
STATUS: Continuing / LAST UPDATING: May 1982.	

1. General Aim

Safety related experiments must always be interpreted through measured data. An evaluation of the performance of the instrumentation used is therefore of primary importance, as well as the development of more reliable measurement techniques.
Such a need exists particularly in two areas : the measurement of two phase flow parameters, and the measurement of high temperatures over a long period of time.

2. Particular Objectives

Attention has been paid on two types of measurements :
- local two phase flow void fraction.
  Measurements with thermocouples pairs, conductive probes and optical reflection sensors are evaluated.
- high temperatures (1000-2000°C)over long periods of time.
  Due to its good long term stability and precision, noise thermometry is taken here as an alternate solution to high temperature thermocouples.

3. Experimental Facility

A two phase flow test facility is available where an electrically heated section can be used to develop different two phase flow patterns. The flow rate, the inlet temperature, the pressure as well as the geometrical dimensions of the heated section can be varied in a large range.

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4. Project Status

Small scale tests are presently under way for the evaluation of different void fraction measurement techniques. A first set of data has been analyzed. High temperature tests have been performed on a laboratory scale. The results have shown the feasibility of the noise thermometry for long term high temperature measurements.

5. Next steps

The experimental evaluation program is continuing with the objective of larger scale tests. In-pile tests of noise thermometry are taken into consideration.

Reference documents

- M. DECRETON, Boiling Crisis in restricted geometries with a near-by cooled wall, 7th International Heat Transfer Conference, Munich 6-10 September 1982.

- M. DECRETON, High Temperature Noise Thermometry for industrial application, 6th Symposium on Temperature, Washington DC, March 15-18, 1982.

	CLASSIFICATION : 1.1.2
TITLE (ORIGINAL LANGUAGE) :	COUNTRY : Denmark
	SPONSOR : Nordic Liason Committee for Atomic Energy
TITLE (ENGLISH LANGUAGE) : NKA/SÅK-3 Reactor Safety Small Break LOCA Analysis	ORGANISATION : Risø National Laboratory
	PROJECT LEADER : Aksel Olsen
INITIATED : 1981 | COMPLETED : Ultimo 1984	SCIENTISTS : Poul Astrup
STATUS : in progress | LAST UPDATING :	

1. General Aim: To study the American TRAC-PF1 codes ability to predict small break loss of coolant accidents.

2. Particular Objective: Together with the other Nordic countries to provide one or more computer codes suitable for small break loss of coolant accident analysis.

3. Experimental Facilities and Programmes: No experiments in this project.

4. Project Status

   1. Progress to Date: Two LOFT experiments analysed and a LOBI experiment analysis in progress.
   2. Essential results: The TRAC-PF1 can predict the LOFT L3-5 and L3-5 and L3-6 test with reasonable accuracy.

5. Next Steps: Calculation of a LOBI case and a preliminary evaluation of the TRAC codes goodness when compared to other codes, studied in Finland and Sweden.

6. Relation to Other Projects and Codes: The project is Internordic. The participants are Risø, Denmark; VTT, Finland; IFE, Norway and Studsvik, Sweden.

7. Reference Documents:

"Annual Report 1981". SÄK-3-D(82)1

"Annual Report 1982". SÄK-3-D(82)5

Poul Astrup: "Test Case Report. LOFT 13-6, TRAC-PF1." SÄK-3-D(83)1

8. Degree of Availability:

Because of content of LOFT and LOBI experimental results, the reports are not all available.

	CLASSIFICATION : 1.1.2.
TITLE (ORIGINAL LANGUAGE) :	COUNTRY : Denmark
	SPONSOR : NKA, Nordic Liaison Committee for Atomic Energy
TITLE (ENGLISH LANGUAGE) : NKA/SÄK-5 REACTOR SAFETY HEAT TRANSFER CORRELATIONS	ORGANISATION : Risø National Laboratory
	PROJECT LEADER : Aksel Olsen
INITIATED : 1981 / COMPLETED : 1984	SCIENTISTS : H. Abel-Larsen
STATUS : in progress / LAST UPDATING :	

1. General Aim

The aim is to establish a set of reliable heat transfer correlations for application in advanced, best estimate computer programmes for LOCA abalysis.

2. Particular Objectives

To assess and recommend heat transfer correlations.

3. Programmes

The starting point has been the correlations in the programmes TRAC-PF1 and RELAP-5.

4. Status

Experimental and calculated results primarily in the post-burnout regime have been compared. It is too early to conclude anything from these examinations.

5. Future Work

Other correlations, both empirically and phenomenologically are included in the examinations.

6. Relation to Other Projects

The project is carried out in collaboration with Technical Research Centre of Finland, Studsvik Energiteknik AB, Sweden and Institute for Energy Technology, Norway.

7. Documents

Work reports have been written. Not available.

8. Contact person

Aksel Olsen, Risø National Laboratory.

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1982 - 31.12.1982	1.1.2	150 0 348 A
Vorhaben/Project Title Entwicklung einer Ultraschall-Stabsonde zur Bestimmung von Füllstand in ein- und zweiphasigen Medien bei blowdown-Bedingungen. Development and operation of rod-shaped ultrasonic probes for liquid-level detection within single- and twophase media under blowdown-conditions		Land/Country Fordernde Institution/Sponsor GRS/BMFT Auftragnehmer/Contractor Technische Universität Berlin Institut für Kerntechnik
Arbeitsbeginn/Initiated 1.1.1981	Arbeitsende/Completed 31.3.1983	Leiter des Vorhabens/Project Leader Prof. Dr.-Ing. U. Wesser
Stand der Arbeiten/Status	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

1. General Aim

Development and operation of rod-shaped ultrasonic probes for collapsed-level detection within transient single- and twophase flows up to $p \leqq 150$ bar and $T \leqq 350^{\circ}$ C.

2. Particular Objectives

2.1 Adaptation of the tube-sealed probe concept (project 150 348) to the envisaged geometrical as well as pressure and temperature conditions within the

- Joint Transient Facility at the KfK
- PKL-Project (KWU-Erlangen)
- LOBI-Project (EURATOM/Ispra)

2.2 Flexibility of the probe system with regard to

- "upwards" respective "downwards" mounting
- optional limited active section

2.3 Provision of matched probe-transducer combinations

2.4 Development of electronics for liquid-level proportional standard output

2.5 Determination of parameters of influence and measures for compensation/calibration

2.6 Investigation of possible external signal-extraction with internal excitation (probe-structure coupling).

3. Research Program

The investigation divides into four (4) interconnected activities:

3.1 Probe design

3.2 Design and optimization of transducer and electronic components

3.3 Measurements at the IKT

3.4 Measurements at external installations.

4. Experimental Facilities, Computing Software

The respective measurement systems will be tested within stationary and nonstationary experimental rigs at the IKT.
Data evaluation and modelling will be effected at computers of the IKT, respct. TUB.
Activities concerning calibration are intended at the "Joint Transient Facility" at the KfK.

5. Progress to Date

Ref. 3.1: The manufacture of the probes has been completed (cf. 2.1).
Ref. 3.2: Transducer- and electronic systems are available.
Ref. 3.3: Functional tests and measurements within pressure-/temperature and flow conditions have been performed according to the possibilities at the IKT (cf. 6).

Ref. 3.4: Measurements concerning susceptibility to electric/acoustic interference have been performed at the PKL-rig. Because of delays in delivery and availability of the external installations no further measurements could be accomplished.

6. Results

Ref. 3.1: The standard probe configuration consists of a 6 mm$^{\phi}$ U-rod, one leg beeing double-tube sealed (sandwich-tube). The max. diameter amounts to 30 mm$^{\phi}$, maximum/minimum probe lengths are 5m/1m .

Optional limitation of the active section (measuring length) as well as "upward"/"downward" mounting are possible. Minor alterations concerning radial dimensions allow simultaneous γ-irradiation (LOBI conditions).

Ref. 3.2: Adaptation of probe-transducer systems yields band-limited spectra (100 - 250 kHz) independent of excitation mode (e.q. FM/ ʃ-pulse). To obtain information concerning temperature/~gradients - temperature compensation - ʃ-pulse excitation will be utilized.

Liquid-level proportional signal output (0 - 10 V/4 - 20 mA) is provided by adjustable RMS-gain setting -according to probe configuration.

Ref. 3.3: Measurements within stationary and transient single-/two-phase air-water surrounding showed good signal stability and reproducability.

Pressure tests up to p = 150 bar confirmed the applicability of the measurement system within pressurized water surrounding.

Transient liquid-level measurements up to p = 50 bar / T = 264 ^{o} C (blowdown-simulation) have been successfully accomplished. With ʃ-pulse excitation, the temperature-signal characteristics is essentially linear. Compensation by temperature/delay time information seems possible. Cooling of the transducer systems is imperative.

Ref. 3.4: Measurements of electric/acoustic interference in the vicinity of the phase-fired power control at the PKL-rig showed no disturbences on the liquid-level signal.

Further system tests and calibration measurements with regard to operational stability and reproducability as well as possible parameters of influence could not be accomplished (cf. 3.4).

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.82 - 31.12.82	1.1.2	150 370
Vorhaben/Project Title Wärmeübertragung im Übergangsbereich zum Filmsieden (Transition Boiling) bei erzwungener Konvektion Transition Boiling Heat Transfer in Forced Convection		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Universität Stuttgart Inst. f. Technische Thermodynamik und Thermische Verfahrenstechnik
Arbeitsbeginn/Initiated 15.3.1979	Arbeitsende/Completed 31.5.1983	Leiter des Vorhabens/Project Leader Stephan/Auracher
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating January 1983	Bewilligte Mittel/Funds

1. General Aim

The design of reactor core-cooling systems is based on the assumption, that at hypothetical emergencies like "loss-of-coolant accidents (LOCA)" or "anticipated transients without scram (ATWS)", nucleate boiling changes immediately to film boiling after burnout conditions are reached. Between bubble and film boiling, however, a "transition boiling region" exists with better heat transfer conditions than in the film boiling region. Measurements are planned to provide reliable data on heat transfer in this transition region. Such data may improve computer codes which simulate LOCA and ATWS emergencies and thus prove the effectiveness of safety installations.

2. Particular Objectives

Experiments on transition boiling heat transfer coefficients as function of pressure, vapour quality, mass flow density and flow pattern. Experiments are carried out under steady-state thermal and hydrodynamic conditions, to explain basic physical aspects of heat transfer in the transition region. Measurements are carried out with temperature-controlled heat transfer surfaces to allow stabilized evaporation in the transient region. Evaporation takes place within tubes of 14 mm inner diameter, to simulate approximately the size of flow channels in the core of BWR- and PWR reactors. Refrigerant R 114 is used as evaporating fluid. The results are then scaled to water.

3. Research Program

3.1 Preparatory theoretical work.

3.2 Building up of experimental equipment.

3.2.1 Condenser cooling circuit.

-2-

3.2.2 Evaporator circuit.

3.2.2.1 Design and construction of a stabilized evaporator.

3.2.2.2 Design and construction of a preevaporator.

3.2.2.3 Design and construction of facilities to study flow pattern and phase velocities (γ-ray attenuation, capacitance probe for void fraction measurements).

3.3 Testing of experimental equipment.

3.4 Experiments. Parameters: mass flow density, vapour quality, pressure.

3.5 Evaluation of data and development of correlations for transition boiling heat transfer.

4. Experimental facilities, Computer Codes

The experimental equipment consists of circuits for the condenser cooling water and for the evaporatingfluid (R114).

5. Progress to Date

Ref.3.3 Works are finished.

Ref.3.4 First experiments at low pressures have been carried out.

6. Results

The first data confirm, that the applied technique enables a steady-state measurement of complete boiling curves. Reliable conclusions of test parameter influence on transition boiling heat transfer are not possible before finishing the experiments.

7. Next Steps

Ref. 3.4 Carrying-out of experiments

Ref.3.5 Evaluation of data and comparison with correlations from literature

8. Relation with other Projects

-

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	1.1.2	150 392
Vorhaben/Project Title Erstellung eines Zweiphasen-Pumpenmodells für die in Druckwasserreaktoren eingesetzten Hauptkühlmittelpumpen Development of a Two-Phase Pump Model for PWR Main Coolant Pumps		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor KRAFTWERK UNION AG Reaktortechnik R 153, Erlangen
Arbeitsbeginn/Initiated 01.08.79	Arbeitsende/Completed 31.12.84	Leiter des Vorhabens/Project Leader H. Watzinger
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31.12.82	Bewilligte Mittel/Funds

1. General Aim

An empirical pump model shall be established which can describe the flow behavior of pressurized water reactor pumps during postulated loss-of-coolant accidents.

2. Particular Objectives

This project is regarded as a follow-up program to research project RS 111, in which the behavior of a pump type used in KWU pressurized water reactors is investigated during postulated loss-of-coolant accidents. An empirical two-phase pump model for digital computer programs shall be developed in cooperation with the manufacturers, who will evaluate the hydraulic aspects of two-phase pumpe experiments. The model shall predict the thermohydraulic behavior of reactor pumps during a loss-of-coolent accident.

3. Research Program

3.1 Critical review of existing experimental results

3.2 Determination of essential differences of similarities

3.3 Clear plotting of the RS 111 experimental results and documentation

3.4 Comparison of the steady-state and transient experimental results

3.5 Comparison of results from pumps of varying sizes and types

3.6 A dimensional analysis which contains all the vital characteristics and constructional data of the pumps.

01.01.82 - 31.12.82 150 392

4. Experimental Facilities

The empirical pump model shall be verified using the LECK-V4 program.

5. Progress to Date

Re. 3.6 KSB and ASTRÖ have been working on their final reports. The work is now finished, and the data is at present being corrected and supplemented.

6. Results

There are no new results.

7. Next Steps

Re. 3.6 KWU is to begin building an empirical pump model in accordance with the scope of supplies and services.

8. Relation with Other Projects

RS 111 (Experimental works)

9. References

-

10. Degree of Availability

-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1982 - 31.12.1982	1.1.2	15o o 521
Vorhaben/Project Title Geschwindigkeitsmeßverfahren für Wasser/Dampf-Strömungen mittels Korrelation überlagerter Temperaturstörungen Velocity-Measurements of twophase flows (water/steam) by correlation of temperature-tracer		Land/Country Fordernde Institution/Sponsor GRS/BMFT Auftragnehmer/Contractor Technische Universität Berlin Institut für Kerntechnik
Arbeitsbeginn/Initiated 1.1o.1981	Arbeitsende/Completed 31.12.1983	Leiter des Vorhabens/Project Leader Prof. Dr.-Ing. U. Wesser
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

1. General Aim

Development of a measurement method for determination of fluid velocity (range from o.o1 to 5o m/s) in single and two phase flows, also under transient conditions and change of flow direction.

2. Particular Objectives

A temperature fluctuation is imposed to the flow by stochastic controlled injection of fluid with different temperature. For the measurement of the fluid velocity this temperature fluctuation can be detected by thermo-couples placed one behind the other in the flow profile. Crosscorrelation analysis produces the velocity information.

This project contains development, construction and test of the measurement insert and of the measurement- and evaluation-procedure. An extensive automatisation of the measurement will be provided by making use of mini-computers.

3. Research Program

The research program can be divided into the following activities:

3.1 Measurement insert

Construction of the fluid injection with valve construction and control, thermocouple-attachment. Improvement of the amplifiers.

3.2 Evaluation

Provision and improvement of evaluation software first for the time series analysers available at the IKT and EURATOM/Ispra, then transfer of the programs to micro-processor or mini-computer-systems.

3.3 Tests

Amongst current tests at the IKT comparative measurements at

- EURATOM/Ispra: LOBI-Project
- KWU/Erlangen: PKL-II-Project
- KfK: "Joint Test Rigs"

are planned.

4. Experimental Facilities

At the IKT there are a transient (blow down) test rig and different stationary loops for most of the necessary systematic measurements. It can be made use of the software provided by the proeceding research projects (RS 135 - 15o 135 B).

5. Progress to Date

The IKT designed temperature noise amplifier 'Tramp2' has been checked for reliable working under conditions of large scale test facilities (LOBI/EURATOM/Ispra, PKL II/KWU Erlangen). Resulting from these experiments the Tramp2-amplifier could be improved in different features. A new measurement-insert (final design to fit into the KfK-'Joint Test Rig') equipped with tracer-injection was completed and installed in our test rig. A real-time correlator that can be controlled by mini-computer is now available and checked for evaluation-quality of temperature noise signals. Another objective of the activities is the investigation of the injection procedure itself. Test facilities to analyse the influence of the tracer injection by Laser-Doppler-measurements have been built up.

6. Results Achieved

The temperature noise measurements with the Tramp2 in the LOBI test rig (EURATOM/Ispra) and in other test rigs gave very satisfactory results. The amplifier (latest model Tramp 2 A) is now equipped with an automatic gain control and serious interference from strong electric/magnetic fields can be removed ('common mode rejection'). A small series of this type of amplifier was produced. The test rig prepared for flow reversal and slow flows was rebuilt and is now available for measurements with the new measurement insert and correlator.

7. Next Steps

The range of application of the measurement method with the new equipment (amplifier Tramp2 A and measurement insert with tracer injection) will be redefined by systematic measurements in our test rigs. As soon as the interface correlator/mini-computer is available the automatic evaluation unit will be built up and programmed. The investigation of the injection procedure (especially measurements of the flow profile with and without tracer injection) will be continued. Further tests in the facilities of the related projects are planned.

8. Relations to other Projects

Cooperation with

- EURATOM/Ispra: LOBI-Project
- KWU/Erlangen: PKL-II-Project
- KfK: "Joint Test Rigs"

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.82 - 31.12.82	1.1.2	RS-109
Vorhaben/Project Title Einfluß der DWR-Umwälzschleifen auf den Blowdown (LOBI-Projekt) Loop Blowdown Investigations (LOBI) Project: Influence of PWR Primary Loops on Blowdown		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Euratom J.R.C.-Ispra
Arbeitsbeginn/Initiated 1.12.73	Arbeitsende/Completed 31.12.87	Leiter des Vorhabens/Project Leader W. L. Riebold
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

1. General Aim

Experimental investigations of the thermohydraulic behaviour of a primary cooling system (PCS) of pressurized water reactors (PWR) during the blowdown phase of a loss-of-coolant accident (LOCA).

2. Particular Objectives

- Design, construction and operation of a large-scale 2-loop blowdown test facility for loss-of-coolant experiments (LOCE) and Special Transients Experiments
- Experimental investigation of the influence of the thermohydraulic behaviour of individual components of a PCS of PWRs on the course of a LOCA by measuring those thermohydraulic quantities relevant for the emergency core cooling (ECC)
- Comparison of experimental results with blowdown computer code prediction calculations; assessment of codes to be used for LWR safety analysis
- Experimental investigation of the LOBI pump and steam generators performance characteristics and of the LOBI discharge nozzles under two-phase flow conditions.

3. Research Programme

- Preliminary Tests to determine the characteristics of the LOBI test facility and to check the measurement instrumentation, data acquisition and process control system
- Experimental Programme A: 60 tests to investigate the influence of rupture size and location, downcomer gap width, pump operation, power input, steam generator secondary side pressure, pressurizer connection, and ECC water injection mode on the blowdown.
  Test Matrix A1: Tests 1 to 30 defined by, and available to the BMFT-Bonn exclusively; test A1-04 used for "blind post-test prediction calculation exercise" (LOBI PREX) with international participation.
  Test Matrix A2: Tests 31 to 60 defined by the BMFT-Bonn taking into account suggestions from Member Countries of the E.C.; results freely available to the latter.
- Experimental Programme B: Tests 61 to 90 defined jointly by and freely available to the Member countries of the E.C.; investigation of the influence of geometrical shape and elevation of some PCS components on the blowdown.

4. Experimental Facilities, Computer Codes

- Simulation of 4-loop PCS of a 1300 MWe PWR by a 2-loop test facility (single-

and treble-loop) for nominal operating conditions: p = 160 bar and T = 325 °C. Heat rejection through a closed loop secondary cooling system (54 bar, 210 - 270 °C).
- Reactor Model with internal annular downcomer and a 64 heater rod bundle of electrically directly heated tubes.
- Scaling-down factor of about 700 applied to power input (5,5 MW), primary coolant mass flow (7 and 21 kg/s respectively) and primary coolant volume (0.6 m^3)
- 1 : 1 representation of components elevation and height, of lengths of heat transfer surfaces (core, steam generator), of pressure drop distribution along flow paths, of power-to-volume ratio, of ratio of components volume among each other.
- Simulation of accumulator and high pressure injection system for ECC.
- 400 channels measurement instrumentation and data acquisition system capability
- Results prediction and evaluation with RELAP4/Mod 6 and RELAP5/Mod 1.

5. Progress to Date
- Continuation and completion of A1-test programme regarding large breaks
- Definition of A1-test matrix for small breaks
- Continuation with results analysis and documentation work for preceding tests
- Continuation and completion of implementation and conversion of original CDC version of RELAP5/Mod 1 code into IBM version for running the code on the AMDAHL computer at Ispra; execution of several test case calculations and of first post-test and design calculations
- Proposal of a preliminary test programme for Special Transients tests with the LOBI-MOD2 facility
- Presentation of 8 conference papers on LOBI results at 3 international conferences (FRG, USA), [1 to 8]
- Start of modification work of LOBI test facility towards version MOD2; execution of laboratory hall, and computer and control room building modifications
- Recalibration and further development of measurement devices and techniques
- Development and testing of software for extended data acquisition and process control system
- Further development of software for, and extension of the LOBI Data Base.

6. Results
- With the execution of 5 A1-tests and 2 B-tests, the A1-test programme phase regarding large break tests has been completed. From December 1979 to June 1982 a total of 28 tests was performed: 18 A1-test, 3 A2-tests, 4 B-tests and 3 "small break scoping tests". The results of 13 of these 28 tests are available to the Community Member States. A summary on the parameter range covered by the 25 large break tests is given in Table 1.
- A total of 14 A1 small break tests has been defined for being executed after the test facility modification at present under way; 2 of them are quasi-steady-state tests without rupture simulation. 3 of them have been proposed as A2-tests, available to the Member States. See Table 2.
- A total of 13 Quick Look Reports (QLR), 15 Experimental Data Reports (EDR), 9 Test Prediction Reports (TPR) and 4 Preliminary Data Reports (PDR) has been prepared in form of Confidential Communication (39) or Technical Note (2), both with restricted distribution list [9].

- The first post-test prediction with RELAP5/MOD1 code after its conversion into IBM version was performed for the double-ended (200 %) cold leg break test A1-66 and yielded very satisfactory results. The same code was applied for design calculations aiming at the optimisation of the two-phase flow conditions downstream of an orifice (simulating a small break) with a view to improve the break mass flow measurement.
- For the preparation of a RELAP5 input data set for LOBI small break tests, detailed modelling of structural heat and of heat losses to the environment, and the estimation of heat losses due to cooling of measurement adapters, of the upper power connecting plate and of the pump bearings (seal water) is necessary and presently under way.
- In view of future investigation of Special Transients with the LOBI-MOD2 facility, a preliminary test programme was proposed and discussed, comprising a total of 11 different transients subdivided into two classes of short and long term transients.
- Evaluation of results regarding the influence of different ECC injection modes on blowdown has shown that combined injection led to a much more efficient core cooling: completely rewetted core after about 70 s into the transient / 2 /.
- The influence of break size on blowdown in case of large cold leg breaks may be characterized by the occurrence of early DNB over the whole core for break sizes of 100 % and larger, no DNB at all for break sizes of 25 % and less; within the intermediate range DNB may occur in the upper core region / 5 /.
- The influence of downcomer volume and gap width for double-ended cold leg breaks has shown increasing significance during the saturated blowdown and during the refill period: the large downcomer (50 mm) results in a better core cooling due to a positive core mass flow during the late blowdown and initial refill period; the small downcomer (12 mm) results in a more typical fluid mass distribution within the primary system during the blowdown period, but in a less typical refill behaviour by inhibiting ECC water penetration and refill of pressure vessel / 6 /.
- Discrepancies observed between RELAP4/MOD6 prediction and experimental results for the later blowdown and refill period are to be attributed to the code deficiency to describe phase separation processes, three dimensional flow patterns, thermal non-equilibrium processes, heat transfer in the post-DNB (film boiling) region and rewetting processes / 3 /.
- The construction work for the laboratory hall modifications necessary for the steam generators replacement, and for the extension of the computer and control room building was completed.
- The modification and extension work at the LOBI test facility towards the LOBI-MOD2 version for the future small breaks and Special Transients tests was started and is still under way; due to fabrication difficulties, the new steam generators delivery was delayed by further 3 months.
- Considerable effort was and is being dedicated to measurement instrumentation service, recalibration and development work, the latter is aiming primarily at an improvement of mass flow measurement techniques and regards pick-off systems, turboprobes and full flow turbine meters, dragbodies, densitometers.
- Comprehensive software development and testing work was performed for the new, extended data acquisition and process control system in view of the forthcoming

small break test programme.
- Further functions for results evaluation, analysis and graphical representation have been added to the LOBI Data Base.

7. Next Steps
- Continuation and completion of results analysis and documentation work of preceding tests
- Completion of test facility modification
- LOBI-MOD2 commissioning tests; start of small break A1-tests
- Pre-test prediction calculations with RELAP5/MOD1 for small break A1-tests

8. Relation with Other Projects
- no changes with respect to previous annual reports

9. Literature

/ 1 / W. L. Riebold, P. Mörk-Mörkenstein, L. Piplies: "LOBI Project: Einfluß der DWR Umwälzschleifen auf den Blowdown". Paper presented at the Conference "Kerntechnik '82", 4 - 6 May, 1982, Mannheim, FRG

/ 2 / W. L. Riebold, F. Chenneaux, R. Kirmse: "Einfluß des Notkühlwasser-Einspeisemodus auf den Blowdown in LOBI-Experimenten". Paper presented at the Conference "Kerntechnik '82", 4 - 6 May, 1982, Mannheim, FRG

/ 3 / H. Städtke, W. Kolar, W. Brewka: "Vorausrechnung der LOBI-Versuche mit RELAP4/MOD6 - Vergleich Experiment und Rechnung -". Paper presented at the Conference "Kerntechnik '82", 4 - 6 May, 1982, Mannheim, FRG

/ 4 / L. Piplies, W. Kolar: "Pumpenverhalten unter Störfallbedingungen: Zweiphasen-Kennfelder der LOBI-Pumpe". Paper presented at the Conference "Kerntechnik '82", 4 - 6 May, 1982, Mannheim, FRG

/ 5 / L. Piplies, C. Addabbo, W. L. Riebold: "Influence of Break Size on Blowdown for Large Breaks". Paper presented at the "International Meeting on Thermal Nuclear Reactor Safety", August 29 - September 2,1982, Chicago, Illinois, USA

/ 6 / H. Städtke, D. Carey, W. L. Riebold: "Influence of Downcomer Volume and Gap Width on Blowdown". Paper presented at the "International Meeting on Thermal Nuclear Reactor Safety", August 29 - September 2,1982, Chicago, Illinois,USA

/ 7 / W. L. Riebold, T. Fortescue, K. H. Günther: "Design and Instrumentation of LOBI U-tube Steam Generators for Small Break and Special Transients Tests". Paper presented at the "International Meeting on Thermal Nuclear Reactor Safety", August 29 - September 2, 1982, Chicago, Illinois, USA

/ 8 / W. L. Riebold, L. Piplies, H. Städtke: "LOBI Experimental Programme Results and Plans: Status September 1982". Paper presented at the "10th Water Reactor Safety Research Information Meeting" of the USNRC, October 12 - 15,1982, Gaithersburg, Maryland, USA

/ 9 / 39 Confidential Communications and 2 Technical Notes, both with restricted distribution list to document LOBI results (QLR, EDR, TRP, TCR, PDR).

Table 1: **LOBI LARGE BREAK TESTS** (1980-1982)

BREAK LOCATION		BREAK SIZE 2 x 1A 50	2 x 1A 12	2 x 0.5A 50	2 x 0.5A 12	1 x 1A 50	1 x 1A 12	1 x 0.5A 50	1 x 0.5A 12	1 x 0.25A 50	1 x 0.25A 12	ECC INJECTION MODE
COLD LEG		A1-04* A1-04R	A1-06*	B-101*	B-222*	A2-59* A2-59R*		A2-55*		B-R1M*		COLD LEG INJ.
COLD LEG		A1-01* A1-02 A1-03 A1-05	A1-08 A1-72 A1-74				A1-69		A1-66		A1-67	COMBINED INJECT.
COLD LEG			A1-07									NO INJECT.
PUMP SUCTION LEG												COLD LEG INJ.
PUMP SUCTION LEG			A1-70									COMBINED INJECT.
HOT LEG					B-302*							COLD LEG INJ.
HOT LEG			A1-10A A1-10B								A1-73	COMBINED INJECT.
		50	12	50	12	50	12	50	12	50	12	
		DOWNCOMER GAP WIDTH (MM)										

*: tests available to the Community Member States

Table 2: **LOBI SMALL BREAK A1-TESTS PRELIMINARY** (1983-1984)

BREAK LOCATION		BREAK SIZE 1 x 0.1A H.L.	1 x 0.1A C.L.	 H.L.	 C.L.	1 x 0.02A H.L.	1 x 0.02A C.L.	1 x 0.01A H.L.	1 x 0.01A C.L.	1 x 0.004A H.L.	1 x 0.004A C.L.	ACCU INJECTION MODE
COLD LEG								X(2/4)	X(2/4)	X (2/4) X (4/4)		NO INJECT
COLD LEG		X (2/4)				X (2/4) X (2/4) X (4/4)	X (2/4)					COMBINED INJECT.
PRESSURIZER Relief Valve										X (2/4)		NO INJECT
PRESSURIZER Relief Valve												COMBINED INJECT
HOT LEG												NO INJECT.
HOT LEG		X (1/4)				X (2/4)						COMBINED INJECT.
		H.L.	C.L.	H.L.	C.L.	H.L.	C.L.	H.L.	C.L.	H.L.	C.L.	
		HPIS INJECTION LOCATION										

Berichtszeitraum/Period 1.1. - 31.12.1982	Klassifikation/Classification 1-1-2	Kennzeichen/Project Number RS 464
Vorhaben/Project Title Behandlung Kleiner Lecks in DWR-Anlagen mit der Methode der Transientenanalyse Treatment of Small Leaks in PWR's with Methods of Transient Analysis		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Gesellschaft für Reaktorsicherheit (GRS)mbH
Arbeitsbeginn/Initiated 1.7.1980	Arbeitsende/Completed 31.03.1983	Leiter des Vorhabens/Project Leader Dr. W. Frisch
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds DM 1.550.533,50

1. General Aim

Investigation of the behaviour of PWR's during small break accidents with the objective of an early detection of such plant conditions and the initiation of optimum countermeasures to mitigate the consequences of the accident.

2. Particular Objectives

The main objective of the project is the analysis of the initial part of the accident, when no extreme plant conditions or damages have occurred so that core damage may still be avoided by appropriate countermeasures. The physical processes during the accident will be described w'th regard to the interfering systems. The description of the course of the event together with the available instrumentation, control room design and possible operator actions will be evaluated.
The plant model ALMOD, i.e. an improved version of this code will be used to do the calculations necessary for these analyses. The main objective of the development of this extended version is the improvement of the thermal-hydraulic models of the ALMOD code. Thus it will be possible to calculate such states of the coolant, which are not in the range of validity of conventional plant models.

3. Research Program

3.1 Calculation of some small break events with the plant model ALMOD

3.2 Development of an extended plant model

3.3 Verification of the extended plant model ALMOD by experiments.

3.4 Sensitivity analysis for the determination of relevant plant parameters and system functions.

4. Experimental Facilities, Computer Codes

LOFT, ALMOD

5. Progress to Date

Ad 3.2 Work has been carried out to eliminate the limitations found in the plant model ALMOD in relation to the calculations ad 3.1. This work has resulted in considerable changes in the structure of the code. In this context it can be mentioned e. g. the calculation of the system pressure after the pressurizer empties. All these structural changes have been brought to an end.

The activities for the extension of the thermal-hydraulics of the primary coolant system to describe the flow reversal and for the improvement of the integration method have been continued. The new methods have been tested in a closed-loop geometry. The dynamic slip model has been tested under two-phase flow conditions, and the difficulties that have arisen in connection with the movement of the phase boundary have been solved with a special numerical technique for describing this phenomena.

Ad 3.3 A selection of adequate experiments and incidents has been made for the qualification of the extended ALMOD code. Up until now the unintentional opening of a pressurizer PORV occurred in a German plant has been calculated.

Ad 3.4 The calculations for the sensitivity analysis have already been started. Firstly, the comparison between a cooldown transient and a small break in the hot leg has been carried out.

6. Results

The extensions of the plant model ALMOD are described in a report /1/.

7. New Steps

The work will bei continued according to the program.

8. Relation to Other Projects

RS 234, RS 471.

9. References

/1/ J. Miró, A. Schaefer
Erweiterung des Anlagenmodells ALMOD zur Beschreibung von Störfällen mit "Kleinem Leck".
GRS-A-791 (December 1982).

Berichtszeitraum/Period 1.1. - 31.12.1982	Klassifikation/Classification 1.1.2	Kennzeichen/Project Number RS 182
Vorhaben/Project Title Analysis of Integral and Separate Effects Tests (LOFT, SEMISCALE, ROSA IV) Auswertung von Integral- und Einzeleffekt-experimenten (LOFT, SEMISCALE, ROSA IV)		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH
Arbeitsbeginn/Initiated 1.4.1976	Arbeitsende/Completed 30.6.1984	Leiter des Vorhabens/Project Leader Dr. K. Wolfert
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 31.12.1982	Bewilligte Mittel/Funds

1. General Aim

Complemantary analytical and phenomenological investigations of LOCA experiments and code development and -assessment in reactor safety research

2. Particular Objectives

2.1 Analytical and phenomenological analysis of LOFT und Semiscale experimental results

2.2 Analytical and phenomenological analysis of integral and separate effects tests of the ROSA IV project

2.3 Assessment of computer codes for LOCA analysis

3. Research Program

3.1 Assignment of a resident engineer to EG&G, Idaho Falls, USA

3.2 Phenomenological analysis of experimental results

3.3 Assessment of computer codes

4. Experimental Facilities, Codes

Experimental Facilities:

- LOFT, - Semiscale, - TPTF and LSTF of ROSA IV, - TOSHIBA

Codes:

- DRUFAN, - FLUT, - SSYST

5. Progress to Date

Ref. 3.1 In this period Dr. M. Firnhaber was working for EG&G till June. Dr. A. Wahba started working for EG&G in May.

Ref. 3.2 Phenomenological analysis of LOFT tests

- L9-3: ATWS transient with loss-of-feedwater and delayed emergency feed water

- L6-6: Transient with boron dilution
- L2-5: Double ended break in the cold leg with pumps off
- L6-8-B1: Uncontrolled rod withdrawal with 0.21 cm/s
- L6-8-B2: Uncontrolled rod withdrawal with 1.06 cm/s
- L6-8-C1: Recovery procedure of a steam generator tube rupture with pressurizer spray, HPIS and secondary feed and bleed
- L6-8-C2: Recovery procedure of a steam generator tube rupture by using the pressure operated relief valve, HPIS and secondary feed and bleed
- L9-4: ATWS loss-of-offsite power transient.

Ref. 3.3
- Documentation of post test analysis of L6-5 (loss-of-feedwater) /1/
- Post test calculation of L2-3 /2/
- Multy-channel analysis of L2-3 /3/
- Pre test calculation of L2-5 and participation in the OECD-CSNI Standard Problem No. 13 /4/
- Post test analysis of L6-8-C1. Parameter study with variation of break mass flow, spray temperature, spray mass flow, secondary feed and bleed
- Post test calculation of a Toshiba BWR-Blowdown experiment (steam line break).

6. Results

Ref. 3.2
- Test L9-3 demonstrated that the facility was controlled by the injection of boron and by opening the power operated and safety relief valve
- The transient L6-6 demonstrated that criticality was achieved after 2 h by injecting 0.47 ltr/sec demineralized water
- the experiment L2-5 demonstrated that contrary to L2-3 (double ended cold leg break with pumps on) no rewetting in the centre of the core occurred
- In test L6-8-B1 the reactor scrammed at 46 sec due to reaching the high pressure (150 bar) set point.
- In test L6-8-B2 the reactor scrammed at 7.3 sec due to reactor peak power trip (130 %)
- In experiment L6-8-C1 the primary pressure reached the secondary pressure at 904 sec, in L6-8-C2 at 275 sec.
- In test L9-4 the safety relief valves on the primary and secondary side were challenged. The reactor power fell to 20 % of the initial value in the first 50 sec.

Ref. 3.3
- A good agreement could be achieved between the post test calculation and the experiment L2-3. To determine precisely the DNB and the maximum cladding temperatures, the simulation of two fluid channels and four fuel rod types with different power levels were necessary.
- A good agreement could be achieved between the pretest calculation and the experiment L2-5. DNB and Dry Out at the beginning of the blowdown was calculated in good agreement with the experiment. The top down rewetting in the upper part of the high powered fuel rod could not be computed.
- The post test calculation of L6-8-C1 showed good agreement between calculation and experiment.

7. Next Steps

- Evaluation and documentation of the experiment L6-8-C1 and the Toshiba experiment
- Analysis of further LOFT experiments

8. Relation to other Projects

RS 314, RS 475

9. References

/1/ W. Pointner: Nachrechnung des LOFT-Transienten-Versuchs L6-5 mit DRUFAN-02, GRS-A-722, April 1982

/2/ W. Pointner: Nachrechnung des LOFT-Blowdown-Versuchs L2-3 mit DRUFAN-02, GRS-A-766, November 1982

/3/ W. Pointner: Mehrkanalanalyse für den LOFT-Versuch L2-3 mit DRUFAN-02, GRS-A-Report in preparation

/4/ W. Pointner: Contribution to OECD-CSNI LOCA Standard Problem No. 13, October 1982.

10. Availability of the Reports

GRS-Forschungsbetreuung.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1982	1.1.2	06.01.03/17A (PNS 4137)
Vorhaben/Project Title Joint test rig for tests and calibration of different methods of two-phase mass flow measurements Gemeinsamer Versuchsstand zum Testen und Kalibrieren verschiedener Zweiphasenmassen-strom-Meßverfahren		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK), Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated 1.10.74	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader Dr. J. Reimann
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

1.1 For LOCE the accurate measurement of two-phase mass flow rate is of primary importance. Measuring techniques are to be developed and tested.

1.2 During LOCA two-phase flow in pipe junctions may occur. The phase and mass distribution in the branches has to be investigated.

2. Particular Objectives

2.1 Test, calibration and comparison of different measuring methods in steady-state two-phase flow.

2.2 Investigation of the flow of a two-phase mixture in a T-junction.

3. Research Program

3.1 Instrumentation tests in steady-state steam-water and air-water flow; parameters are mass flow rate, quality and pressure.

3.2 Measurement of the redistribution of the two-phase flow (mass flow rate, quality) in a T-junction with horizontal inflow and outflow.

Parameters are: mass flow rate and quality of inflow, system pressure, differential pressure between inflow and branch, branch angle in respect to horizontal.

Investigations are to performed with the following test sections:

a) T-junction with D=d=50 mm (D=diameter of main pipe, d=diameter of branch); air-water and steam-water experiments; special interest: measurement of void and momentum profiles

b) T-junction with D=50 mm, d=10 mm; air-water and steam-water experiments; special interest: Critical flow in branch

c) Test section with D=200, d=8-20 mm: "Small leak separate effect tests", air-water experiments, special interest: stratified flow in main pipe.

4. Experimental Facilities

Loops for steady-state steam-water flow ($p_{max} \approx$ 15 MPa) and air-water flow (p_{max} = 1 MPa)

5. Progress to Date

5.1 A gyrostatic mass flow rate measurement instrument (Micro Motion Inc., USA) was tested in water and air-water flow.

5.2 Air-water experiments were performed with test geometry a) and a horizontal branch. Steam-water experiments started.
With test geometry c) experiments were carried out, using a downward orientated branch; experiments with an upward orientated branch are presently performed.

6. Essential Results

6.1 The instrument had a high accuracy in single phase water flow (error $\approx$ 1 %). With increasing void fractions ($\gtrsim$ 10 %) the signal was no longer interpretable.

6.2 Test geometry a): Phase separation is the largest at a large mass flow rate fraction through the branch. The redistribution is only weakly dependent on inflow mass flux and quality. An improved model for the pressure drop across the branch was developed.

Test geometry c): For the downward orientated break a correlation was developed for the onset of gas pull-through. Furthermore a generalized graph was found for the break mass flow rate and quality as a function of the liquid level. For the upward orientated break a criteria for the onset of liquid entrainment was developed. The increase of break mass flow rate with entrainment is small as long as stratified flow in the main pipe exists.

7. Next Steps

Air-water and steam-water experiments with geometries a) and b); air-water experiments with geometry c) and a horizontal branch.

1.1. - 31.12.1982 06.01.03/.17A (PNS 4137)

8. Relation with other Projects

Lobi (Euratom Ispra), CEA Grenoble, EG&G Idaho

9. Literature

Reimann, J., John, H., Müller, U.: Measurements of Two-Phase Mass Flow Rate: A Comparison of Different Techniques; Int. J. Multiphase Flow, Vol. 8, No. 1, 1982.

Reimann, J.: Developments in Two-Phase Mass Flow Rate Instrumentation; Nato Advanced Workshop on Advances in Two-Phase Flow and Heat Transfer, Spitzingsee/Schliersee, FR Germany, August 31 - Sept. 3, 1982.

John, H.; Hain, K.; Brüderle, F; Reimann, J.; Volmer, T.: Tests of an advanced true mass flow meter (TMFM) in gas liquid flow. AIAA/ASME 3rd Joint Thermophysics, Fluids, Plasma and Heat Transfer Conference, St. Louis, Missouri, June 1982, ASME 82-FE 23, 1982.

Reimann, J.; Khan, M.: Flow Through a Small Pipe at the Bottom of a Large Pipe with Stratified Flow; Annual Meeting of the European Two-Phase Flow Group, Paris la Defense, June 2-4, 1982

10. Degree of Availability of the Reports

Unrestricted

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	1.1.2	06.01.03/18A PNS 4139
Vorhaben/Project Title Experimental Facility for Nonsteady State Two-Phase Flow Versuchsstand für instationäre Zweiphasen-Strömungen		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KFK) Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated 01.01.79	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader Dip.-Ing. H. John
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

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

2. Particular Objectives

Different measuring methods for two-phase mass flow rate developed by other institutes are to be tested and calibrated in nonsteady state steam-water flow.

3. Research Program

Instrumentation tests in nonsteady state (blowdown) steam water flow.

4. Experimental Facility

A blowdown loop for steam water flow (150 bar, 0,7 m^3) with changeable pipe sections (50 and 80 mm inner diameter) and a high accurate reference mass flow rate measurement with a True Mass Flow Meter (TMFM-50) up to 50 kg/s during blowdown. The blowdown will be initiated by a quick opening valve. Steady state single phase mass flow measured by orifices is possible before blowdown up to 10 % of the blowdown mass flow, to make additional tests of the components.

5. Progress to Date

- The built up of the loop, inclusive the electrical equipment for control and data transfer was finished. The data lines from the transducers to the computer PDP11 were tested.
- Funktion tests of the blowdown loop were started with cold water, where compressed air was the driving force.
  Several regular blowdowns with boiling water and different entrance pressures up to 12 MPa were successfully performed.

- First blowdowns with massflow measuring devices from the LOBI-Loop in Ispra (Euratom) and with radio tracer tests (KFK-LIT) started. The results are calculated now.

6. Essential Results

The successful test of the True Mass Flow Meter, a new 5 Beam-γ-Densitometer and the quick opening valve under blowdown conditions.

7. Next steps

Transient calibration tests of the LOBI-Spool pieces, the radio-tracer measuring method, the ultrasonic system for liquid level detection and the temperature correlation.

8. Relation with other Projects

RS 109 - EURATOM/Ispra, RS 135 - TU/Berlin, RS 150348 - TU/Berlin, PNS 4136 - KfK/LIT, PNS 4138 KFK/IT, Battelle/Frankfurt, KWU-Erlangen, R 512

9. References

Projekt Nukleare Sicherheit, Halbjahresberichte 1979/82.
H. John, K. Hain, F. Brüderle, J. Reimann, F. Schloß
Test des Massenstrom-Meßgerätes TMFM-50 für Zweiphasenströmungen
KFK 3215 Okt. 81

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	1.1.2	06.01.03/20A (PNS 4136)
Vorhaben/Project Title Entwicklung von Radionuklidverfahren zur Bestimmung transienter Zweiphasenmassenströme Development of radionuclide methods for measuring transient two-phase mass flows		Land/Country Fördernde Institution/Sponsor Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) PNS/LIT
Arbeitsbeginn/Initiated 01.01.81	Arbeitsende/Completed 31.12.83	Leiter des Vorhabens/Project Leader R.Löffel, LIT
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds -

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 measuring

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

2. Particular Objectives

2.1 Measurement of density

The phase distribution can be determined using a 6-beam-γ-densitometer with special beam geometry and high time resolution (1 ms). The local and global densities and void fractions are calculated from the intensity signals.

2.2 Measurement of velocities

The radiotracer method is suitable to measure separately the velocities of the individual phases and components. For this purpose, short-lived gaseous and liquid radionuclides are injected into the flow.

3. Research Programm

- Test of the radionuclide method on the "Transient Two Phase Flow Loop" of KfK-IRB (PNS 4139),
- Application of the RN-method on the DUESE-experiment.

4. Experimental Facilities

Installation of a radiotracer velocity measuring device and of a 6-beam-γ-densitometer at the Test Loop.

5. Progress to Date

111 experiments with water-steam (64 exp.) and with water-air (47 exp) flows were fperformed on the DUESE-test facility.

6. Results

Fig. 1 shwos a typical result of a density measurement of an inhomogeneous steady-state two-phase flow (in this case water/air flow) in a tube of 80 mm mean diameter. Three different types of flow have been recorded successively in this example. In all three cases wave flows were obtained which differ only by their void factors. Void factors of 0.48, 0.37 and 0.27 were determined for these three flow conditions.

7. Next Step

Test at the "Transient Two Phase Flow Loop" of KfK-IRB will be performed.

8. Relation with other Projects

PNS 4139 "Transient Two Phase Flow Loop"

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

9. Reference Documents

-

10. Degree of Availability of the Reports

-

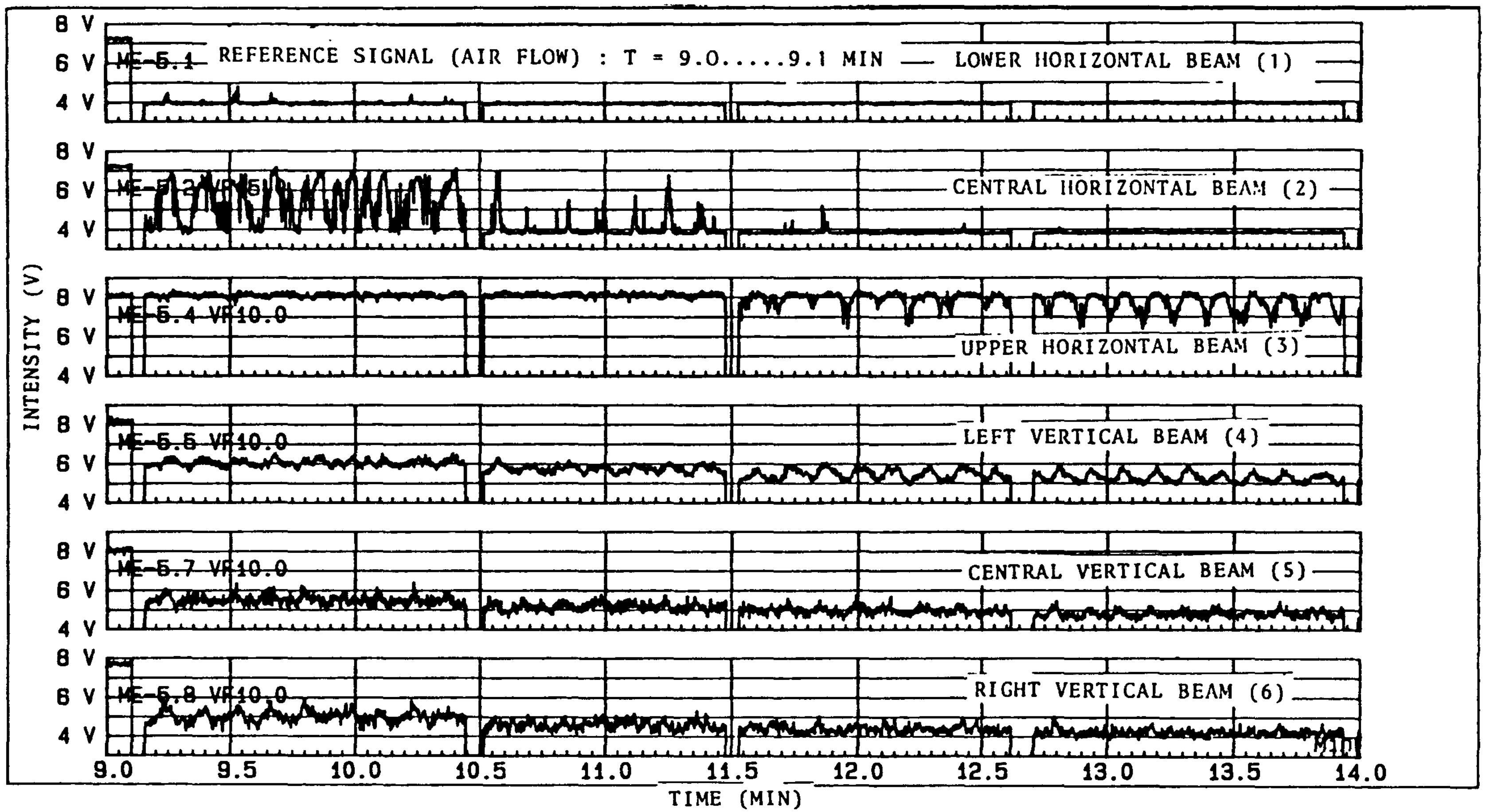

FIG.6: PLOT OF "DENSITY-SIGNALS" FROM A INHOMOGENEOUS STEADY-STATE AIR-WATER FLOW IN A HORIZONTAL 4"-PIPE.
THE DIAGRAM SHOWS 4 EXAMPLES OF WAVE FLOWS WITH DIFFERENT VOID FRACTIONS : 0.48 / 0.37 / 0.27 / 0.22

145-1-03			1 - 1 - 2
TITRE THERMOHYDRAULIQUE DU LOCA. ETUDE EXPERIMENTALE DES DEBITS CRITIQUES ET ECOULEMENTS EN DOUBLE PHASE : PROGRAMME MOBY DICK ET SUPER MOBY DICK			*Pays* FRANCE
			Organisme directeur CEA/IPSN -EDF/SEPTEN
TITRE en anglais LOCA THERMOHYDRAULICS EXPERIMENTAL STUDY OF CRITICAL FLOW AND TWO PHASE FLOW : MOBY DICK AND SUPER MOBY DICK PROJECT			*Organisme éxécuteur* CEA/DRE/STT (CEN-GRENOBLE)
			Responsables M. COURTAUD DRE/STT/SETRE
Date de démarrage 01.01.1972	*Etat actuel* en cours		*Scientifiques* M. GROS D'AILLON M. JEANDEY
Date d'achèvement 1983	*Dernière mise à jour* 01.1983		

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. Etudier les phénomènes de stratification et de séparation dans les tês à haute pression.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

Installations expérimentales

MOBY DICK : Boucle dans laquelle sont réalisés des écoulements double phase à grande vitesse soit en eau-vapeur par auto-vaporisation, soit en eau-air.

La pression maximum est de 10 bars.

SUPER MOBY DICK : même type d'installation mais pouvant aller jusqu'à des pressions de 150 bars.

Programmes

MOBY DICK

- essai en eau-vapeur sur un tube de section de 20 mm ID terminé par un divergent de 7 degrés 1974

- essai en eau-vapeur sur un tube de section de 14 mm ID terminé par un divergent de 7 degrés 1975

- essai en eau-azote sur un tube de section de 14 mm ID terminé par un divergent de 7 degrés 1976

- essai en eau-vapeur avec élargissement brusque de la section d'essai (Ø 14 et Ø 20) à 2 bars et 7 bars 1977/1978

- essai en eau-air sur un tube de section de 14 mm ID terminé par un divergent de 7 degrés .... fin 1978

SUPER MOBY DICK

Construction de la boucle jusqu'en mars 1979.

1ère campagne : détermination des lignes piezométriques sur une section d'essai comprenant un convergent conique de 30°, un tube de section 20 mm ID et un divergent de 7 ° pour des pressions comprises entre 20 et 150 bars..... 1979

2ème campagne : reprise de certains essais de la 1ère campagne pour déterminer le taux de vide............... Sept 1980

3ème campagne : essais avec élargissement brusque, mesures de débit critique et de taux de vide .... Avril à Sept 1981

Etude du frottement interfacial dans un écoulement vertical (tuyère courte);
écoulement stratifié;
séparation dans un "té" ....Fin 82 et 83

4. ETAT D'AVANCEMENT

Avancement à ce jour

MOBY DICK : les essais sont terminés .... Fin 1978
Un rapport final a été écrit.

SUPER MOBY DICK : la 1ère campagne est terminée............... 1979

la 2ème campagne est terminée ....Sept 1980

la 3ème campagne s'est déroulée en 1981

Essais avec tuyau vertical :
- élargissement brusque tuyère longue....... 4 à 9/81
- élargissement brusque avec ou sans grille. 9 à 12/81
(avec mesure de taux de vide)
- élargissement brusque, tuyère courte:
mesures de transferts d'impulsion interfaciaux
(et reprise d'essais antérieurs) .... Fin 82 début 83

Principaux résultats

SUPER MOBY DICK

1ère et 2ème campagne

Un domaine étendu de valeurs des paramètres définissant l'état du fluide en amont du col a été parcouru : pression de 20 à 120 bars état du fluide très éloigné ou très proche de la double phase (ce qui permet d'atteindre diverses valeurs pour le taux de vide au col); les valeurs du débit critique sont comprises entre 10 000 et 60 000 Kg/(s-m^2).

- Installation avec divergent d'angle faible :

Mesures effectuées : pression longitudinalement, densité longitudinalement et radialement,cartes de débit critique en fonction des conditions d'entrée.

La mesure radiale de densité a montré que le taux de vide est très élevé près de la paroi.
Signalons aussi un fort déséquilibre thermique au col (20° de surchauffe).

La comparaison entre débits mesurés et calculés avec un modèle de détente isentropique montre des différences d'autant plus sensibles que les conditions d'entrée sont plus proches de la saturation.

Les résultats expérimentaux ont été utilisés pour l'élaboration de modèles d'échange (masse et chaleur) entre phases. L'analyse a été menée avec un modèle d'écoulement monodimensionnel à deux fluides avec déséquilibre thermique et sans glissement.

Le transfert de chaleur entre les deux phases du fluide a été exprimé en fonction des conditions locales du fluide : taux de vide, pression, vitesse, surchauffe.

- Elargissement brusque :

Le comportement bidimensionnel du jet a été particulièrement étudié, montrant la forte influence de la pression et du débit sur sa structure.

....

Il n'apparaît pas nettement de différence sur les résultats obtenus pour le débit critique selon que l'installation comprend un divergent ou un élargissement brusque (c.f. note TT 82/83).

Un rapport sur la grille complète d'essais (au nombre de 70) est en cours de rédaction (lignes piezométriques, densité axiale et verticale...).

Des calculs d'interprétation et de " reconstitution" ont été faits avec le code CATHARE (cas d'écoulement non établis).

5. PROCHAINES ETAPES

(Nouvelle campagne)

Essais avec tuyau vertical .... Fin 82-début 83
- élargissement brusque, tuyère courte
mesure du débit critique et des transferts interfaciaux

Essais avec tuyau horizontal .... à partir de 9/83
- effet de stratification
mesure de τ_i

Essais avec piquages et "tés" .... fin 1983

6. RELATIONS AVEC D'AUTRES ETUDES

Des comparaisons entre les résultats SUPER MOBY DICK, MOBY DICK et MARVIKEN (débits critiques) ont été effectuées- voir communication à Pasadena, § 7.

REBECA : Etude de débits critiques d'un mélange à 3 composants :

eau, air, vapeur. (voir 142-3-2).

Action commune avec FRAMATOME : mesure du taux de vide par ultra-sons.

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

"Etude expérimentale d'écoulement eau-air à grande vitesse"
MOBY DICK (1ère partie)
C. JEANDEY, G. BARRIERE - Note TT 599, 1979

"Autovaporisation d'écoulements eau-vapeur"
C. JEANDEY, L. GROS D'AILLON, R. BOURGINE, G. BARRIERE, Rapport TT 163
Juillet 1981

"Two-phase flow under steep pressure gradients"
Third CSNI Specialist Meeting on Transient two-phase flow
PASADENA, March 1981 C. JEANDEY, L. GROS D'AILLON

"Jet diphasique critique confiné".
L. GROS D'AILLON, C. JEANDEY TT /STRE/ 82.31

"Mesure de débits critiques sur l'installation SUPER MOBY DICK dans une section d'essais comportant un divergent ou un élargissement brusque".
C. JEANDEY, L. GROS D'AILLON, G. BARRIERE, R.BOURGINE,VERDUN,
TT/SETRE/82.32

"Transferts d'impulsion entre phases, écoulement ascendant eau-vapeur; essais préliminaires".
C. JEANDEY, L. GROS D'AILLON, R. BOURGINE, G. BARRIERE TT/SETRE/82.33

145-1-04			1 - 1 - 2
TITRE THERMOHYDRAULIQUE DU LOCA ETUDE EXPERIMENTALE DE LA PHASE DEPRESSURISATION DES ACCIDENTS DE PERTE DE REFRIGERANT DES REACTEURS A EAU : PROGRAMME OMEGA			*Pays* FRANCE *Organisme directeur* CEA/IPSN -EDF/SEPTEN
TITRE en anglais LOCA THERMOHYDRAULICS EXPERIMENTAL STUDY OF BLOWDOWN IN A LOSS OF COOLANT ACCIDENT (LOCA) OF A P.W.R.:OMEGA PROJECT			*Organisme éxécuteur* CEA/DRE/STT CEN/GRENOBLE *Responsables* M. COURTAUD DRE/STT/SETRE
Date de démarrage 01.01.1972		*Etat actuel* TERMINE	*Scientifiques*
Date d'achèvement 31.12.1982		*Dernière mise à jour* 01.01.1983	R. RICQUE J.C. ROUSSEAU

1. OBJECTIF GENERAL

Etude expérimentale du type analytique menée pour tester et améliorer les corrélations d'échange thermique dans la phase de dépressurisation d'un accident sur le circuit primaire (LOCA) d'un réacteur P.W.R.

2. OBJECTIFS PARTICULIERS

- Obtention de résultats expérimentaux détaillés sur géométrie tubulaire pour :
  . tests et vérification des corrélations d'échanges thermiques et de flux critique utilisées dans les codes existants (RELAP, TRAC...).
  . développement de modèles et de corrélations (code 2ème génération CATHARE).
- Obtention de résultats expérimentaux sur grappes pour qualification et validation de codes sur simulation réaliste (effet des grilles,...).

3. INSTALLATION EXPERIMENTALE ET PROGRAMME

Boucle OMEGA ; pression 170 bars, débit max. 20 kg/s, puissance 4.5 MW.

Deux dispositifs :

- dispositif avec canal d'essai tubulaire, Ø = 12 mm, taille et position de brèche variables (amont, aval, aux deux extrémités)
- dispositif avec grappe de 36 barreaux, combustible type 17 X 17 L = 3.65 m, flux cosinus; une taille de brèche et position variable

Programme

- essais avec section d'essais (S.E) tubulaire .... Nov. 1976 (1ère campagne)
- essais avec grappe 36 barreaux .... Nov. 1977 (1ère campagne)

- essais avec S.E. tubulaire.............................. Déc. 1978 (2ème campagne)
- essais avec grappe 36 barreaux 1981

4. ETAT D'AVANCEMENT

Avancement à ce jour.

Les quatre étapes du programme expérimental ci-dessus sont achevées ainsi que leurs dépouillements,ceux-ci font l'objet de différents documents.

Le travail de passage des résultats bruts (mesures) aux résultats physiques sur les grandeurs intéressantes - après transformations et corrections - a été effectué pour les quatre séries d'essais . Des calculs de conduction inverse, pour remonter du flux au coefficient d'échange ont aussi été faits.

Principaux résultats

Pour définir les essais (TUBE ou GRAPPE), il est possible de jouer sur les caractéristiques de la (ou des) brèche (s) (Aval, Amont, Double taille), sur le flux, sur le débit, donc sur la température du fluide à la sortie de la section chauffante et sur les caractéristiques des capacités cylindriques (pour la 2ème campagne TUBE) : capacité d'origine, ou réduite , en hauteur ou en section.

Première campagne TUBE (1976)

Parmi les phénomènes observés, notons particulièrement la stratification du fluide(dans la capacité du bas dans le cas d'une brèche aval).

Deuxième campagne TUBE (1978)

Une étude exhaustive, caractéristique par caractéristique, n'a pas été faite. Cependant, certains résultats sont acquis :

. entre une grande brèche et une petite brèche, on observe une disparition très nette du " burn-out", ou seulement un ralentissement de la croissance de la température de paroi.

. entre une brèche aval et une brèche amont :
brèche aval : la capacité aval se met immédiatement aux conditions de saturation, puis un front de vaporisation se propage vers le bas; il reste toujours de l'eau dans la capacité amont et la section expérimentale n'est jamais totalement en vapeur.
brèche amont : la capacité amont reste assez longtemps en simple phase liquide; la section d'essais est en vapeur sèche à la fin de la dépressurisation.
brèche double : le point de stagnation (point où le débit est nul) varie fortement dans le temps.

-

Première campagne GRAPPE (1977) .

Certains essais cnt été dépouillés et analysés (c.f. note) .

Deuxième campagne GRAPPE (1981) .

Une importante grille d'essais a été parcourue
- Taille de brèche grande (500 mm^2) ou moyenne (190 mm^2)
- Brèches simples (aval ou amont) ou doubles (avec différents rapports de dissymétrie)
- Flux variable : nul ,
  moyen (60 W/cm^2) ,
  fort (100 W/cm^2) ,
  par référence aux flux dans le coeur d'un réacteur
- Différentes températures de fluide à la sortie de la grappe (ΔT = 35 ou 55 ° C) .

Le dépouillement est achevé et fait l'objet d'un rapport.
Toutefois,il subsiste une difficulté d'interprétation des mesures des températures de paroi (affectées par la présence des thermocouples).

5. PROCHAINES ETAPES

Le travail expérimental est achevé; le dépouillement également - toutefois un travail d'analyse du problème mentionné ci-dessus est envisagé pour la mi-83.

6. RELATION AVEC D'AUTRES ETUDES

. Calculs des accidents de refroidissement des réacteurs à eau :
  Programme général coordonné de mise au moint des codes....Fiche 145-1-01

. Développement du code de 2ème génération CATHARE pour l'étude de l'accident de perte de caloporteur dans les réacteurs à eau pressurisée : Programme général coordonné ....Fiche 145-1-02

. Qualification validation RELAP-TRAC.
  Etude des transferts de chaleur : dépressurisation (OMEGA) renoyage (ERSEC) ....Fiche 145-1-11

7. DOCUMENTS DE REFERENCE

- Dépressurisation d'un sous-ensemble du circuit OMEGA comprenant une section d'essais chauffante tubulaire
  R. RICQUE et al. Rapport DTCE/STT 146, sept 1977

- Dépouillement de la première campagne d'essais de "décompression tube" sur la boucle OMEGA
  M. BONNETON - Note DTCE/STT 580, Août 1978

- Dépressurisation d'un sous-ensemble du circuit OMEGA comprenant une grappe chauffante à 36 barreaux :
  . Description du dispositif expérimental.
  . Présentation des résultats expérimentaux.
  R. FRANK et al. Rapport DTCE/STT 152 Septembre 1978

- Thèse de docteur-ingénieur
  M. BONNETON 1979

- Boucle OMEGA.
  Essais de dépressurisation en géométrie tubulaire; résultats de la seconde campagne
  M. JUHEL - Rapport TT 158 1980

- OMEGA - Dépressurisation d'un ensemble comportant une grappe de 36 barreaux chauffants.
  Campagne d'essais réalisée de Février 81 à Janvier 82
  CH. CHAULIAC Note TT/SETRE 82-26

<table>
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<tr><td colspan="2" rowspan="2">TITRE THERMOHYDRAULIQUE DU LOCA :
ETUDE DU RENOYAGE : PROGRAMME PERICLES</td><td>Pays FRANCE</td></tr>
<tr><td>Organisme directeur
CEA/ IPSN, IRDI
EDF/SEPTEN-FRAMATOME
CCE</td></tr>
<tr><td colspan="2" rowspan="2">TITRE en anglais LOCA THERMOHYDRAULICS
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CEA/DRE/STT
CEN Grenoble</td></tr>
<tr><td>Responsables
M. COURTAUD -STT/SETRE</td></tr>
<tr><td>Date de démarrage 01.11.1979</td><td>Etat actuel en cours</td><td rowspan="2">Scientifiques
R. DERUAZ</td></tr>
<tr><td>Date d'achèvement 31.12.84</td><td>Dernière mise à jour 01/1983</td></tr>
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1. OBJECTIF GENERAL

Etude de renoyage dans des conditions représentatives de grosses brèches (effets bidimensionnels, basse pression) et des petites brèches (haute pression) .

Etude des intéractions avec le plenum supérieur.

2. OBJECTIFS PARTICULIERS

- Etude des phénomènes de renoyage en vue de l'élaboration de modèles analytiques dans les cas suivants :

  . conditions grosse brèche et petite brèche
  . débit d'injection imposé
  . distributions radiales et axiales de flux. (distributions typiques d'un assemblage ou de l'ensemble du coeur).
  . blocages
  . décroissance de la puissance

- Etude des conditions du couplage du coeur avec le plenum supérieur (accumulation d'eau,désentrainement, effet de l'embout supérieur de grappe).

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Installations expérimentales

La boucle Périclès-coeur comprend :

- soit un coeur rectangulaire composé d'un réseau type 17 X 17 de 7 crayons pleine hauteur par 51 (largeur de 3 assemblages), la pression de fonctionnement de cette section d'essais est limitée à 6 bars.

- soit un coeur cylindrique de réseau 17 X 17 comportant 357 crayons pleine hauteur - pression de fonctionnement limitée à 70 bars.

Les crayons sont à l'échelle 1, chauffés électriquement, flux cosinus, avec entrée et sortie de courant par le bas du crayon de façon à pouvoir représenter correctement les embouts supérieurs des éléments combustibles.

Programmes

- Réalisation et qualification des prototypes d'éléments chauffants .... (01.12.1980)
- Spécifications et étude du circuit et des sections d'essais .... (31.12.1980)
- Réalisation des dispositifs expérimentaux .... (31.12.1981)
- Essais bouillote en géométrie rectangulaire .... (31.12.1982)
- Essais renoyage en géométrie rectangulaire............ (30.06.1983)
- Essais en géométrie cylindrique.

4. ETAT D'AVANCEMENT

- Boucle montée, qualification effectuée
- Grille d'essais (coeur rectangulaire) définie : prise en compte de la décroissance de puissance et de la présence de l'embout supérieur.
- Essais bouillote, coeur rectangulaire, terminés.
- Essais renoyage, coeur rectangulaire, en cours.

5. PROCHAINES ETAPES

- Suite des essais coeur rectangulaire.
- Définition de la grille d'essais coeur cylindrique.
- Montage coeur cylindrique et essais.

6. RELATIONS AVEC D'AUTRES ETUDES

- Calculs des accidents de refroidissement des réacteurs à eau : Programme général coordonné de mise au point des codes (fiche 145.1.01)
- Développement du code avancé CATHARE .... (fiche 145.1.02)
- Thermohydraulique du LOCA. Etude expérimentale de la phase renoyage des accidents de perte de réfrigérant des réacteurs à eau : Programme ERSEC ....)

7. DOCUMENTS DE REFERENCE

Qualification des crayons chauffants Périclès.
Rapport DRE/STT/SETRE 3.4.81.

145-1-10		1 - 1 - 2
TITRE THERMOHYDRAULIQUE DU LOCA : COMPORTEMENT DES GENERATEURS DE VAPEUR EN CONDITIONS ACCIDENTELLES : PROGRAMME PATRICIA,I ET II		*Pays* FRANCE
		Organisme directeur CEA/ IRDI EDF/SEPTEN-FRAMATOME
TITRE en anglais LOCA THERMOHYDRAULICS : STEAM GENERATOR BEHAVIOR IN ACCIDENT CONDITIONS : PATRICIA I AND II PROJECT		*Organisme exécuteur* CEA/DRE/STT
		Responsables M. COURTAUD-STT/SETRE
Date de démarrage 22.02.1980	*Etat actuel* en cours	*Scientifiques*
Date d'achèvement 31.12.1983	*Dernière mise à jour* 01/1983	M. ROUMY

1. OBJECTIF GENERAL

Etude des conditions thermohydrauliques et des mécanismes de transfert de chaleur dans les générateurs de vapeur en conditions accidentelles, notamment dans les situations suivantes :

- ébullition côté primaire (situation rencontrée en cas de petites brèches, perte d'eau alimentaire, ATWS, transitoires spéciaux type TMI)

- assèchement et renoyage du côté secondaire (situation rencontrée en cas de perte d'eau alimentaire, rupture de tuyauterie d'eau alimentaire ou de tuyauterie vapeur, ATWS).

2. OBJECTIFS PARTICULIERS

Déterminer le comportement des tubes de générateur de vapeur côté primaire avec à l'entrée du liquide, un mélange eau-vapeur ou de la vapeur, en ce qui concerne :

- l'établissement ou l'arrêt de la convection naturelle.
- le transfert de chaleur en condensation, en écoulement eau-vapeur contre-courant.
- le transfert de chaleur en condensation, en écoulement eau-vapeur contre-courant (situation de "reflux").
- l'influence de gaz incondensables sur les phénomènes précédents.

Déterminer les lois de transfert de chaleur et les conditions locales de l'écoulement côté secondaire dans la géométrie réelle en faisceau de tubes équipé avec des plaques entretoises en cas d'assèchement.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

Installations

Deux installations sont prévues :

PATRICIA GV 1

Ce système doit permettre l'étude de comportement d'un tube en U avec ébullition côté primaire. La section d'essais est constituée d'un tube réel de G.V. modèle 51 avec un diamètre de cintre de 160 cm. Le secondaire est constitué de 4 tronçons alimentés par du gilotherm permettant d'effectuer des bilans thermiques sur les tronçons correspondants.

La gamme prévue des paramètres explorés est :

- pression de 40 à 120 bars
- vitesse massique de 0,3 à 15 % de la vitesse massique nominale
- qualité à l'entrée comprise entre 0 et 1
- différence de température entre primaire et secondaire de l'ordre de 10° C
- injection de gaz incondensable .

PATRICIA GV 2

Ce système doit permettre l'étude du dénoyage du côté secondaire. La section d'essai est une grappe de tubes équivalente à 9 tubes avec boitier permettant de simuler la hauteur droite des tubes GV 51 et munie de plaques support. Le côté primaire est alimenté par un circuit à liquide organique (10 bars, 1 100 KW, 380°) en écoulement ascendant ou descendant.

la gamme prévue des paramètres explorés est :

- pression entre 1 et 75 bars
- débit et enthalpie permettant en permanent l'obtention de niveaux aux secondaires
- diminution du débit de transitoire à partir du débit nominal
- simulation d'injection d'eau froide alimentaire.

Programme

Le programme pour PATRICIA I et PATRICIA II comprend les étapes suivantes (pour les deux projets successivement) :

- revue de l'avant-projet
- étude de l'installation
- construction de l'installation
- établissement de la grille d'essais et calculs prévisionnels
- réalisation des essais
- interprétation des essais

Le déroulement effectif ou prévu des essais est le suivant :

9/81 à 5/82 pour PATRICIA GV1 (cela pour une partie du programme, dont certains aspects pourraient être repris ultérieurement),

1983 pour PATRICIA GV 2.

4. ETAT D'AVANCEMENT

Lancement de l'action............................ Février 80

- PATRICIA GV 1

  Etude de l'installation......................... 7 à 10/80

  Réalisation du circuit gilotherm associé à la boucle d'origine, PATRICIA .... Fin 80 - début 81

  Réalisation de la section d'essais et mise en place : achèvement .... Juillet 81

  Essais préliminaires .... 6 - 7/81

  Essais physiques .... 10 -12/81 et jusqu'à mai 82

Les essais effectués sont décrits dans des fiches; leur interprétation a débuté. cf note TT/SETRE/82-29 pour les conclusions

- PATRICIA GV 2

  Etude de l'installation et de la section d'essais début 9/80

Réalisation de la boucle et du circuit gilotherm .. } 4/82 à 3/83
Construction de la section d'essais .... }

5. PROCHAINES ETAPES

- PATRICIA GV 1

  Poursuite interprétation

- PATRICIA GV 2

  Déroulement des essais .... 4 à 12/83

- Possibilité d'un retour à PATRICIA GV 1 par la suite (début 84).

6. RELATIONS AVEC D'AUTRES ETUDES

- Calculs des accidents de refroidissement des réacteurs à eau. Programme général coordonné de mise au point des codes (fiche 145-1-01)
- Développement du code avancé CATHARE (fiche 145-1-02)

7. DOCUMENTS DE REFERENCE

" Comportement d'un générateur de vapeur en conditions accidentelles; fonctionnement en double phase côté primaire".
F. de Crécy, R.ROUMY, F. SALC, G. MAYET. Note TT/SETRE/82-29

145-1-14		1 - 1 - 2
TITRE EVALUATION - VERIFICATION PROGRAMME LOBI		*Pays* FRANCE *Organisme directeur* CEA/ IPSN - CEA/DSN
TITRE en anglais EVALUATION - VERIFICATION LOBI PROJECT		*Organisme exécuteur* CEA/DSN/SEAREL (FAR) CEA/DRE/STRE (SAC) FRAMATOME *Responsables* M. REOCREUX /SEAREL M. LERIDON /STRE M. MEGNIN FRAMATOME
Date de démarrage 1976	*Etat actuel* en cours	*Scientifiques* MME VIVIANDE/ SEAREL M. LEBERRE / STRE MME TENEGAL/FRAMATOME
Date d'achèvement 1986	*Dernière mise à jour* 01. 1983	

1. OBJECTIF GENERAL :

- Suivi et participation à la définition du programme expérimental dans le cadre d'EURATOM.
- Evaluation-vérification des codes disponibles (CATHARE, FRARELAP, RELAP 4 Mod 6, TRAC, RELAP 5 ...)
- Participation aux travaux de comparaison du groupe de travail LOBI B.
- Prise en compte des résultats de l'étude dans les calculs d'accidents des réacteurs de puissance.

2. OBJECTIFS PARTICULIERS :

2.1 - Elaboration de propositions de matrices d'essai couvrant :
- les grosses brèches
- les petites brèches
- les transitoires spéciaux

2.2 - Calculs préliminaires - confrontation des résultats avec ceux des calculs des autres participants. Interprétation.

2.3 - Calculs previsionnels et d'interprétation des essais exploratoires petites brèches avec les codes : RELAP 4 Mod 6 - TRAC - FRARELAP. Comparaisons des résultats avec ces 3 codes et avec le code allemand DRUFAN.

2.4 - Définition des modifications de l'installation pour l'étude des petites brèches et transitoires spéciaux.

2.5 - Réalisation d'essais support.

2.6 - Calculs previsionnels.

2.7 - Calculs d'interprétation des essais. Etudes de sensibilité.

2.8 - Calculs de problèmes standard LOBI.

3. INSTALLATION EXPERIMENTALE ET PROGRAMMES

Installation expérimentale

BOUCLE LOBI (ISPRA - EURATOM)

- Simulation d'un PWR
- 2 boucles : 1 boucle rompue
  1 boucle non rompue représentant 3 boucles du réacteur.
- Composants actifs sur les deux boucles (G.V.- pompes)
- Puissance 5 MW (chauffage électrique direct programmable en temps) pression : 150 bars.
- Dans le but d'étudier les petites brèches et transitoires, l'installation sur la boucle de 2 nouveaux générateurs de vapeur ainsi que d'une autre cuve de réacteur devait s'effectuer pendant l'été 1982. L'adaptation d'un dispositif d'instrumentation complémentaire était également prévue. Cependant, en raison de problèmes techniques relatifs aux thermocouples, la livraison des générateurs de vapeur est annoncée pour fin Avril 1983 seulement, et l'installation complète de la boucle ne sera achevée que vers Juin 1983.

Programmes

Programme A1 : Programme allemand .... 30 essais ~

Programme A2 : Programme décidé par les allemands
résultats disponibles pour la communauté ... 30 essais ~

Programme B : Programme communautaire .... 30 essais ~

Programme transitoires spéciaux : Programme communautaire

Les deux programmes A et B comprennent des essais simulant à la fois des grosses et petites brèches.

Le programme A1 a démarré effectivement fin 1979, début 1980.
Il a été interrompu pour une période intérim de 6 mois allant d'Octobre 1980 à Avril 1981, en raison de problèmes techniques relatifs au "downcomer" réduit dont l'installation a donc dû être différée. Cette période intérim a permis la réalisation de 5 essais (3 essais du programme A2 et 2 essais du programme B). On a pu procéder à l'installation du "downcomer" réduit en Mai 1981 et le programme A1 a pu reprendre avec un rythme de l'ordre de 12 essais par an.

En 1982, deux essais B grosses brèches sont venus s'intercaler dans ce programme A1.

Le programme d'essais a été ensuite interrompu en Juillet 1982 pour l' installation de deux nouveaux générateurs de vapeur et de la nouvelle cuve, et ne pourra reprendre que vers mi-83. il est alors prévu d'effectuer 2 à 3 essais exploratoires, suivis d'un essai de vérification. Le programme A dont 2 essais seront disponibles pour la communauté en 1983, se poursuivra par 2 essais en circulation naturelle et une douzaine d'essais petites brèches. Il est envisagé d'intercaler plusieurs essais B chaque année, dans le programme A, selon un nombre qui reste à définir.

4. ETAT D'AVANCEMENT

Avancement à ce jour

Objectif 2.1

La dernière proposition française pour la matrice d'essais du programme B a été présentée en Septembre 79. Cette matrice reprenait la proposition de Novembre 78 en y ajoutant certains essais de la proposition de Mars 77 et en y apportant certaines inflexions résultant de l'accident de T.M.I.

Afin d'avoir une idée plus précise du comportement de l'installation dans les études de petites brèches, il a été décidé de faire 3 essais exploratoires qui ont eu lieu mi 1980.

On a examiné l'harmonisation des matrices A (programme allemand) et B (programme européen) afin d'alléger le programme d'essais.

L'harmonisation des programmes a été retenue comme principal objectif, lors du choix des essais du programme de la période intérim.

La configuration de l'espace annulaire réduit prévue pour plusieurs mois, a rendu nécessaire la définition de nouveaux essais B avec cette nouvelle configuration. Parmi ceux-ci, deux essais grosses-brèches, avec localisation respectivement sur la branche froide et sur la branche chaude, ont été choisis par la communauté européenne, pour être les deux essais B à intercaler dans le programme A1 en 1982.

Deux autres essais petites-brèches ont également été sélectionnés pour être les deux essais A2 qui seront réalisés en 1983, sur l'installation modifiée LOBI-mod 2.

Une proposition de spécifications générales d'essais de transitoires spéciaux a été élaborée par la France et présentée en Septembre 1979.

Chaque pays membre a examiné l'état de ses connaissances sur les transitoires spéciaux, et a étudié les essais intéressants qui pourraient être réalisés sur la boucle LOBI, compte-tenu de la nouvelle instrumentation mise en place pour ces études.

Une liste d'essais a été établie, regroupant l'ensemble des transitoires et des phénomènes physiques que les pays membres souhaitaient étudier sur LOBI. Une distinction a été faite entre deux types de transitoires, ceux à court terme et ceux à long terme.

Dans cette liste, trois essais ont été choisis parmi lesquels deux au moins devraient être effectués en tant qu'essais exploratoires transitoires en 84. Il s'agit d'un ATWS de perte des alimentations électriques et d'un accident de perte d'eau alimentaire.

Objectif 2.2

Courant 1978 et 1979 ont été réalisés des calculs préliminaires (utilisation du code de calcul RELAP 4 Mod 5, puis Mod 6) concernant des essais du type grosse brèche en prenant comme base la matrice d'essais définie par le groupe des pays européens.

Fin Avril 1980,un calcul du type aveugle ("blind problem") a été achevé qui consiste à calculer un essai déjà réalisé mais dont les résultats d'expérience ne sont pas communiqués. Ce calcul (calcul LOBI-PREX) a été exécuté avec le code de calcul RELAP 4 Mod 6.

Objectif 2.3

- Préparation du jeu de données correspondant aux essais exploratoires petites brèches.
- Réalisation de 2 calculs prévisionnels du 1er essai exploratoire petite brèche avec respectivement les codes :

  RELAP 4 Mod 6 (calcul effectué par la SEAREL)

  FRARELAP (calcul effectué par le DRE/STRE)
- Réalisation d'un calcul prévisionnel du second essai exploratoire petite brèche avec le code de calcul FRARELAP.
  En 83 un premier calcul d'interprétation du SDSL 03 a été réalisé. (FRARELAP)

Objectif 2.4

- Une adaptation d'un dispositif d'instrumentation complémentaire a été étudié sur la boucle LOBI, afin d'obtenir une meilleure modélisation des petites brèches. Cette étude est actuellement en voie d'achèvement. La conception d'un nouveau générateur de vapeur, ainsi que d'une nouvelle cuve a été décidée par l'ensemble des pays membres.
- Ces nouveaux composants en sont au stade de la fabrication. Les problèmes techniques qui se sont posés, pour la tenue sous pression des soudures des thermocouples sur les tubes en U des générateurs de vapeur, sont en cours de résolution.
- La liaison entre la branche principale et l'orifice brèche sera assurée par une jonction en T dont la configuration est actuellement en cours d'étude.

Objectif 2.6

En 1982 les calculs prévisionnels de l'essai B 222 ont été réalisés

Objectif 2.7

En 1982 des calculs d'interprétation des essais B 101 et A1 66 ont été faits.

Principaux résultats

Objectif 2.2

- Les calculs préliminaires effectués par les différents pays européens sur la matrice d'essai grosse brèche ont fait l'objet d'un rapport de synthèse. Les conclusions de ce rapport (notamment en ce qui concerne l'influence de l'épaisseur du "downcomer") ont été prises en compte dans la définition de la matrice finale d'essais grosse brèche du programme B.

- Le travail de confrontation des résultats des calculs - résultats d'expérience, pour l'exercice LOBI-PREX a permis de montrer l'importance de la modélisation adoptée pour obtenir une bonne description de certains phénomènes physiques. En particulier, il apparaît nécessaire de prendre en compte, même dans les transitoires grosses brèches, les phénomènes de glissement et de séparation de phase. Pour ces derniers, l'aptitude des modèles les décrivant dans les codes a été mise en cause,ainsi que celle:

  . des modèles de débit critique qui n'ont pas permis de prédire une bonne loi de pression, déterminante dans le comportement de la boucle durant l'ensemble du transitoire.

  . du modèle d'injection accumulateur qui n'évalue pas bien le débit injecté.
  Un rapport de comparaison de ce problème standard a été établi. En ce qui concerne le calcul français, mis à part les débits de sortie de la cuve, qui dans certaines portions du transitoire sont assez différentes des mesures, les options choisies ont conduit à une concordance très convenable entre les résultats de calcul et les résultats d'essais, se comparant favorablement par rapport à de nombreux autres participants

Objectif 2.3

- Un calcul d'interprétation du troisième essai exploratoire petite-brèche a été effectué avec le code FRARELAP. La confrontation des résultats d'essai et de calcul montre que :

  . Le débit brèche devrait être fortement augmenté pour que la première phase de dépressurisation soit correctement prédite.

  . Les densités, en branche chaude comme en branche froide ne sont pas prédites de façon satisfaisante.
  Les résultats d'essais prouvent en effet que les branches chaudes restent pratiquement pleines de liquide pendant tout le transitoire, alors que le haut du downcomer reçoit rapidement de la vapeur.
  En revanche le calcul fait apparaître de la double-phase en premier lieu dans la branche chaude et le plenum-supérieur.
  Des études de sensibilité ont permis notamment, de mettre en évidence l'importance de la position du by-pass coeur.

Objectif 2.6

- Des calculs prévisionnels ont été effectués par différents pays membres, sur les essais grosses brèches du programme de la période intérim. La France s'est chargée de la prédiction des essais B.101 et B.101-M. Ces calculs ont permis de déterminer les spécifications de puissance et de vitesses de pompes à fournir à Ispra pour préparer et réaliser l'essai.

  Une méthode a été établie pour la détermination de la puissance électrique devant être fournie par les crayons chauffants LOBI. Cette méthode consiste essentiellement à déduire la puissance d'un calcul avec coeur nucléaire fictif, permettant alors de simuler sur le coeur électrique à inertie thermique réduite, des conditions thermiques réalistes correspondant au destockage d'énergie des pastilles dans le combustible des réacteurs de puissance.

  Les premiers résultats expérimentaux comparés aux résultats de calculs montrent des différences entre les conditions spécifiées et les conditions réelles des essais.

- Un calcul de prédiction a été réalisé sur l'essai grosse-brèche B-222, avec la configuration de l'espace annulaire réduit. La détermination des spécifications de puissance et de vitesses de pompes s'est effectuée suivant la méthode décrite ci-dessus.

  La comparaison des résultats de calcul avec les résultats expérimentaux n'a pas encore été entreprise.

Objectif 2.7

- Un calcul d'interprétation de l'essai B-101 a été réalisé, en prenant en compte les conditions initiales et aux limites réelles de l'essai.

  La confrontation des résultats des calculs avec les résultats d'essai a montré que :

  . les corrections effectuées sur les conditions de l'essai n'amélioraient pas les résultats de calcul.

  . l'établissement d'une nouvelle distribution de pression dans le circuit calquée sur les mesures de ΔP, n'entraînait pas de différence sensible dans les résultats.

  . la vitesse de dépressurisation trop élevée dans le calcul a été partiellement réduite par l'utilisation d'un coefficient de contraction à la brèche. Cela a permis d'améliorer la description par le calcul du comportement thermohydraulique général de la boucle.

  Néanmoins la comparaison des résultats met en évidence toujours des écarts importants sur certains paramètres qu'il n'est pas facile d'interpréter en raison du manque de mesures notamment pour les débits, sens de circulation du fluide et pour les conditions initiales précises du pressuriseur ou de l'accumulateur.
  L'aptitude de certains modèles du code est également à mettre en cause, notamment en ce qui concerne la séparation de phase dans les branches horizontales et la description des effets tridimensionnels qui sont très

importants dans le plenum-supérieur et le haut de l'espace annulaire. Les mêmes conclusions ont été tirées de l'interprétation des résultats des autres essais de la période intérim au sein du groupe de travail ad hoc.

- Les résultats du calcul réalisé sur le premier essai (A1 - 66) avec configuration de l'espace annulaire réduit ont été comparés avec les résultats expérimentaux et ont permis d'effectuer l'adaptation de notre jeu de données à la nouvelle géométrie de la cuve.

+ effet du coefficient de contraction à la brèche

La comparaison entre les résultats du calcul et de l'essai a mis en évidence la grande sensibilité des résultats de calcul, notamment la loi de pression et l'évolution de température de gaine au coefficient de contraction à la brèche. Une valeur moyenne de 0.85 pour ce coefficient constitue un compromis pour obtenir un accord convenable sur l'ensemble des résultats, et a donc été adoptée provisoirement pour réaliser l'étude de sensibilité sur l'effet des pompes.

+ effet des pompes

La comparaison de 2 calculs portant sur les deux essais :

- A1-66 dont le fonctionnement du rotor de la pompe de la boucle rompue est " bloqué".
- B-221 qui présente les mêmes conditions que l'essai A1-66 sauf le fonctionnement du rotor de la même pompe qui est "libre".

a permis de voir que l'effet du fonctionnement pompe, dans le cas de la configuration avec espace annulaire réduit, semble, d'après les calculs effectués, aussi peu sensible qu'avec le grand espace annulaire.

Ce résultat a servi à la définition des essais B avec l'espace annulaire réduit.

5. PROCHAINES ETAPES

Objectif 2.1

- Définition des futurs essais communautaires petites-brèches pour 1984.
- Révision des matrices B, à la suite de l'interprétation des résultats expérimentaux et de calculs obtenus sur les essais réalisés.
- Définition des conditions initiales et aux limites des 2 essais exploratoires transitoires spéciaux pour 1984.
- Définition détaillée de la matrice d'essais des transitoires spéciaux.

OBJECTIF 2.3

- Réalisation d'études de sensibilité suite au calcul post-essai, pour le troisième essai exploratoire petite brèche SDSL 03 avec le code FRARELAP. (Calcul effectué par STRE dans le cadre de la fiche de collaboration FRA-DSN sur les codes petites brèches).
- Confrontation des résultats obtenus par les différents pays s'intéressant

à ces essais, et prise en compte des résultats de l'analyse dans la réalisation future des études.

Objectif 2.4

- Suivi du redémarrage du programme d'essais suite à la fabrication des 2 nouveaux générateurs de vapeur, de la cuve de réacteur et de leur installation sur la boucle LOBI.

Objectif 2.5

- Définition de l'orientation et de la géométrie des orifices petites brèches.
- Réalisation d'essais d'étalonnage des dispositifs de brèche sur la boucle SUPER MOBY DICK (travail effectué par STT).

Objectif 2.6

- Réalisation des calculs prévisionnels des essais de la matrice A1 qui sont disponibles à la communauté.

  Ces calculs seront effectués pour les essais petite brèche avec le code FRARELAP (fiche FRA-DSN) et avec éventuellement le code TRAC.
- Réalisation de calculs de prédictions et définition des spécifications des futurs essais A2 et B réalisés sur la boucle LOBI en 1983-84.

Objectif 2.7

- Calculs d'interprétation des essais B grosses-brèches avec l'espace annulaire réduit.
- Calculs d'interprétation des futurs essais petites-brèches et transitoires réalisés sur LOBI-Mod 2 et dont les résultats seront disponibles à la communauté.

Objectif 2.8

- Réalisation d'un calcul du type aveugle ("double-blind problem") sur un des premiers essais petites-brèches réalisés sur l'installation modifiée LOBI-Mod 2, dans le cadre du problème standard international N° 18. Il s'agira de calculer un essai déjà réalisé mais dont les résultats d'expérience ne seront pas communiqués, de même que les résultats des autres essais réalisés sur LOBI-Mod 2. Ce problème sera donc aveugle à double titre, du point de vue de l'essai et du point de vue de l'installation modifiée dont les participants ne connaîtront pas les réponses.

6. RELATIONS AVEC D'AUTRES ETUDES

Cette étude est une des facettes de l'étude générale :

- Calculs des accidents de refroidissement des réacteurs à eau.
  Programme général coordonné de mise au point de codes (fiche 145-1-01)

Elle est la suite logique des études :

- Qualification-validation RELAP-TRAC .... (fiches 145-1-11

Elle est le complément de l'étude :

- Evaluation-vérification RELAP-TRAC
  Calculs SEMISCALE - LOFT - PKL........................ (fiche 145-1-13)

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A. SONNET
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Pre-test predictions
Specification of power control curve
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Comparison of the predictions with RELAP 4 Mod 6 of the LOBI interim tests B-101 and B-101 M using the preliminary two phase characteristics of the LOBI pumps.
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F. LE BERRE
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Calculs FRARELAP sur LOBI
Brèche de 10 % et 1 %
Note technique Framatome : EP.TT.DC.0005

MA VIVIANDE - R. POCHARD
Rapport d'analyse des calculs de prédictions des essais LOBI B-101 et B-101 M avec le code RELAP 4 Mod 6
Note technique SEAREL N° 80/83

MA VIVIANDE - R. POCHARD
Post-test Analysis on LOBI test B-101 with RELAP 4 Mod 6 code
Note technique (à paraître)

<table>
<tr><td colspan="2">145-2-05</td><td>1 - 1 - 2</td></tr>
<tr><td colspan="2" rowspan="2">TITRE

MODELISATION DE LA THERMOHYDRAULIQUE DANS UN
COEUR DEGRADE</td><td>Pays FRANCE</td></tr>
<tr><td>Organisme directeur
CEA/IPSN</td></tr>
<tr><td colspan="2" rowspan="2">TITRE en anglais

MODELING OF THE THERMAL-HYDRAULICS IN A
DEGRADED CORE</td><td>Organisme éxécuteur
CEA/STT GRENOBLE</td></tr>
<tr><td>Responsables
M. NIGON STT</td></tr>
<tr><td>Date de démarrage 1981</td><td>Etat actuel En cours</td><td rowspan="2">Scientifiques
M. DUCO DSN
M. GERBAUX STT
M. GRANDJEAN STT</td></tr>
<tr><td>Date d'achèvement 1985</td><td>Dernière mise à jour Mars 1983</td></tr>
</table>

1) OBJECTIF GENERAL

Définition de modèles, mise au point et qualification de moyens de calcul de thermohydraulique dans un coeur avec gaines fortement déformées et dans des zones de débris d'un coeur sévèrement dégradé.

2) OBJECTIFS PARTICULIERS

a) Coeur avec gaines fortement déformées

Etablissement, mise au point et qualification des moyens de calcul de thermohydraulique dans un coeur avec barreaux gonflés, gaines fortement déformées ou éclatées.

Etude du refroidissement global et local du coeur dans ces conditions en vue de définir les conditions thermiques sur le combustible et les conditions limites sur une zone fortement dégradée.

b) Coeur avec zone de débris

- Définir les modèles utilisables pour traiter un écoulement avec transferts thermiques dans des zones de débris à l'intérieur d'un coeur dégradé (milieux poreux).

- Recherche de lois physiques décrivant les échanges fluide-solide.

- Réalisation d'outils de calcul utilisant ces modèles et lois physiques et qualification.

- Définir les limites en deça desquelles le refroidissement est certainement possible, en fonction de paramètres globaux aussi simples que possible (volume de la zone dégradée, porosité, puissance résiduelle, température...).
  Cette dernière phase a pour objectif la définition de situations enveloppes indépendamment des scénarios qui pourraient y conduire.

- Prise en main et amélioration physique et informatique d'un outil existant (BOIL K) traitant de façon simple le découvrement et la dégradation du coeur pour des températures de gaine dépassant 1200°C.
  La version améliorée de cet outil devrait permettre, début 1984, de préciser certains points des calculs de séquences accidentelles prototypes de PWR, utilisées pour la définition des spécifications d'essais PHEBUS phase III.

PLANNING

a) Coeur avec gaines fortement déformées

Choix des moyens de calcul adaptés au coeur avec combustible déformé	(1982 1983 - 1er semestre)
Prise en main des outils, adaptation, calcul d'expériences existantes	(1983)
Calcul de situations accidentelles sur l'ensemble du coeur	(1984)
Application au coeur de TMI ; comparaison avec les constatations faites dans le réacteur	(1984 - 1985)
Application à d'autres expériences (PHEBUS, PERICLES)	(1984 - 1985)

b) Coeur avec zone de débris

Recherche des modèles utilisables pour le calcul d'un écoulement diphasique avec transfert de chaleur dans un milieu poreux	1981 - 1982
Comparaison avec des expériences existantes	1983
Etude paramétrique des conditions limites de refroidissement	1983 - 1984
Comparaison avec la situation du coeur constatée dans TMI	1984
Application à PHEBUS	1984 - 1985

3) INSTALLATIONS EXPERIMENTALES

Sans objet.

4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

a) Coeur avec gaines fortement déformées

Une étude bibliographique sur les expériences de thermohydraulique étrangères a été réalisée.

Examen de la faisabilité d'un calcul de coeur partiellement bouché (configuration de l'expérience allemande FEBA) à l'aide du module 3D VESSEL du code TRAC-PF1.

b) Coeur avec zone de débris

Recherche bibliographique sur la phénoménologie des phases successives d'un accident sévère, et sur l'état des connaissances actuelles et des programmes en cours ou en projet.

Définition de l'outil de calcul milieu poreux 1D.

Réalisation informatique de l'outil de calcul.
Recherche de lois physiques d'échange fluide/particules.
Premières qualifications.

5) PROCHAINES ETAPES

a) Coeur avec gaines fortement déformées

Prise en main des outils existants

COBRA - TRAC ...

et s'ils peuvent être disponibles :

COBRA-TF, MABEL 2

Qualification sur expériences existantes (RFA, UK, USA).

Adaptation pour le couplage avec les modèles combustible développés en France.

Couplage éventuel avec un module traitant la dégradation avancée d'une zone après apparition des premiers débris.

b) Coeur avec zone de débris

Prise en main complète du code BOII K expertise - Améliorations physiques et informatiques.

Achèvement de l'outil de calcul 1D et poursuite de l'extension 3D pour le calcul de la thermohydraulique double phase dans un milieu poreux.

Choix des corrélations et qualification sur résultats existants.

6) RELATIONS AVEC D'AUTRES ETUDES

Fiche étude 145-2-01. Développement de modèles et de codes de sûreté sur le comportement du combustible au cours d'accidents (APRP).

Fiche étude 170-2-01. Préparation du programme PHEBUS CSD (coeur sévèrement dégradé).

7) DOCUMENTS DE REFERENCE

Note DRE/STT/SEMTH/82-1. Accident sévère et coeur dégradé dans les réacteurs à eau légère. Position des problèmes. Contexte et programmes d'études.

8) DEGRE DE DISPONIBILITE

Documents internes non disponibles, sauf accord du CEA.

	CLASSIFICATION : 4 / 6 / 1.1.2
TITLE (ORIGINAL LANGUAGE) : TRANSITORI TERMOIDRAULICI DI BWR PER MALFUNZIONAMENTI DEL SISTEMA DICONTROLLO	COUNTRY : ITALY SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : THERMOHYDRAULIC TRANSIENTS OF BWR'S FOR SYSTEM CONTROL FAULTS	ORGANISATION : CISE PROJECT LEADER : G. BARZONI
INITIATED : 1980 COMPLETED : 1982 STATUS : IN PROGRESS LAST UPDATING : APRIL 1982	SCIENTISTS : C. MEDICH C. SANDRI

	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: ENEA 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, C. CECCARELLI, G. BITETTI

	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: ENEA 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	

	CLASSIFICATION : 1.1.2 , 7.1
TITLE (ORIGINAL LANGUAGE) : Ricerca teorica e sperimentale sull'incidente di perdita di refrigerante nei reattori nucleari. Forze di getto prodotte da efflusso critico di miscele acqua-vapore	COUNTRY : Italy
	SPONSOR : C.N.R.
TITLE (ENGLISH LANGUAGE) : LOCA Analysis in LWR: Blowdown Jet Forces	ORGANISATION : University of Pisa
	PROJECT LEADER : Piero VIGNI

INITIATED : 1978	COMPLETED : 1982	SCIENTISTS : U. ROSA, F. D'AURIA
STATUS : in progress	LAST UPDATING : May 1982	

1. General aim
   To investigate the blow-down jet behaviour and impingement forces on supporting structures of pressure vessel and on external targets.

2. Particular objectives
   To evaluate the effects of the most important parameters connected with the thermal-hydraulic phenomena following LOCAs in LWR, with particular reference to the technical effects and scaling laws evaluation.

3. Experimental facilities
   The experimental apparatus includes a pressure vessel 3 m high and a plane target, far from it 5 ÷ 10 blow-down nozzle diameter. The vessel is equipped with an electrical heating device, rupture disk assembly and instrumentation for pressure, temperature and liquid level measurements. The dinamic loads in the vessel supporting structure and on the plane target are measured by piezoelectic transducers. The jet profile can be also filmed at high velocity. The main design features of the vessel are:

- pressure	100 kg/cm^2
- temperature	310 °C
- volume	90 l
- blow-down nozzle	two equal, at different height, with an internal diameter of 50 mm and a length of 400 mm.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
LOCA Analysis in LWR: Blow-down Jet Forces	1.1.2, 7.1

4. Project status

Some tests at different initial conditions (pressure, liquid level and quality) have been performed using a full area break (50 mm diameter) under the initial water level. The dynamic loads on supporting vessel structures were measured; the pressure and temperature histories at different stations in the outflow nozzle were also investigated.
An analytical procedure has been developed to obtain all blowdown related quantities in the pressure vessel and at the exit of the broken pipe from pressure and jet reaction thrust measurements.

5. Next steps

Further experiments are in progress to investigate the values of impingement forces on external target, pressure and temperature histories at outflow section as a function of break area and position, liquid level height, initial pressure and quality.

6. Relation to other project and codes

SOPRE 2 research on the behaviour of MARK III pressure suppression containment system after LOCA - Project Leader: M. MAZZINI
Analysis of thermo-hydraulic transients in LWR following LOCAs
Project Leader: M. MAZZINI.

Reference documents

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

2. P.VIGNI, F. D'AURIA :"Forze di reazione prodotte da getti di fluido bifase in condizioni di efflusso non stazionario".
Rivista di Ingegneria Nucleare N. 5 - 1979

3. P. VIGNI, F. D'AURIA :"Problemi relativi alla valutazione delle forze di reazione durante un incidente di perdita di refrigerante in un impianto nucleare".
Atti Istituto di Impianti Nucleari, RP 359(79).

4. F. D'AURIA, P. VIGNI :"Two-phase Critical Flow Models"
OECD-CSNI - Report n.47 - Paris, May 1980

5. F. D'AURIA, P. VIGNI :"Blowdown two-phase flowrate evaluation method from pressure and thrust measurements"
Atti Istituto di Impianti Nucleari, RP 437(80).

6. F. D'AURIA, P.VIGNI :" Fluid dynamic Analysis of steam-water flow from a pressure vessel"
Proceeding of Workshop on Jet impingement and pipe whip, sponsored by CNEN and Ansaldo AMN. Genova, June 29 - July 1, 1981

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
LOCA Analysis in LWR: Blow-down Jet Forces.	1.1.2., 7.1

7. F. D'AURIA, P. VIGNI :" Blowaes: a Computer Program for evaluating blow-down two-phase flowrate from pressure and thrust measuremets".
In preparation.

8. Degree of availability
The references 1 ÷ 5 are free; the reference 6 is restricted.
Please contact:
Piero VIGNI
Dipartimento di Costruzioni
Meccaniche e Nucleari
Università di Pisa
Via Diotisalvi, 2
56100 PISA -

9. Additional information
An expense of about 20 millions of Lire is foreseen to complete the research, without any consideration of salaries. Man-power is equivalent to one researcher and one technical operator.

		CLASSIFICATION : 1.1.2, 1.1.4., 1.2
TITLE (ORIGINAL LANGUAGE) : Analisi dei transitori termo-idraulici nel circuito primario dei reattori ad acqua leggera a seguito di LOCA.		COUNTRY : ITALY
		SPONSOR : ENEA (formerly CNEN)
TITLE (ENGLISH LANGUAGE) : Analysis of thermo-hydraulic transients in LWR following LOCA		ORGANISATION : University of Pisa
		PROJECT LEADER : Marino MAZZINI
INITIATED : 1979	COMPLETED : 1987	SCIENTISTS : N.CERULLO F.ORIOLO, U.ROSA, P.VIGNI, F.D'AURIA, R.BOVALINI, G.GALASSI, M.CARCASSI.
STATUS : in progress	LAST UPDATING : January 1982	

1. General aim

The research has the purpose of setting-up a best estimate model for the analysis of thermal and hydraulic transients following LOCA in light water reactors, achieving a better knowledge of major phenomena occurring during blowdown, refill and reflood phases.

Particular objectives

Qualification of "Best Estimate" computer codes in the analysis and design of experimental tests, by their applications to pre-test calculations and post-test analysis of experimental results obtained in the experimental facilities of Scalbatraio Center (University of Pisa) or available from international research programs (LOBI-CCR-Ispra; LOFT and Semiscale, with reference to CSNI International Standard Problems).

2. Facilities

The experimental small scale facilities PIPER and SOPRE of the Scalbatraio Center, University of Pisa.
A new experimental facility, named PIPER-ONE is in design stage; it simulates the thermal-hydraulic behaviour of BWR plants during small LOCA.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Analysis of thermo-hydraulic transients in LWR following LOCA	1.1.2, 1.1.4, 1.2

4. Project status

In 1979 the research program for 80's was outlined on the basis of the Italian situation and of the extensive work previously carried out at the University of Pisa for the study of blowdown, theoretically and experimentally.
According to these research lines, the following work has been performed in the years 1980 and 1981:

- participation to the preparation of a State of Art Report on two-phase critical flow in OECD-NEA frame;
- application of the RELAP4 Mod. 5 and/or 6 computer codes to the international Standard Problems n. 8, 9 and 11 proposed by NEA-CSNI and related to LOFT and Semiscale research programs and to the tests A1-04, A2-55 and SDSL-03 performed at LOBI test facility;
- feasibility study and thermal-hydraulic design of a small scale (about 1/3000) apparatus named PIPER-ONE, which should simulate thermal-hydraulic transients in BWR following small LOCAs.

5. Next Steps

Partecipation to LOBI program: in particular analysis of other experimental tests (A1-66, B-302, etc).
Realization of PIPER-ONE apparatus and execution of the first tests.
Eventually modification of RELAP4 code to improve its capability for analyzing small LOCA and reflood transients.

6. Relation to other projects

- LOCA analysis in LWR: blowdown forces. Project Leader: P. VIGNI.
- SOPRE-2: Research on the behaviour of MARK III pressure suppression systems after LOCA. Project Leader: M. MAZZINI
- Analysis of thermo-hydraulic transients in LWR containment systems following LOCAs. Project Leader: M. MAZZINI

7. Reference documents

1. 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,Bruxelles (Belgium) October 16-19, 1978.
2. F. D'AURIA, P. VIGNI :"Two-phase Critical Flow Models. State of the Art Report". OECD-CSNI Report n. 47. Paris, May 1980

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Analysis of thermo-hydraulic transients in LWR following LOCA	1.1.2, 1.1.4, 1.2

3. N. CERULLO et alii :"Sull'impiego del programma RELAP4 per lo studio del transitorio termoidraulico conseguente ad un incidente di perdita di refrigerante in un LWR".
Atti XXXV Congresso Associazione Termoelettrica Italiana. Saint Vincent, Settembre 1980.

4. M. MAZZINI et alii :"PIPER-ONE: An experimental apparatus to Evaluate Thermal-hydraulic Transients in BWRs after Small Breaks' ANS Specialists Meeting on "Small Breaks LOCA Analysis". Monterey (CA), August 1981.

5. G. GALASSI et alii : "The Analysis of L3-6/L8-1 LOFT Experiments"

8. Degree of-avalilability

All the references are free. Please contact M. MAZZINI - Dipartimento di Costruzioni Meccaniche e Nucleari - Facoltà di Ingegneria, Via Diotisalvi, 2 - 56100 PISA, ITALY .

9. Additional information

For one year the budget will be about 200 Millions of Lire. The realization of PIPER-ONE apparatus will add to this amount about 500 Millions of Lire. Man-power is completed by 4 technical operators.

	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' di PALERMO (+)
TITLE (ENGLISH LANGUAGE) : Studies on two-phase critical flow in connection with LOCA in light water reactors	ORGANISATION : Università di Palermo (+) PROJECT LEADER : ELIO OLIVERI

INITIATED : 1976	COMPLETED :	SCIENTISTS : P. CASTIGLIA, G. VELLA
STATUS : In progress	LAST UPDATING : May 1982	

(+) ISTITUTO DI APPLICAZIONI E IMPIANTI NUCLEARI

DESCRIPTION: The program has been set up with the aim of developing a theoretical model for the prediction of steam/water critical pressure and critical flow rate in terms of up-stream 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) .

At present, studies are running in order to consistently incorporate non equilibrium and entrainement phenomena in the model.

REFERENCE DOCUMENTS:

[1] - P. CASTIGLIA-E. OLIVERI-G. VELLA: '' Sulla determinazione della portata nell'efflusso critico bifase ''
Atti dell'Acc. di Scienze, Lettere ed Arti di Palermo - Serie IV - Vol. XXXV - 1975/76 - Parte I -

[2] - P. CASTIGLIA-E. OLIVERI-G. VELLA: '' Un programma in BASIC per il calcolo delle proprieta' termodinamiche dell'acqua ''
Atti dell'Acc. di Scienze, Lettere e Arti di Palermo - Serie IV - Vol. XXXVIII - 1978/79 - Parte I -

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Studies on two-phase critical flow in connection with LOCA in light water reactors	1.1.2.

[3] - F. CASTIGLIA-E. OLIVERI-G. VELLA: '' Sull'efflusso critico di miscele bifasi monocomponenti ''
Energia nucleare - Vol. 26 - n. 4 - Aprile 1979

PERSONNEL INVOLVED: 3 Researchers

UNIVERSITA' - Istituto di Applicazioni e Impianti nucleari
Parco d'Orléans , I-90128 PALERMO

	CLASSIFICATION : 1.1.2/1.2/1.3/7.2
TITLE (ORIGINAL LANGUAGE) : ATTIVITA' DI RICERCA FINALIZZATA NEL CAMPO DELLA TERMOIDRAULICA DEI TRANSITORI CONSEGUENTI A LOCA PER IL REATTORE PWR ENEL UNIFICATO	COUNTRY : ITALY SPONSOR : ENEL
TITLE (ENGLISH LANGUAGE) : FYNALIZED RESEARCH ACTIVITY IN THE FIELD OF THERMO-HYDRAULIC TRANSIENTS FOLLOWING LOCAS FOR THE ENEL UNIFIED PROJECT PWR REACTOR	ORGANISATION : ENEL PROJECT LEADER : G. TREBBI
INITIATED : 1980 / COMPLETED :	SCIENTISTS : L. BELLA, F. DONATINI
STATUS : IN PROGRESS / LAST UPDATING : May 1982	

1. General aim

The research has the purpose to develop an extensive know-how about the thermal-hydraulic aspects of LOCA'S transients for the ENEL reference PWR plant. The following aspects will be investigated:

a) Primary system thermal-hydraulic behaviour
b) Fuel assembly thermal behaviour (refill - reflood phase)
c) Containment behaviour/ECCS performance.

In the next years such three aspects could be splitted up in three distinct research activities.

2. Particular objectives

Qualification of RELAP 4 codes chain for LOCA'S analyses applied to PWR ENEL reactor (EM/Best estimate calculations), extended to the refill-reflood transients. The analyses will cover a great range of breaks (large DE-small) and ECCS performance, under various hypotheses. Comparisons with experimental results will be made with OECD/CSNI Standard Problems program.

The containment will be analized with CONTEMPT LT and COMPARE MOD1 codes and the effect of various emergency systems will be investigated.

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TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Thermo-hydraulics Transients following LOCAs for the ENEL Unified Project PWR Reactor	1.1.2/1.2/1.3/7.2

3. Facilities

Numerical analyses and comparisons with LOFT and SEMISCALE facilities results. Use of a CISE facility (in Piacenza).

4. Activity status

A best estimate analysis of a large LOCA on (SP-8-S-06-3) test on the SEMISCALE facility was performed with RELAP IV/M6 code in blowdown, refill and reflood phases. Evaluation model and best estimate calculations were carried out for a large LOCA in PWR ENEL reference plant during the blow-down phase, using RELAP IV/M5 code.

RELAP IV mod 5 and mod 6, TRAC-P1A, FRAP-T3, CONTEMPT-LT26, COMPARE mod 0 and mod 1 codes and ENELPLT plotting program were implemented.

An evaluation of the performance of all containment codes available (including RELAP IV mod 5) was made versus the experimental data of CASP 1 test.

A number of long term containment analyses were carried out under various containment system configurations for PWR Enel reference plant.

5. Next steps

-Blind large LOCA analysis of SP 13 test on LOFT facility. PWR Enel reference plant analyses for large LOCA in refill and reflood phases and hot channel analysis. Alternate ECCS analyses. Comparisons between numerical analyses and experimental results obtained on the CISE loop for the reflood phase.

6. Relation to other projects

OECD CSNI Standards programs (LOFT, Semiscale facilities).

7. Additional information

Personnel involved: 3 researchers, 6 technicians.

G. TREBBI, ENEL, Centro Ricerca Termica e Nucleare, V. C. Battisti 69, 56100 PISA.

	CLASSIFICATION : 1.1.2.
TITLE (ORIGINAL LANGUAGE) : COMPORTAMENTO TERMOIDRAULICO DI UN PRESSURIZZATORE IN CONDIZIONI INCIDENTALI	COUNTRY : ITALY SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : THERMOHYDRAULIC BEHAVIOUR OF A PRESSURIZER DURING ACCIDENT CONDITIONS	ORGANISATION : CISE PROJECT LEADER : R. MARTINI

INITIATED : 1980	COMPLETED : 1983	SCIENTISTS : L. AUGELLO C. MEDICH
STATUS : IN PROGRESS	LAST UPDATING : APRIL 1982	

1. General aim: to determine the water level transient of a pressurizer during relief valve discharge.

2. Particular objectives: i) to determine the behaviour of steam and water in counter-current flow along the surge-line ii) to study single and two-phase critical flow in the safety relief valve under dynamic conditions.

3. Experimental facilities and programme: CIRCE and IETI-1 loops in EMILIA Power Station, Piacenza, Italy. Tests with different surge-line geometries in reduced scale; parametrization of pressure and level in the pressurizer. Measurement of critical flow discharge in reduced scale under steady-state and dynamic conditions, with parametrization of upstream pressure, quality and valve opening.

4. Project status

   4.1 Progress to date: experimental measurements of counter current flow limitation (flooding) completed. Design of experimental arrangement for Safety Relief valve (SRV) investigation initiated.

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TITLE (ENGLISH LANGUAGE): THERMOHYDRAULIC BEHAVIOUR OF A PRESSURIZER DURING ACCIDENT CONDITIONS	CLASSIFICATION: 1.1.2.

4.2 Essential results: curves of maximum liquid flowrate vs. steam flowrate obtained under different conditions as well as pressure drops across the surge-line.

5. Next steps: comparisons of obtained experimental data with correlations; detailed design of SRV test facility and procedure.

6. Relation to other projects and codes: none in particular.

7. Reference documents

- L. Mazzocchi: "Proposta per uno studio sperimentale del comportamento del pressurizzatore e della surge-line di un PWR in particolari condizioni" CISE-N.T. 80.069
- L. Augello, R. Martini: "Programma sperimentale di misure in flooding in miscela acqua-vapore ad alta pressione (Pressurizzatore):modalità di esecuzione delle prove" Promemoria CISE-DTN-81.041, 16/2 1981.

8. Degree of availability: to a limited extent.

9. Additional information

Budget: about 460 millions Lit; personnel involved: 5,5 men year.

(CISE , P.O. Box 12081, I-20100 MILANO)

		CLASSIFICATION : 1.1.2. 4.2.
TITLE (ORIGINAL LANGUAGE) : TERMOIDRAULICA DEI GENERATORI DI VAPORE IN CONDIZIONI INCIDENTALI		COUNTRY : ITALY
		SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : THERMOHYDRAULICS OF STEAM GENERATORS UNDER ACCIDENT CONDITIONS		ORGANISATION : CISE
		PROJECT LEADER : L. MAZZOCCHI
INITIATED : 1980	COMPLETED : TO BE DEFINED	SCIENTISTS : G. CATTADORI, G. MASINI, C. MEDICH
STATUS : IN PROGRESS	LAST UPDATING : APRIL 1982	

1. General aim: to study the thermohydraulics of the secondary (and possibly of the primary) side of a steam generator during various accidental and operational transients both in the primary and secondary circuit.

2. Particular objectives: to measure pressure drops, mass holdup, void fraction, heat transfer coefficients and critical powers in the secondary fluid both in steady state and transient conditions; to verify existing prediction methods or to develop new ones.

3. Experimental facilities, programs: experimental facility is obtained by modifying and connecting two existing CISE loops: the IETI-4 loop (for the primary circuit) and the CIRCE loop (for the second. circuit). The experimental program includes:
   - steady state tests (both in adiabatic and diabatic conditions) with different values of secondary pressure, mass flow rate and inlet quality;

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
THERMOHYDRAULICS OF STEAM GENERATORS UNDER ACCIDENT CONDITIONS	1.1.2. 4.2.

- transient tests, with variations of the primary mass flow rate and inlet temperature and of the secondary pressure, mass flow rate and inlet quality, simulating various accidental and operational situations.

The possibility of performing tests with primary pressure variations (simulation of LOCA conditions) is also under study.

4. Project status
   1) Progress to date: construction of test section completed; experimental circuit almost ready
   2) Essential results: none.
5. Next steps: test section installation; instrumentation calibration; experimental tests.
6. Relation to other projects and codes: TRAGEN code, for the thermohydraulic analysis of steam generator behaviour under steady-state and transient conditions.
7. Reference documents:
   - L. Mazzocchi, L. Meini: "Proposta di un programma sperimentale sui transitori termoidraulici di generatori di vapore PWR" CISE, N.T. 80.095;
   - L.Meini, E. Vocino: "Esperienze con l'elemento di prova 3x3 G.V.: impianto, matrice sperimentale e procedure di prova" Promemoria CISE SAT-81.085;
   - G. Cattadori, L. Mazzocchi: "Esperienze sulla termoidraulica dei generatori di vapore PWR: impianto, sezione di prova, programmi e procedure sperimentali" CISE N.T. to be published.
8. Degree of availability: to a limited extent
9. Additional information: Budget 630 millions Lit; personnel 5,6 men year.

(CISE - P.O. Box 12081, I-20100 MILANO, Italy)

	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: E.N.E.A.
TITLE (ENGLISH LANGUAGE): Influence on DNB of Subchannel Obstructions and Rod Bowings in BWR Bundles with Non-Uniform Heat Flux Profiles	ORGANISATION: E.N.E.A. 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) : COMPORTAMENTO TERMOIDRAULICO DI UN PRESSURIZZATORE IN CONDIZIONI INCIDENTALI	COUNTRY : ITALY SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : THERMOHYDRAULIC BEHAVIOUR OF A PRESSURIZER DURING ACCIDENT CONDITIONS	ORGANISATION : CISE PROJECT LEADER : R. MARTINI
INITIATED : 1980 / COMPLETED : 1983 STATUS : IN PROGRESS / LAST UPDATING : APRIL 1982	SCIENTISTS : L. AUGELLO, C. MEDICH

1. General aim: to determine the water level transient of a pressurizer during relief valve discharge.

2. Particular objectives: i) to determine the behaviour of steam and water in counter-current flow along the surge-line ii) to study single and two-phase critical flow in the safety relief valve under dynamic conditions.

3. Experimental facilities and programme: CIRCE and IETI-1 loops in EMILIA Power Station, Piacenza, Italy. Tests with different surge-line geometries in reduced scale; parametrization of pressure and level in the pressurizer. Measurement of critical flow discharge in reduced scale under steady-state and dynamic conditions, with parametrization of upstream pressure, quality and valve opening.

4. Project status

   4.1 Progress to date: experimental measurements of counter current flow limitation (flooding) completed.Design of experimental arrangement for Safety Relief valve (SRV) investigation initiated.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
THERMOHYDRAULIC BEHAVIOUR OF A PRESSURIZER DURING ACCIDENT CONDITIONS	1.1.2.

4.2 Essential results: curves of maximum liquid flowrate vs. steam flowrate obtained under different conditions as well as pressure drops across the surge-line.

5. Next steps: comparisons of obtained experimental data with correlations; detailed design of SRV test facility and procedure.

6. Relation to other projects and codes: none in particular.

7. Reference documents
   - L. Mazzocchi: "Proposta per uno studio sperimentale del comportamento del pressurizzatore e della surge-line di un PWR in particolari condizioni" CISE-N.T. 80.069
   - L. Augello, R. Martini: "Programma sperimentale di misure in flooding in miscela acqua-vapore ad alta pressione (Pressurizzatore):modalità di esecuzione delle prove" Promemoria CISE-DTN-81.041, 16/2 1981.

8. Degree of availability: to a limited extent.

9. Additional information

   Budget: about 460 millions Lit; personnel involved: 5,5 men year.

(CISE, P.O. Box 12081, I-20100 MILANO)

	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 (AMN)
INITIATED: January 1978 / COMPLETED: June 1980 STATUS: Completed / LAST UPDATING: June 1980	SCIENTISTS: A. Azzalin (CISE) C. Medich (CISE)

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. Analysis of data has been performed and results agree with those found in 1980 with a tubular test section.

45.897

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, AMN , 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 (AMN)
INITIATED: January 1979 / COMPLETED: June 1980	SCIENTISTS: R. Granzini (CISE), R. Martini (CISE), G. Barzoni (CISE)
STATUS: Completed / LAST UPDATING: June 1980	

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 an Programme

The experiments were carried out on the CISE IETI-1 loop.
The following test conditions were investigated:
- pressure: 30-70 bar
- specific mass flowrate: 125-1000 kg/m^{2}s
- outlet quality : up to superheated steam
- wall temperature: up to 800° C.

Project Status

The test section, which was particularly instrumented, was proved to work successfully.
A lot of temperature profiles in the liquid deficient region and in a vapor superheated region were obtained.
They were compared with predictions from most current correlations like Dougall-Rohsenow, Heinemann and Groeneveld.
Data are also compared with those obtained in 4X4 geometry.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Post-dryout experiments in tubular geometry	1.1.2

Reference Documents

- G. Barzoni, R. Martini, "Procedure sperimentali e di elaborazione dati prove ultracrisi NUCLITAL" Promemoria DTN/79/SAT/007.
- G. Barzoni, R. Martini, "Studio del coefficiente di scambio termico in condizioni di ultracrisi mediante esperienze effettuate con un elemento di prova tubolare", CISE - NT 80.054.
- G.P. Gaspari, G. Barzoni, R. Martini, "Post-dryout heat transfer tests in a two-section heated tube", Paper C2 at European Two-Phase Flow Group Meeting, Glasgow 1980.
- G.P. Gaspari, "Post dryout heat transfer in a two-section heated tube", NUCLITAL T-RP - 80.006.

Degree of Availability

Proprietary.
G.P. Gaspari, AMN, Via G. D'Annunzio 113, I-16121 Genova.

		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: 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	

	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: ENEA 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

1983-06-29		CLASSIFICATION : 1.1.2
TITLE (ORIGINAL LANGUAGE) : FIX-II som ISP-15		COUNTRY : Sweden SPONSOR : Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE) : FIX-II as international standard problem 15		ORGANISATION : STUDSVIK ENERGITEKNIK AB PROJECT LEADER : O Sandervåg
INITIATED : 82-12-01	COMPLETED : 84-01-01	SCIENTISTS : D Wennerberg, ETU
STATUS : In progress	LAST UPDATING : June 1983	

General aim

To host a blind international standard problem using a FIX-II experiment.

Particular objectives

To provide an adequate description of the facility for blind calculations and to specify parameters to be calculated, to evaluate experimental results with respect to the calculated parameters, to evaluate the calculations and to produce final draft conclusions.

Project status

Basis for the calculations have been distributed and calculated results are received.

Next steps

Evaluation of the calculations.

Reference documents

Quarterly progress reports are issued.

Availability

Open. Requests are sent to Division of Research, Swedish Nuclear Power Inspectorate, Box 27106, S-102 52 STOCKHOLM, SWEDEN.
NR6 PG

8369

1983-06-29	CLASSIFICATION : 1.1.2/1.2
TITLE (ORIGINAL LANGUAGE) : Termohydraulisk haverianalys (THYHAV-I)	COUNTRY : Sweden SPONSOR : Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE) : Thermal-hydraulic Accident Analysis	ORGANISATION : STUDSVIK ENERGITEKNIK AB PROJECT LEADER : O Sandervåg
INITIATED : 82-01-01 — COMPLETED : 83-11-01 STATUS : In progress — LAST UPDATING : June 1983	SCIENTISTS : R Persson Ö Rosdahl A Sjöberg J Eriksson K-A Flygvall

General aim

To provide and maintain advanced thermal-hydraulic computer codes for independent LOCA analysis of Swedish nuclear power reactors, and to perform research on hypothetic and real accident sequences.

Particular objectives

Maintenance of RELAP5 and development of a program package for LOCA analysis of BWRs, analysis of the Oskarshamn 2 reactor (large breaks) and Ringhals 2 reactor (small breaks), qualitative code assessment using FIX-II experiments and international standard problems, developmental assessment of heat transfer correlation, investigation of modelling requirements for prediction of system behaviour during small breaks.

Project status

Most activities listed are in progress. Planned efforts on O2 calculations, standard problems and small break modelling requirements are near completion.

Application and asessment of RELAP5/Mod1 reveal needs for improvements in some of the physical models. It has been found that the code predicts well an overall system behaviour during small breaks with a relatively small number of nodes.

NR6 PG

8369

Next steps

Proposed future work encompass: Transient analyses, development of an integrated program package for PWR analysis, development work for plant analyzers, continued research on small break and large break LOCAs.

Reference documents

Quarterly progress reports are issued.

Availability

Generally open. Requests are sent to Division of Research, Swedish Nuclear Power Inspectorate, Box 27106, S-102 52 STOCKHOLM, SWEDEN.

		CLASSIFICATION: 1.1.2/1.1.3/1.1.4/1.1.
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: AA DEBENHAM, S BOOTH
STATUS:	LAST UPDATING: SEPTEMBER 1980	

BACKGROUND

Currently the USNRC specified 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 continuing of 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 uncertainty in predictions. (Report on L1-5 calculations available during 1980.)

3. Make initial review of uncertainty margins applicable to commercial PWR LOCA analysis.

FACILITIES

Analytical codes.

RELAP TRAC

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:

INITIATED: 1974	COMPLETED:	SCIENTISTS: AA DEBENHAM, S BOOTH
TATUS: CONTINUING	LAST UPDATING: SEPTEMBER 1980	

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

Standard Problem 9 (LOFT L3-1) in progress. Sensitivity of predictions to modelling ıssumptions available.

4. FUTURE WORK

CSNI meetings identify outstanding problem areas in the LOCA/ECCS field and try to select suitable experimental work for comparisons. Calculations for LOFT L2-5 (large break) and L3-6 tests (small break with pumps left running) proposed for 1981. These are Standard Problems 13 and 11 respectively.

	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: J. Ralph, G. Costigan (Harwell)
STATUS: in progress / LAST UPDATING: July 1980	

1. The aim of the project is to provide new experimental data to enable a new correlation for post dryout heat transfer in the low quality low flow regions to be compiled.

2. The experimental facility for the work is a closed loop low pressure (5 bar) water rig situated at Harwell. The test sections to date have all had the common feature of a large thermal inertia to enable data to be obtained in both transient and steady state (1,2).

3. A considerable amount of data have been taken in both up and down flow covering mass velocities of 48 - 720 kg m^{-2} s^{-1} and qualities from -0.05 to 0.5 at a pressure 3 bar (3). The data show that the heat flux for any wall temperature and mass velocity is a minimum near an equilibrium mass quality of zero. Experiments with upward and downward flow show that a minimum heat flux occurs for low mass velocity downward flow.

4. Recent work has concentrated on test section manufacturing techniques. It is planned in 1980/81 to carry out more detailed rewetting studies using surface probes.

5. Recently published information from the ARGONNE NATIONAL LAB (4) indicates similar findings to the Harwell studies using in fact very similar techniques.

(1) NEWBOLD, F, RALPH, J.C. and WARD, J.A., AERE. R8390 (1977)

(2) RALPH, J.C., SANDERSON, S and WARD, J.A., Proc. ASME Winter Meeting on Nuclear Reactor Safety Heat Transfer. Atlanta, Georgia, Nov. 27 - Dec. 2 1977.

(3) BARNARD, D, GLASTONBURY, A.G. and WARD, J.A. Paper presented to the European Two Phase Flow Group, ISPRA, June 1979.

(4) CHENG, S.C. et al. NUREG/CR.1054 and ANL-79-78.

	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: P B Whalley (Harwell)
STATUS: In progress / LAST UPDATING: July 1980	

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. Steadystate flow in a rod bundle[3,4]. It has been shown that reasonable results are obtained for PWR bundles as long as the dryout occurs at a quality of above about 10%.

Objectives

1. To continue investigations into the processes of entrainment and deposition in annular two-phase flow.

2. To verify the extension of the analysis and numerical methods for transient flow to cover pressure transients in a round tube, and to compare the results with experimental data.

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

Objective 2 should be completed in early 1981 and items 3 and 4 by mid 1982. Objective 1 is of a continuing nature.

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

Classification 1.1.2

Title 1 : Blowdown of a steam drum

Title 2

Initiated 1976

Completed

Status

In progress

Last Updating Oct. 1980

Country U.K.

Sponsor C.E.G.B.

Organisation C.E.R.L.

Project Leaders

Dr P.R. Farmer

Dr D.J. Woodford

1. General Aim

To study depressurization phenomena in large vessels, with particular emphasis on rapid pressure loss typical of LOCA conditions, using Freon 12 as a modelling fluid.

2. Specific objectives

To formulate computer codes capable of predicting the major effects of downcomer voidage, liquid flashing, vapour drawdown and liquid level swell, with and without a subcooled layer in the drum.

To establish a data-base against which future models may be compared.

To perform an integral effects experiment to determine the interations between the separate effects of bubble nucleation, flashing, bubble rise, bubble coalescence, vapour separation in the pool and heat transfer from the walls, especially watching for any unexpected effects.

3. Experimental Facilities

A drum slice 1.6m dia. by 1.6m long is connected to a receiver vessel, containing 3 tonnes of copper plates. By precooling the receiver vessel with liquid nitrogen and evacuating it, a low pressure sink is provided to allow drum blowdown into a reasonably constant pressure enclosure. The receiver vessel is mounted on load cells to measure the total mass flow, and the whole drum circuit is fully instrumented with void fraction meters, thermocouples, pressure transducers and orifice plates. Also windows in the drum allow visual observations and photographic studies.

The drum experiment is connected to the CERL Freon 12 facility, and uses the dedicated computer to digitally record all 64 channels of data onto magnetic disk at aggregate rates up to 40,000 readings per second.

4. Project status

The experimental work programme is due to be completed in December 1980.

5. Present results and references

Blowdowns from both the liquid and vapour spaces, separately and together have been performed at rates up to 1 bar per second, and for initial pressures up to 15 bar. Both bursting discs and a fast acting valve have been used to initiate blowdown, and the effects of subcooled layers have been studied. Some of these results were reported at the European Two Phase Flow Group meeting in Glasgow as paper A6.

A computer model for the steady state behaviour of the drum has been completed and work on a transient code is in progress.

Classification 1.1.2

Title 1: Thermohydraulic Safety Studies Multipin Cluster Rig	Country: U.K.
Title 2: Effects of fuel pin ballooning on Reflood Heat Transfer	Sponsor: CEGB/CEC
	Organisation: CEGB
Initiated 1.6.78 / Completed	
Status continuing / Last Updating 27.10.80	Project Leaders: S.J. Board, S.A. Fairbairn

The aim of this project is to study the effects on heat transfer of emergency coolant flow diversion around localized regions of gross fuel pin distortion.

Electrically heated rods (∿ 1 metre heated length are used as fuel pin simulators. These rods are supported vertically in a cylindrical silica sleeve (to provide a facility for flow visualization) at near atmospheric pressure. Pin ballooning is 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 coplanar blockage over a fraction (∿ 30%) of the area of the rod cluster. Both solid and hollow (thicknesses down to ∿ 0.25 mm) sleeves will be used. Direct measurements will be made of superheat in the steam and steam-water flows used to cool the rods. Rod temperatures are measured by internal thermocouples. Tests with an undistorted rod bundle to provide a basis for comparison with the "ballooned" pin tests outlined above are now underway.

1.1.3. REACTIVITY EFFECTS

	CLASSIFICATION: 1.1.2/1.1.3/1.1.4/1.1.5
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: AA DEBENHAM, S BOOTH
STATUS:	LAST UPDATING: SEPTEMBER 1980	

BACKGROUND

Currently the USNRC specified 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 continuing of 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 uncertainty in predictions. (Report on L1-5 calculations available during 1980.)

3. Make initial review of uncertainty margins applicable to commercial PWR LOCA analysis.

1.1.4. DECAY HEAT

		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: ENEA 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	SCIENTISTS: F. ORIOLO G. MEI G. CECCARELLI G. BITETTI
STATUS: in progress	LAST UPDATING: May 1979	

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1.1.2, 1.1.4., 1.2</td></tr>
<tr><td colspan="2" rowspan="2">TITLE (ORIGINAL LANGUAGE) :
Analisi dei transitori termo-idraulici nel circuito primario dei reattori ad acqua leggera a seguito di LOCA.</td><td>COUNTRY :
ITALY</td></tr>
<tr><td>SPONSOR : ENEA
(formerly CNEN)</td></tr>
<tr><td colspan="2" rowspan="2">TITLE (ENGLISH LANGUAGE) :
Analysis of thermo-hydraulic transients in LWR following LOCA</td><td>ORGANISATION :
University of Pisa</td></tr>
<tr><td>PROJECT LEADER :
Marino MAZZINI</td></tr>
<tr><td>INITIATED :
1979</td><td>COMPLETED :
1987</td><td rowspan="2">SCIENTISTS : N.CERULLO
F.ORIOLO, U.ROSA, P.VIGNI,
F.D'AURIA, R.BOVALINI,
G.GALASSI, M.CARCASSI.</td></tr>
<tr><td>STATUS :
in progress</td><td>LAST UPDATING :
January 1982</td></tr>
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Page not used

	CLASSIFICATION: 1.1.2/1.1.3/1.1.4/1.1.5
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: AA DEBENHAM, S BOOTH
STATUS: / LAST UPDATING: SEPTEMBER 1980	

BACKGROUND

Currently the USNRC specified 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.

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(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 continuing of 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 uncertainty in predictions. (Report on L1-5 calculations available during 1980.)

3. Make initial review of uncertainty margins applicable to commercial PWR LOCA analysis.

	CLASSIFICATION: 1.1.2/1.1.3/1.1.4/1.1.5
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: AA DEBENHAM, S BOOTH
STATUS: / LAST UPDATING: SEPTEMBER 1980	

BACKGROUND

Currently the USNRC specified 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 continuing of 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 uncertainty in predictions. (Report on L1-5 calculations available during 1980.)

3. Make initial review of uncertainty margins applicable to commercial PWR LOCA analysis.

1.2. PERFORMANCE OF ECCS

Berichtszeitraum/Period 1.01.82 - 31.12.82	Klassifikation/Classification 1.2	Kennzeichen/Project Number 150 366 A
Vorhaben/Project Title Fluiddynamische Effekte im Brennelementkopfbereich während des Wiederauffüllens und Flutens Fluid dynamic effects in the upper end box region of the fuel bundle during refill and reflooding		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Institut für Verfahrenstechnik der Universität Hannover
Arbeitsbeginn/Initiated 01.01.1981	Arbeitsende/Completed 31.12.1983	Leiter des Vorhabens/Project Leader Prof.Dr.-Ing.F.Mayinger
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General aim

Investigation of the phenomena during the Emergency-Core-Cooling (ECC) of Light-Water-Reactors (LWR)

2. Particular Objectives

Experimental and theoretical investigations of the phase-separation mechanisms in orifices, throttling devices and in the upper End-Box (EB) of a German Pressurised Water Reactor (FRG-PWR) during refilling and reflooding. This investigations are aimed to improve the reactor computer codes; for example TRAC.

Starting with investigations using simple geometries - single and multihole orifices - the basic phenomena of the de- and reentrainment of air-water-flow and steam-water-flow will be investigated with step by step approaching and imitating the real reactor geometries. In parallel, a theoretical model and it's mathematical formulation will be developed.

3. Research Program

3.1 Basic investigations of entrainment and counter-current flow (CCF) in an air/water test section with a 4 x 4 rod bundle geometrie.

3.2 Fluid dynamic effects of steam-water-mixtures

3.2.1 Test device with 4 x 4 rod bundle geometrie.

3.2.2 Test device with 16 x 16 rod bundle geometrie.

01.01.82-31.12.82 150 366 A

4. Experimential Facilities

To 3.2.1 The 4 x 4 rod bundle is instrumented with 8 thermocouples to measure the fluid temperatures.

To 3.2.2 The manufacture of the test section for the 16 x 16 rod bundle is started.

5. Progress to Date

To 3.1 At the air-water test facility the investigations are finished. Furthermore the computer program of automatical image analysis for an objective and efficient evaluation of droplet photographs was improved.

To 3.2.1 At the steam/water test facility the experiments were performed to investigate the counter-current flow behaviour at a system pressure of 5 bar. The water was injected with saturation and subcooling.

To 3.2.2 The design of the test device with 16 x 16 rod bundle geometrie is finished.

6. Results

To 3.1 Extensive theoretical investigations have confirmed that the flooding behaviour of air/water counter-current flow can be described by dimensionless groups formed by superficial velocities and thermodynamic properties /1/ derived from theoretical models. The flooding curve can be expressed in general form by the Kutateladze correlation:

$$\sqrt{K_G} = C - M \cdot \sqrt{K_L}$$

$$\sqrt{K_i} = j_i \cdot \rho_i^{1/2} \left[g \cdot \sigma \left(\rho_L - \rho_G \right) \right]^{-1/4} \qquad i = G, L$$

For the predetermination of the coefficients M and C mathematical models, which consider the geometries and the thermodynamic properties are developed.

To 3.2 The evaluation of the first experiments at the steam/water test device shows that the results of saturated water injection are comparable with these of the air/water tests. If the water is subcooled injected the counter-current flow limitation (CCFL) is shifted to greater steam fluxes in consequense of a partial steam condensation.

Inspection of the defective heater rods indicated a short-circuit between two power zones.

Re. 3.9 Ordering and delivery of the hardware components for PKL II B has been financed by additional funds by the BMFT in IV/82.

7. Next Steps

Re. 3.9 Further preparatory work on the PKL II B tests.
Repair of the bundle by replacing the defective heater rods and insulation of the power zone boundaries.
Extension of the instrumentation for the PKL II B tests (tests including the end-of-blowdown phase).

Re. 3.10 Completion of the project and documentation of the tests carried out in the course of the project.
The PKL test series, in particular the EoB tests, will be continued as part of a different project.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	1.2	150 287
Vorhaben/Project Title Wiederauffüllversuche mit Berücksichtigung der Primärkreisläufe Versuchsphasen IB-IE und II Reflood Tests with Regard to the Primary Loops (PKL) Test Phases IB-IE and II		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor KRAFTWERK UNION AG Reaktortechnik R 513, Erlangen
Arbeitsbeginn/Initiated 01.09.77	Arbeitsende/Completed 31.03.83	Leiter des Vorhabens/Project Leader Prof. Dr. D. Hein
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31.12.82	Bewilligte Mittel/Funds

1. General Aim

Experimental investigation of the refill and reflood phases during a LOCA using a sufficiently large model of the entire PWR primary loop.

2. Particular Objectives

The goal of this program is to conduct experiments at the PKL Test Facility under simulated LOCA conditions with large, medium-size and small breaks at a maximum pressure of 40 bar and bundle heatup power of 1,45 MW (approx. 5 % decay heat), to investigate the effects of: break size, break position, mode of injection, rate and distribution of hot and cold leg coolant injection, distribution of water inventory in the primary system, energy transfer to EC coolant within the primary system, back pressure (containment), on the thermal hydraulic behaviour of the system and the heat removal from the core during ECC.

3. Research Program

3.1 Performance of additional refill reflood tests simulating large break LOCA conditions (PKL IB)

3.2 Performance of steady-state and transient tests simulating LOCA conditions with small breaks (PKL ID)

3.3 Performance of one "End of Blowdown Test" (PKL IE)

3.4 Additional instrumentation especially for the 2nd test bundle

3.5 Modification of the test facility

3.6 Installation of the second bundle

3.7 Shake Down Tests

3.8 Performance of Test Series PKL II A (coupling tests to PKL I)

01.01.82 - 31.12.82 150 287

3.9 Modification of the facility for PKL II B test including End of Blowdown (EoB)

4. Experimental Facilities

The PKL test facility simulates as closely as possible the primary loops including active steam generators of a 1300 MW plant. The scaling factor is 1 : 145 (refering to the number of heater rods). The functional design of the test facility has the following specific features:

- Exact simulation of core geometry and decay heat
- Exact simulation of all reactor elevations (1 : 1) and flow resistances
- Close simulation of the loop volumes (1 : 145) and the thermal capacities of the primary and secondary loops

5. Progress to Date

Re. 3.7 Completion and putting into operation of the test facility. Commissioning work has been carried out on the advanced instrumentation in the presence of the instrument manufacturers from the USA. In-service test have been carried out on all the instrumentation and the resultant data handedover to the instrument manufacturers.

Re. 3.8 Owing to shortage of funds, the PKL II A series had to be reduced to a single test. This test has been carried out and documented. The lower plenum has been dismantled. Several defective heater rods have been inspected and new ones ordered.

Re. 3.9 Planning work on PKL IIB (EoB) has been continued in respect of modifications to the test facility, the mode of operation and the instrumentation. The ordering of parts for the extension of the PCM data acquisition system has been effexted ahead of schedule.

6. Results

Re. 3.8 The shake down tests II A and the II A 9 follow-up test have been carried out and II A 9 has been evaluated. The results yielded by conventional instrumentation were in good agreement with those from the advanced instrumentation.

7. Next Steps

To 3.1 Improvement of the computer code for the automatical image analysis.

To 3.2.1 Evaluation of CCFL measurements.

To 3.2.2 The test facility for the 16 x 16 rod bundle will be set up and put into operation.

8. Relation with other Projects

BMFT-Research programme RS 2D/3D,
special 150 500, RS 314

9. References

/1/ Kröning, Hawighorst, Mayinger:
The influence of flow restrictions on the counter current flow behaviour in the fuel element top nozzle area, ETPFGM June 1982, Paris

/2/ Progress Project Report No 4A (01.07.82 - 31.12.82)

10. Degree of Availability of the Reports

The reports are available by the GRS, Köln.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	1.2	1500 500
Vorhaben/Project Title Errichtung und Inbetriebsetzung der UPTF Construction and startup of the UPTF		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf /Contractor KRAFTWERK UNION AG Reaktortechnik VE 95, Erlangen
Arbeitsbeginn/Initiated 18.09.81	Arbeitsende/Completed 30.04.85	Leiter des Vorhabens/Project Leader Dr. M. Sawitzki
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31.12.82	Bewilligte Mittel/Funds

1. General Aim

A joint research program was agreed upon between Japan, the USA and the Federal Republic of Germany for investigating multi-dimensional flow effects during the refill and reflooding phase of a pressurized water reactor after a loss-of-coolant accident (2D/3D project). The German contribution encompasses the construction of the upper plenum test facility (UPTF) and the performance of tests.

2. Particular Objectives

The supplies and services for the construction of the Test facility UPTF as specified in the BMFT (Federal Ministry of Research and Technology) R&D contract 1500 500 including interim amendments.

3. Research Program

3.1 General activities
3.2 Civil engineering
3.3 Mechanical engineering
3.4 Electrical engineering

4. Experimental Facilities

The facility will be erected at the large fossile power plant in Mannheim (GKM).

5. Progress to Date

Re. 3.1 As a result of the TRAC computation results and subsequent amendments, additional planning and calculation became necessary for Points 3.2, 3.3 and 3.4.

Re. 3.2 The site management hut and amenties area have been erected. The site was opened in June 1982. 90 % of the reinforced concreting of the administration and switchgear buildings have been completed. As a result of 3.1, the test building and its foundations will have to be re-dimensioned. Test bores have been drilled and an expert analysis of the ground prepared.

Re. 3.3 The contract was awarded for the test vessels, the materials ordered and the manufacturing documents were prepared. The entire vessel internals are at present in the process of manufacture or being converted. The fuel assembly Endboxes to be equipped with instrumentation were forwarded to the USNRC for this purpose.

As a result of 3.1, it was necessary to perform extensive piping modifications. Piping manufacture on the site was commenced.
The documents for the placement of valve orders were completed. The pumps and fans are at present in the process of manufacture.
The manufacturing documents for the building crane were completed.
Orders were placed for the heating, ventilation and air conditioning plant.

Re. 3.4 Federal German scope of supply: Dimensioning of the site power supply system, design of the earthing unit, switchgear and motors were performed. Investigations and planning activities with regard to the conversion of system JGK and JKF from open to closed-loop control (core simulation feedback). Equipment specifications in respect of process instrumentation and the planning of switchgear and cable trays were completed.
US scope of supply: Instrumentation control document, Rev. 0 was prepared for the advanced instrumentation and planning activities performed with regard to its installation in test vessels, test vessel internals and primary loops.

6. Results

Orders have been placed for the following: accumulators, core simulator, fuel assembly dummys, fuel assembly Endboxes, process-related I&C, hot-water tank, steam storage tank, containment-simulator, water collecting tank, pumps, pump simulators, silencers, insulation, flow rectifiers, motors, communication system, fire alarm system, lighting, water coolers, N_2 tanks, steam generator simulators, moisture separators, air conditioning.

7. Next Steps

Re. 3.1 Coordination of activities up to hand-over of plant to contractor.

Re. 3.2 Continued work on the erection of the administration and switchgear building, and the test building.

Re. 3.3 Continued work on manufacture and installation of the components in the UPTF.

Re. 3.4 Continued work on the manufacture, supply and installation of electrical components, and planning coordination and assembly of A. I. and DAS.

8. Relation with Other Projects

Sponsored projects BMFT 150 395, 150 363, 150 619

9. References

TRAC-Code PDII, PFI, Las Alamos National Laboratory

10. Degree of Availability

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1/1 - 12/31/82	1.2	1500522
Vorhaben/Project Title Supplementary Supplies and Services for the Construction and Startup of the UPTF within the 2D/3D Project		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Grosskraftwerk Mannheim AG
Arbeitsbeginn/Initiated 3/18/1981	Arbeitsende/Completed 12/31/85	Leiter des Vorhabens/Project Leader Dipl.Ing.Baumüller
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

Japan, USA and the Federal Republic of Germany are conducting a cooperative research program to study the multidimensional flow effects during the refill and reflood phases following a loss of coolant accident in a pressurized-water reactor (2D/3D Project). The German project share consists of the construction of the "Upper Plenum Test Facility "(UPTF) and the performance of the test program.

2. Particular Objectives

Supplementary supplies and services for the construction and startup of the UPTF including the provision of the necessary site area.

3. Research Program

3.1 Provision of the site area required for the construction of the test facility, the site facilities and the assembly sites.

3.2 Procurement of the required authority licenses in the name and by the assistance of the rpincipal and the KWU.

3.3 Rerouting of cables (burying of the 110 kV open-air transmission line).

3.4 Preparation and development of the site.

3.5 Gate guard and patrol services.

3.6 Furnishing of the operating media steam, condensate, and cooling water including supply and discharge lines to the test building.

3.7 Electric power supply to the test facility including cabling up to the input terminal.

3.8 Furnishing of space for and installation of the pump and appurtenances to be erected in the contractor's turbine building.

3.9 Infrastructural measures at the site.

4. Experimental Facilities, Computer Codes

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5. Progress to Date

ad 3.1 The boundaries of the 3D-test site have been defined.

" 3.2 The construction license and release of construction work have been procured. Test stand statics about to be completed.

" 3.3 Cabling of the 110 kV open-air transmission line has been completed.

" 3.4 Test site has been cleared, area for site management office and quarters coated with asphalt. A road to the test building will be opened in spring 1983.

" 3.5 So far all 3D-site traffic has been run through a GKM gate. With the start of concrete pouring and steel work, presumably beginning of February 1983, the test site gate will be continuously guarded.

" 3.6 Steam, cooling water and condensate supply and discharge lines (ND 500 mm; lätter - 150 mm) are installed in a piping bridge extending from the GKM turbine hall to the 3D-test building approx.100 m away. The condensate and cooling water lines are attached to the test site systems at +9 m. The steam supply line (20 bar/530°C) is routed across the test building annex roof (+13 m) towards the steam cooler installed in the building. Piping, valve assemblies and steam cooler have been ordered, an invitation for piping supports tenders is being prepared. The start of the steam cooler and the pipe bridge is scheduled for June 1983 corresponding to the construction progress of the test building.

" 3.7 The test facility power of 1600 kVA is supplied from a GKM 6 kV houseload compartment. The 6 kV supply, a continuous 380 V supply, instrument and control cables are attached to three cable trays integrated in the pipe bridge.

" 3.8 The cooling water pump for the test system will be located in the GKM Erection site, thermal hydraulics and scope of supply coordinated with KWU.

ad 3.9 Infrastructural site measures concerning water supply, waste water management, site power and lighting are largely completed. District heat connection of the test building by summer 1983.

6. Results

6.1 Based on construction license (11/81) and release (mid-1982) KWU started construction work in June 1982.

6.2 The test site is established, fenced and provided with an infrastructure.

6.3 The steam, condensate and cooling water supply and discharge lines connections have been coordinated with KWU; the valve assemblies and steam cooler have been ordered.

6.4 The connection of the test facility to the GKM houseload system has been designed and coordinated with KWU.

7. Next Steps

7.1 Piping supports contracts will be awarded in spring 1983. Assembly of piping bridge and steam cooler will start mid-1983.

7.2 An invitation for power supply tenders for the test facility is being prepared.

8. Relation to Other Porjects

1 500 500 "Erection of the UPTF" (KWU).

9. References

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10. Degree of Availability of the Reports

GRS-Forschungsbetreuung

Berichtszeitraum/Period 01.01.1982-31.12.1982	Klassifikation/Classification 1.2	Kennzeichen/Project Number 1500 602
Vorhaben/Project Title Realisierung von Überströmvorrichtungen im 2D/3D-Projekt (UPTF) Realization of Vent Valves within the 2D/3D-Project (UPTF)		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf /Contractor Brown Boveri Reaktor GmbH
Arbeitsbeginn/Initiated 01.01.1982	Arbeitsende/Completed 30.06.1984	Leiter des Vorhabens/Project Leader Dr. Lahner
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 03.01.1983	Bewilligte Mittel/Funds

1. General Aim

The basic objective of this R&D effort is identical to the objective of the international 2D/3D - project. Emphasis is laid, however, on the 3D-test objectives of the 3-dimensional tests planned in Germany, namely:
Identification of the 3-dimensional thermal-hydraulic characteristics of the fluid in the upper plenum and downcomer during the final stages of the blowdown phase, refill phase, and reflood phase of a LOCA in PWR systems.
The so-called Upper Plenum Test Facility (UPTF) needed for the experiments shall, in particular, provide the following boundary conditions:

360-degree mockup of a 4-loop PWR of the 1,200-MW class, including the reactor vessel, the downcomer, the lower plenum, a core simulator, the upper plenum, the containment, and simulated primary loops.
- the UPTF design will largely reflect the design of the KWU-type PWR.
- particular characteristics of other PWR designs are to be accounted for as well (insofar as they affect phenomena in the upper plenum).

2. Particular Objectives

The thermohydraulic processes in the upper plenum of a KWU reactor during the refill/reflood phase differ fundamentally from the corresponding processes in a B&W/BBR reactor. Whereas in the KWU concept cold water is fed, through emergency cooling injection even into the hot legs, directly to the upper plenum where the water will condense the steam and enter the core region from above, in the BBR concept the injection takes place through the cold legs only. To prevent a pressure buildup in the upper plenum vent valves are arranged in the core support cylinder of the BBR reactor which open towards the outside when overpressure in the upper plenum is building up, and they allow the steam to escape into the downcomer and from there through the leak into the containment. USNRC and BMFT have therefore agreed that the B&W/BBR concept shall also be considered by way of installing 8 vent valves in the core support cylinder of the 3D test facility.

This requires that the planning and realization of the vent valves in the UPTF core support cylinder be monitored, procurement of the hardware be ensured, that the tests with unlocked vent valves be planned and specified, and that the implementation of the entire project be followed in depth.

3. Research Program

3.1 Revision of the vent valve design prepared within the scope of project no. 150394; adaptation to the new boundary conditions of the UPTF core support cylinder.

3.2 Contributions to planning, and detailed definition of the tests to be performed.

3.3 Analytical investigations, in particular contributions for the preparation of a TRAC input data record for the BBR reference reactor.

3.4 Contributions to the planning and the specifications for the instrumentation.

3.5 Coordination of project activities with GRS and KWU and with the partners to the tripartite agreement.

3.6 Manufacture of the vent valves.

4. Experimental Facilities, Computer Codes

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5. Progress to Date

Ref. 3.1 Design revisions have been completed; delivery regulations have been prepared and tenders were invited.

Ref. 3.2 The processes in the BBR reactor during the refill/reflood phase, in particular the emergency core cooling injection quantities and locations, have been specified.

Ref. 3.3 Work on contributions for the TRAC input data record has started on schedule.

Ref. 3.4 Data for sensor locations in the region of the instrumented vent valve have been prepared.

Ref. 3.5 Participation in national and international coordination meetings.

Ref. 3.6 The vent valves have been ordered.

6. Results

Ref. 3.1 After the revised engineering drawings were discussed with KWU and GRS the vent valves were released for fabrication.

Ref. 3.2 The present design of the UPTF guarantees a satisfactory modelling of the processes in a BBR reactor during the refill/reflood phase.

Ref. 3.4 The sensor locations in the downcomer and in the region of the vent valves were defined not only by BBR requirements but also by the boundary conditions of the design of the experimental vessel. Satisfactory compromises have been reached.

7. Next Steps

Activities still to be carried out will be performed and completed in accordance with the progress of the overall project.

8. Relation with Other Projects

The project is closely related to the activities for the 2D/3D project (UPTF) as a whole, with respect to both contents and schedule.

9. References

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10. Degree of Availability of the Reports

GRS-Forschungsbetreuung

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1982	1.2	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 31.12.1982	Leiter des Vorhabens/Project Leader Prof. Dr. E. Hicken
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

1. General Aim

Parallel and complemantary 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.

2.4 Development of DRUFAN.

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-code.

3.8 Implementation of foreign codes.

3.9 Development of DRUFAN.

4. Experimental Facilities, Computer Codes

Experimental facilities: PKL, CCTF, SCTF (in operation).
UPTF (construction).
Computer Codes: TRAC, DRUFAN, FLUT.

5. Progress to Date

ad 3.1 Steady state calculation with TRAC-PF1 for the German Reference Reactor (GPWR) has been performed and input data for this reactor have been transmitted to LANL. Calculations for all phases of a LOCA with a double ended cold leg break and degraded ECCS (only 2 hot leg injections) have been performed with TRAC-PF1 successfully. SCTF run 507 was post-calculated with TRAC-PF1 (7.4). Assistance for the Upper-Plenum-Test-Facility (UPTF) with special emphasis on instrumentation and core simulator design and performance.

ad 3.3 From PKL test series I 4 posttest-calculations and for series II 3 predictions with TRAC-PF1 have been performed. In addition 2 posttest-calculations (series I) and 1 prediction (series II) with FLUT.

ad 3.6 Models in FLUT have been improved.

ad 3.9 Documentation for the DRUFAN-02 code was completed.

6. Results

ad 2.1 The main results from the TRAC-PF1 calculation for the double ended cold leg break GPWR have been

- the pressurizer flow during the blowdown phase results in a precooling of the core,
- near the hot leg injection locations water break-through into the core while in areas near hot legs with no ECC injection steam flow from the core into the upper plenum was calculated,
- the maximum cladding temperatures exist during the blowdown phase.
- The oscillatory behaviour of water plugs are different in the 4 loops.

Post-calculations of 4 PKL tests with TRAC-PF1 are in a satisfactory agreement with experimental data.

ad 2.2 Posttest-calculations with FLUT for tests in FLECHT, FEBA,SCTF and PKL have been satisfactory.

ad 2.3 TRAC-PF1 (7.6) has been implemented on the Amdahl 470/V8 computer. This version could be used also on computers of IBM type.

7. Next Steps

As specified.

8. Relation to other Projects

Within code development some other test data are used.

9. References

A number of interoffice notes were prepared which will later be published in reports.

10. Degree of Availability of the Reports

Through GRS-Forschungsbetreuung.

Berichtszeitraum/Period 01.Jan.19 82-31.Dec.1982	Klassifikation/Classification 1.2	Kennzeichen/Project Number RS 428
Vorhaben/Project Title Abstellung eines Resident Engineer für das 2D/3D-Projekt nach Japan(JAERI) 2D/3D-Project Resident Engineer for Japan (JAERI)		Land/Country F.R.G. Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit (G R S)
Arbeitsbeginn/Initiated 1.Sept. 1979	Arbeitsende/Completed 31.Aug.1984	Leiter des Vorhabens/Project Leader Dipl.-Phys. H.-G.Herdtle
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

1. General Aim

The main objective of the resident engineer program in Japan is to promote communication between JAERI and the GRS in the 2D/3D cooperative program.

2. Particular Objectives

The Japan Resident Engineer shall provide international communication and liaison between JAERI and GRS for the 2D/3D program which involves coupling tests between the Slap Core Test Facility (SCTF) and the Upper Plenum Test Facility (UPTF), experimental test information for GRS reflood studies, and informaton for TRAC code calculations in support of experimentation at JAERI8s reflood test facilities in Japan. Some of the specific tasks to be performed by the Japan Resident Engineer are as follows:

2.1 Maintain contact with and cognizance of the activities and accomplishments of the Japanese 2D/3D Program.

2.2 Attend JAERI meetings concerned with the discussion of instrumentation requirements, design reviews, schedules, program status, test plans, and the review of test data pertinent to the 2D/3D program.

2.3 Provide GRS and its FRG contractors with the up-to-date information on the status of design, requirements, test results be sending appropriate technical reports, drawings, and data promptly to the GRS.

2.4 Participate in JAERI design and experimental activities as designated by and with with the approval of the JAERI Project Manager to the extent that such activities do not impact on the time required for liasion duties.

2.5 Assist GRS in organizing and scheduling meetings in Japan/ USA/ FRG.

2.6 Present technical reviews of the JAERI 2D/3D program to GRS and its contractors on GRS's request.

2.7 Participate in all appropiate technical meetings.

2.8 Identify potential problem areas in the Japanese program involving the FRG interest and alert GRS of the need for information and/or action.

3. Research Program

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4. Experimental Facilities, Computer Codes

5.&6. Progress to Date and Results

Work performed during the reporting period:
Assistance to JAERI was provided in interpretation and particial translation of technical documents concerning the German Upper Plenum Test Facility (UPTF) and PKL.
Participation in the 2D/3D coordination and expert meetings at JAERI in March 82. Assistance in preparing papers and presentations was provided. Current information on the tests of the Cylindrical Core Test Facility (CCTF) and of the Slap Core Test Facility (SCTF), problems in the instrumentation performance during the experiments and progress in CCTF and SCTF testing has been reported.
Assistance in writing Quick Look Reports of the CCTF and SCTF tests was provided. Participation in meetings JAERI and its subcontractors. Communication to the GRS and its contractors on the problems caused by the delay on UPTF project related to the Japanese SCTF-Core-III project.
Activities in software development for plotting and indivi-

dual analysis of the CCTF and SCTF test data on the FACOM-OS-IV/F4 computer system.
Assistance in preparing the data reports and analysis reports of CCTF and SCTF test runs..

7. Next Steps

Assistance in technical information exchange will be provided to JAERI and GRS. Progress in CCTF and SCTF testing will be continously reported to GRS.

8. Relations with other Projects

This project is part of the international 2D/3D program performed in Japan, USA, and Germany.

9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.4. - 31.12.1982	1.2	RS 603
Vorhaben/Project Title: Assignment of a Resident Engineer to Los Alamos National Laboratory (LANL)		Land/Country:
		Fördernde Institution/Sponsor: BMFT
		Auftragnehmer/Zuwendgsempf./Contractor: Gesellschaft für Reaktorsicherheit (GRS) mbH
Arbeitsbeginn/Initiated: 1.4.1982	Arbeitsende/Completed: 31.3.1983	Leiter des Vorhabens/Project Leader: Prof. Dr. E. Hicken
Stand der Arbeiten/Status: continued	Berichtsdatum/Last Updating: 31.12.1982	Bewilligte Mittel/Funds:

1. General Aim

The primary objective is to represent the German interest in LANL and the USNRC. The resident engineer observes the US activity in the 2D/3D project and assists in the transfer of information from the 2D/3D project to the BMFT. The resident engineer is involved in many private discussions concerning the planning and organization of the 2D/3D project and aids in the communication between FRG and USA representatives.

2. Particular Objectives

2.1 Assistance in the GPWR and UPTF calculations.

2.2 Assistance in the analysis of the GPWR and UPTF.

2.3 Obtainment of information pertinent to the 2D/3D project.

2.4 Exchange of information concerning the verification of TRAC-PF1.

3. Research Program

the same as item 2.

4. Experimental Facilities, Codes

Performance of analysis with the TRAC-PF1 code.

5. Progress to Date

Ref. 2.1 Listing and modification of the GPWR base case input deck, preparation of the input for the GPWR ECC system model, preparation and assistance in the input model for the UPTF base case calculation.

Ref. 2.2 Assistance in analysis of the GPWR results, verification of the ECC system model and presentation of the results at the Washington TCC Meeting.

Ref. 2.3 Representation of the German interests concerning the TRAC-PF1 models (condensation, interfacial friction), and making available the results to scientists in Germany.

Ref. 2.4 Representation of the German interests concerning the TRAC-PF1 verification and making available the results to TRAC-PF1 code users in Germany.

6. Results

The GPWR base case calculation is complete. The input for the first UPTF system calculation has been prepared.

7. Next Steps

Ref. 2.2 Pursuance of information concerning the analysis of the GPWR and UPTF.

Ref. 2.3 Pursuance of any new information.

Ref. 2.4 Continuation of work on the UPTF calculations.

8. Relation with Other Projects

None.

9. References

Coauthor - "GPWR-1982 TRAC-PF1 Input Deck Description"
Los Alamos Technical Note LA-2D/3D-TN-82-10, November 1982

10. Degree of Availability of the Reports

GRS-Forschungsbetreuung.

145-1-11		1 - 2
TITRE QUALIFICATION-VALIDATION RELAP-TRAC ETUDE DES TRANSFERTS DE CHALEUR : DEPRESSURISATION (OMEGA) - RENOYAGE (ERSEC)		*Pays* FRANCE *Organisme directeur* CEA/DgCS - CEA/DSN
TITRE en anglais QUALIFICATION-VALIDATION OF RELAP AND TRAC CODES:HEAT TRANSFER STUDIES IN BLODOWN (OMEGA) AND REFLOOD (ERSEC)		*Organisme exécuteur* CEA/DSN/SEAREL (FAR) CEA/DRE/STT (GRENOBLE) *Responsables* M. REOCREUX - SEAREL M. COURTAUD - STT
Date de démarrage 01.01.1976	*Etat actuel* en cours	*Scientifiques* DSN : M. MEZZA M. FORGE
Date d'achèvement	*Dernière mise à jour* 01.1983	

1. OBJECTIF GENERAL

Tester les dernières versions des codes américains RELAP 4 et TRAC sur des expériences françaises du type analytique afin de déterminer les possibilités et les limites des modèles de transfert de chaleur utilisés dans ces codes pour les phases dépressurisation et renoyage.

2. OBJECTIFS PARTICULIERS

Tester le jeu de corrélations de coefficients d'échange et les modèles hydrauliques disponibles dans RELAP 4 Mod 6 (ou des versions antérieures Mod 3 et Mod 5),

a) pour la phase dépressurisation en les utilisant pour représenter des expériences de la boucle OMEGA.

b) pour la phase renoyage en les appliquant à des calculs d'expériences réalisées sur la boucle ERSEC.

Tester les modèles du code TRAC sur ces mêmes expériences.

Le contenu de ces tests est le suivant :

Phase dépressurisation

Analyse d'essais OMEGA dont la géométrie de l'élément chauffant est soit un tube refroidi intérieurement, soit une grappe de 36 barreaux dans la géométrie 17 X 17 (pression initiale 150 Bars; flux thermique : 0,60 ou 125 W/cm^2).

Phase renoyage

Calcul d'essais ERSEC, en tube, couvrant une gamme intéressante de paramètres (débit d'injection, température de l'eau d'injection température initiale de paroi, pression) puis calculs d'essais ERSEC en grappe 6 X 6.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

Utilisation des résultats expérimentaux des boucles ERSAC et OMEGA situées au Service des Transferts Thermiques du Centre d'Etudes Nucléaires de Grenoble (fiches 145-1-04 et 145-1-05).

4. ETAT D'AVANCEMENT

4.1 Avancement à ce jour : avec code RELAP 4

Phase dépressurisation : OMEGA

Configuration tube.

Travaux sur essais de la première campagne achevés; essais adiabatique et avec puissance.
Balayage d'options sur les versions successives du code RELAP, Mod 3 et Mod 5.

Etudes intermédiaires : modèle de glissement (Mod 5 et 6), représentation géométrique de l'élément chauffant (plaque équivalente/ tube réel), corrélations (selon version : Mod 5/ Mod 6).

Essais 2ème campagne : travail d'analyse commencé.

Configuration grappe.

1ère campagne .... Fin 80, début 81

Analyse de 2 essais représentatifs .
Etudes paramétriques .
Comparaison expérience-calcul,
détermination des conditions optimales
d'utlisation de RELAP 4.

2ème campagne

. Calculs préliminaires .... ler semestre 81
pour un ensemble de situations figurant dans la grille d'essais (et différentes des configurations de la 1ère campagne), selon la taille et l'emplacement des brèches, et la puissance.

. Calculs relatifs à des essais effectivement réalisés .... 2ème sem 81, 82

Optionspour calculs RELAP : en règle générale figées.

Ensemble étendu de configurations expérimentales :
(toujours brèches doubles)

section totale de brèche grande (500 mm^2) ou moyenne (190 mm^2)

rapport variable des sections de brèches situées en aval ou en amont de la section chauffante (par rapport au sens initial de l'écoulement),

flux variable, nul, Ø = 60 W/cm^2(flux moyen dans un coeur de réacteur),Ø = 100 W/cm^2 (flux maximum).

Phase renoyage : ERSEC

Calcul et interprétation du problème standard OCDE N° 7 (essai ERSEC N° 1857) à faible débit imposé 5 g/cm^2 s.

Calcul de l'essai ERSEC N° 2034 à fort débit imposé : 12 g/cm^2 s.

Etude paramétrique de différentes options du code : coefficient d'échange en écoulement dispersé, corrélation d'entraînement.

Calcul d'essais grappe à pression 3 bars, 1 bar et 6 bars.

4.2 Avancement à ce jour : avec le code TRAC

Phase dépressurisation : OMEGA

Calcul de deux essais tube : TUBA 2 - adiabatique avec brèche aval totale

TUBA 6 - flux de 125W/cm^2 et brèche aval totale

Calcul de deux essais grappe: MEGA 9 - Brèche aval maximum

MEGA 7 - Brèche double maximum

Phase renoyage : ERSEC-grappe

Quatre essais ont été calculés :

	ΔT sat°C	P (bars)	
2517 ter	20	3	
2625	50	3	
2624 bis	50	3	débit oscillant
2505	20	1	

Ø = 4,5 W/cm^2 débit massique 6 g/cm^2 s

Aucun nouveau calcul TRAC n'a été réalisé en 82

4.3 Principaux résultats : code RELAP 4

Phase dépressurisation : OMEGA

Configuration tube .

De l'analyse d'essais de la 1ère campagne (examen de p , T_{fl}, M, G, T_{paroi}) un ensemble d'options pour l'hydraulique s'est dégagé :

- débit critique : Henry-Fauske/HEM
- inhomogénéité : glissement (avec découpage assez fin de la région la plus éloignée de la brèche et de grande section- où la vitesse est faible).

Une étude de sensibilité au découpage pour le modèle de glissement a montré que de bons bilans de masse et une propagation d'un front de taux de vide (niveau) pouvaient être obtenus.

Une première comparaison des options sur les corrélations d'échange thermique semble montrer que le nouveau jeu de Mod 6 apporte des progrès surtout pour la prédiction des températures maximales de l'élément chauffant.

Configuration grappe

- Analyse d'essais de la 1ère campagne.

Le code RELAP 4 Mod 6 permet d'obtenir de bons résultats avec les modèles HEM (pour le débit critique) et de glissement (pour l'inhomogénéité de l'écoulement), et avec la modélisation géométrique simplifiée qu'il permet. Des écarts sur les temps de montée en température calculés et mesurés sont apparus.

Il est donc nécessaire de tenir compte de la perturbation dûe à la prise de mesure des températures de paroi (constante de temps des thermocouples) - soit en corrigeant les mesures expérimentales, soit en donnant une description approximative de la jonction paroi-thermocouple pour le calcul.

- Calculs 2ème campagne.

Les essais de la deuxième campagne ont fait l'objet d'une comparaison expérience/calcul; il apparaît que, globalement, ils ont été décrits de façon satisfaisante; cependant, la qualité des résultats varie avec les configurations géométriques. La bonne tendance des conclusions tirées des calculs " 1ère campagne" se trouve confirmée - particulièrement si,se plaçant dans une optique de calculs "en aveugle" on cherche une bonne qualité moyenne. Mais un meilleur choix d'options peut être réalisé cas par cas par l'analyse à posteriori, en fonction des différences de configuration.

Phase renoyage : ERSEC

L'interprétation de nos calculs et la comparaison des résultats des différents participants au problème standard OCDE ont montré que si la température maximale de gaine est relativement bien prédite (+ 10 %), par contre, les temps de trempe sont beaucoup trop faibles surtout pour la partie inférieure du tube. De plus certains paramètres physiques tels que la perte de pression le long du canal ou la température de vapeur en sortie sont très mal calculés, ce qui s'explique par le fait que la physique des phénomènes de renoyage est très mal représentée dans les modèles utilisés dans le code.

Une erreur de code a été mise en évidence dans la méthode numérique implicite de couplage des modèles d'entraînement liquide et de surchauffe de vapeur.

4.4 Principaux résultats : code TRAC

Phase dépressurisation : OMEGA

Tube - Dépressurisation trop rapide, brèche 50 % donne des résultats assez bons

Grappe - MEGA 9 dépressurisation assez bonne jusqu'à t = 10 secs

MEGA 7 trop rapide

Phase renoyage : ERSEC

- Vitesse de trempe généralement nettement inférieure à la valeur expérimentale. La corrélation de température minimum d'ébullition pour film en est partiellement responsable.
- Température maximum souvent atteinte trop tôt il semble que cela soit dû en grande partie à une surestimation des échanges paroi-vapeur à taux de vide élevé.
- Les températures de paroi calculées sont beaucoup plus irrégulières que celles tirées des mesures . Ceci est dû à la fois aux fluctuations importantes de taux de vide et de vitesse et aussi aux changements brutaux des coefficients d'échange paroi-vapeur et paroi-liquide.
- Le calcul est très couteux le pas de temps étant limité par la condition de Courant à environ 10^{-2} sec. 150 secondes de temps réel ont nécessité 1H15 de CDC 7600.

5. PROCHAINES ETAPES

5.1 Code RELAP 4 Mod 6

Phase dépressurisation : OMEGA

Configuration du tube, 2ème campagne d'essais

Le travail d'analyse sera moins systématique du point de vue des options du code (puisqu'un jeu d'options optimum a déjà été dégagé); en revanche il le sera davantage par rapport à l'ensemble des différentes configurations d'essais (variation du volume et de la forme des "plenums", de la taille et de l'emplacement de la brèche, de la puissance). Cette plus grande gamme d'essais étudiés pourra avoir une répercussion sur le jeu d'options antérieur.

Configuration grappe

L'analyse est pratiquement achevée; une note est en cours de rédaction. Toutefois, les mesures de températures de paroi présentant un défaut systématique (effet d'inertie des thermocouples), un complément à l'analyse pourra être effectué si on vient à disposer d'un bon modèle pour améliorer le dépouillement de ces mesures.

5.2 Code TRAC

Des calculs avec la version TRAC PF 1 sont envisagés. En ce qui concerne OMEGA l'intérêt de TRAC PF 1 réside dans la modélisation de la brèche et l'utilisation du système à 6 équations. En ce qui concerne ERSEC cette nouvelle version pourrait apporter une diminution sensible du coût calcul.

Il est prévu de reprendre les calculs ERSEC avec TRAC PF 1. Ces derniers seront réalisés avec deux modèles différents : le modèle 3-D et le modèle 1-D. Le modèle 3-D étant très proche de TRAC PD 2 on s'attend à ce que les résultats soient proches de ceux déjà obtenus. Par contre le modèle 1-D devrait fournir des résultats nettement différents; en particulier on espère que le coût d'un calcul sera très réduit.

6. RELATIONS AVEC D'AUTRES ETUDES

- Calcul des accidents de refroidissement des réacteurs à eau .... (fiche 145-1-01)
- Développement du code de 2ème génération CATHARE pour l'étude de l'accident de perte de caloporteur dans les réacteurs à eau pressurisée : Programme général coordonné .... (fiche 145-1-02)
- Thermohydraulique du LOCA : Etude expérimentale de la phase dépressurisation des accidents de perte de réfrigérant des réacteurs à eau. Programme OMEGA..... (fiche 145-1-04)
- Thermohydraulique du LOCA: étude expérimentale de la phase renoyage des accidents de perte de réfrigérant des réacteurs à eau. Programme ERSEC ....
- Evaluation-vérification RELAP-TRAC. Calculs SEMISCALE - LOFT - PKL .... (fiche 145-1-13)
- Evaluation-vérification : Programme LOBI .... (fiche 145-1-14)

7. DOCUMENTS DE REFERENCE

Analyse par le code RELAP 4 Mod 5 d'un essai adiabatique OMEGA 1ère campagne
M. MEZZA - Note technique SEAREL 79/10

Analyse par le code RELAP 4 Mod 5 d'un essai OMEGA avec puissance
M. MEZZA - Note technique SEAREL 79/13

Le modèle de glissement du code RELAP 4; sensibilité au découpage
M. MEZZA - Note technique SEAREL 80/23

Le traitement de la thermique en dépressurisation dans le code RELAP 4
M. MEZZA - Note technique SEAREL 80/25

Evaluation du code RELAP 4 Mod 6 pour les calculs d'expériences et d'accident de réacteurs.
(communications françaises à la réunion OCDE Paris 18/20.6.80)
B. ADROGUER, M. MEZZA, R. POCHARD, R. SENEMEAUD, N. TELLIER
Note technique SEAREL 80/26

Calculs d'essais OMEGA, configuration GRAPPE (1ère campagne, 1977) avec le code RELAP 4 Mod 6
M. MEZZA - Note technique SEAREL à paraître

Calçuls préliminaires d'essais OMEGA, configuration GRAPPE (2ème campagne, 1981) avec le code RELAP 4 Mod 6.
M. MEZZA - Note technique SEAREL 81/37

Calculs d'essais OMEGA, configuration GRAPPE (2ème campagne, 1981) avec le code RELAP 4 Mod 6
M. MEZZA - Note technique SEAREL à paraître

Analyse d'une expérience de renoyage ERSEC à l'aide du code RELAP 4 Mod 6
N. TELLIER, M. PICHON - Note technique SEAREL 79/06

Draft Comparison Report on OECD-CSNI. LOCA standard problem N° 17 : Analysis of a reflooding experiment.
R. DERUAZ, N. TELLIER - CSNI report N° 55 (1979)

"Qualification de TRAC - Rapport d'avancement au 1.3.81
B. NOEL Rapport TT N° 167

Calculs de renoyage sur les expériences ERSEC GRAPPES avec le code TRAC PD 2
B. NOEL, A. FORGE, rapport TT/SEMTH/ 82 - 3 SEAREL 82/66

	CLASSIFICATION : 1.2
TITLE (ORIGINAL LANGUAGE) : Studio teorico-sperimentale della termoidraulica durante la refrigerazione di emergenza per allagamento dal basso.	COUNTRY : ITALY
	SPONSOR : Politecnico di Torino
TITLE (ENGLISH LANGUAGE) : Theoretical and Experimental Study of Bottom Flooding Thermal-Hydraulics.	ORGANISATION : (•) Politecnico di Torino
	PROJECT LEADER : M. De Salve
INITIATED : January 1976 / COMPLETED :	SCIENTISTS : B. Panella
STATUS : In progress / LAST UPDATING : April 1982	

(•) 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 the reflooding.

2. Particular objectives.
To find, for different linear thermal capacities and lenght 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 following parameters are measured: wall temperatures along the test section, flooding rate, exit fluid temperature and void fraction (by an impedance probe), liquid temperature and flow rate in a liquid collector, pressure drop across the test section, pressure in the bottom and top plenum, the liquid carried-over by the steam. Investigations are restricted to atmospheric pressure, small and high flooding rates, high initial wall temperatures ($T \sim 800°C$) and high and small inlet subcooling. Bottom flooding tests have been performed in a tubular test section with variable inlet subcooling, heated power, flow rate for two thikness and lenghts.

4. Progress to date.
Many tests in both rod annular and tubular test sections have been performed (I.D.8mm 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 s; inlet subcooling from 80 to 10°C. For the tubular

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TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Theoretical and Experimental Study of Bottom Flooding Thermal-Hydraulics.	1.2

test section parametric studies have performed for I.D.= 10 mm; O.D.= 12;13 mm and lenghts of 1 m and 1.5 m.
The experimental results show the dependency of the quenching temperature and rewetting front velocity from the rate, initial wall temperature, inlet subcooling, the linear thermal power and axial position.

.- Next steps.
Theoretical models have been analysed and an attempt to evaluate the heat transfer coefficients in all regions during rewetting is going on. The instrumentation will be improved particularly as regârds the superheated vapor temperature, the void fraction at the exit and the liquid and steam flow rate measurements. Tests with an Inconel test section will be performed and also the rewetting with constant head flow rate will be investigated.

.- 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 reactyor safety, Brussels, October, 15-19, 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.
- M.De Salve, B.Panella: "Un modello per il calcolo termoidraulico nel ribagnamento di un tubo per allagamento dal basso". PT IN 111, luglio 1979, Politecnico di Torino.
- M.De Salve, B.Panella: "Thermal-Hydraulic of bottom reflooding in tubes with different thickness and length". Paper presented at the topical meeting on "Nuclear Reactor Thermal-Hydraulics" Saratoga, New York, October 1980.

(Politecnico di Torino, Istituto di Fisica Tecnica ed Impianti Nucleari,
Corso Duca degli Abruzzi, 24 - 10129 Torino)

'. Additional information.
Budget: 15 million Lit. Personnel involved: 4 persons.

	CLASSIFICATION : 1.1.2/1.2/1.3/7.2
TITLE (ORIGINAL LANGUAGE) : ATTIVITA' DI RICERCA FINALIZZATA NEL CAMPO DELLA TERMOIDRAULICA DEI TRANSITORI CONSEGUENTI A LOCA PER IL REATTORE PWR ENEL UNIFICATO	COUNTRY : ITALY SPONSOR : ENEL
TITLE (ENGLISH LANGUAGE) : FYNALIZED RESEARCH ACTIVITY IN THE FIELD OF THERMO-HYDRAULIC TRANSIENTS FOLLOWING LOCAS FOR THE ENEL UNIFIED PROJECT PWR REACTOR	ORGANISATION : ENEL PROJECT LEADER : G. TREBBI
INITIATED : 1980 / COMPLETED :	SCIENTISTS : L. BELLA, F. DONATINI
STATUS : IN PROGRESS / LAST UPDATING : May 1982	

1. General aim

The research has the purpose to develop an extensive know-how about the thermal-hydraulic aspects of LOCA'S transients for the ENEL reference PWR plant. The following aspects will be investigated:

a) Primary system thermal-hydraulic behaviour
b) Fuel assembly thermal behaviour (refill - reflood phase)
c) Containment behaviour/ECCS performance.

In the next years such three aspects could be splitted up in three distinct research activities.

2. Particular objectives

Qualification of RELAP 4 codes chain for LOCA'S analyses applied to PWR ENEL reactor (EM/Best estimate calculations), extended to the refill-reflood transients. The analyses will cover a great range of breaks (large DE-small) and ECCS performance, under various hypotheses. Comparisons with experimental results will be made with OECD/CSNI Standard Problems program.

The containment will be analized with CONTEMPT LT and COMPARE MOD1 codes and the effect of various emergency systems will be investigated.

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	CLASSIFICATION : 1.1.2, 1.1.4., 1.2
TITLE (ORIGINAL LANGUAGE) : Analisi dei transitori termo-idraulici nel circuito primario dei reattori ad acqua leggera a seguito di LOCA.	COUNTRY : ITALY SPONSOR : ENEA (formerly CNEN)
TITLE (ENGLISH LANGUAGE) : Analysis of thermo-hydraulic transients in LWR following LOCA	ORGANISATION : University of Pisa PROJECT LEADER : Marino MAZZINI

INITIATED : 1979	COMPLETED : 1987	SCIENTISTS : N.CERULLO F.ORIOLO, U.ROSA, P.VIGNI, F.D'AURIA, R.BOVALINI, G.GALASSI, M.CARCASSI.
STATUS : in progress	LAST UPDATING : January 1982	

1. General aim

The research has the purpose of setting-up a best estimate model for the analysis of thermal and hydraulic transients following LOCA in light water reactors, achieving a better knowledge of major phenomena occurring during blowdown, refill and reflood phases.

Particular objectives

Qualification of "Best Estimate" computer codes in the analysis and design of experimental tests, by their applications to pre-test calculations and post-test analysis of experimental results obtained in the experimental facilities of Scalbatraio Center (University of Pisa) or available from international research programs (LOBI-CCR-Ispra; LOFT and Semiscale, with reference to CSNI International Standard Problems).

2. Facilities

The experimental small scale facilities PIPER and SOPRE of the Scalbatraio Center, University of Pisa.
A new experimental facility, named PIPER-ONE is in design stage; it simulates the thermal-hydraulic behaviour of BWR plants during small LOCA.

		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: ENEA 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	SCIENTISTS: F. ORIOLO, G. MEI, G. CECCARELLI, G. BITETTI
STATUS: in progress	LAST UPDATING: May 1979	

1983-06-29		CLASSIFICATION : 1.1.2/1.2
TITLE (ORIGINAL LANGUAGE) : Termohydraulisk haverianalys (THYHAV-I)		COUNTRY : Sweden SPONSOR : Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE) : Thermal-hydraulic Accident Analysis		ORGANISATION : STUDSVIK ENERGITEKNIK AB PROJECT LEADER : O Sandervåg
INITIATED : 82-01-01	COMPLETED : 83-11-01	SCIENTISTS : R Persson, Ö Rosdahl, A Sjöberg, J Eriksson, K-A Flygvall
STATUS : In progress	LAST UPDATING : June 1983	

General aim

To provide and maintain advanced thermal-hydraulic computer codes for independent LOCA analysis of Swedish nuclear power reactors, and to perform research on hypothetic and real accident sequences.

Particular objectives

Maintenance of RELAP5 and development of a program package for LOCA analysis of BWRs, analysis of the Oskarshamn 2 reactor (large breaks) and Ringhals 2 reactor (small breaks), qualitative code assessment using FIX-II experiments and international standard problems, developmental assessment of heat transfer correlation, investigation of modelling requirements for prediction of system behaviour during small breaks.

Project status

Most activities listed are in progress. Planned efforts on O2 calculations, standard problems and small break modelling requirements are near completion.

Application and asessment of RELAP5/Mod1 reveal needs for improvements in some of the physical models. It has been found that the code predicts well an overall system behaviour during small breaks with a relatively small number of nodes.

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	CLASSIFICATION: 1.2
TITLE (ORIGINAL LANGUAGE): Undersökning av behovet av nya härdstrilprov. Etapp 1	COUNTRY: Sweden SPONSOR: Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE): Investigation of the need for new core spray tests. Phase 1	ORGANISATION: ASEA-ATOM PROJECT LEADER: David Burns

INITIATED: 1981-09-24	COMPLETED: 1983-04-29	SCIENTISTS: Hernan Tinoco
STATUS: Completed	LAST UPDATING: June 1983	

1. General Aim.

   To determine the distribution of ECCS spray coolant under all reactor conditions.

2. Particular Objectives

   1. Study of conditions in core, upper plenum and bypass after spray initiation.
   2. How is spray distribution affected ?
   3. Effect of reduced spray flow on core cooling.
   4. Complete documentation of previous tests to verify core spray function.
   5. Decision on the need for further tests.

3. Experimental facilities and programme

   The previous facility is dismantled. Any new facility will depend on the outcome of Phase 1.

4. Project Status

   1. Completed
   2. No further tests required. Existing spray shown to be acceptable with good margin.

5. Relation to other Projects and Codes

No other related work.

6. Reference Documents

1. AA PM RCC 81-119.
   Projektbeskrivning. Projekt DOWSE: Etapp 1.
   Undersökning av behovet av nya härdstrilprov (Project description).

2. AA PM KPA 82-81 (H. Tinoco).
   Model for description of spray cone with condensation.

3. AA PM KPA 83-05 (H. Tinoco).
   Study of the conditions under which good spray distribution is required.

4. AA PM KPA 83-01 (D. Burns).
   Final report on the question of renewed spray distribution testing for external pump reactors.

7. Degree of Availability

ASEA-ATOM reserves the right to deem which reports should be regarded as confidential. The sponsor may use the non-confidential reports as he thinks fit, though on a non-commercial basis.

Contact person: David Burns
AB ASEA-ATOM
BOX 53
721 04 Västerås
Sweden

	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:
INITIATED: 1974 | COMPLETED:	SCIENTISTS: AA DEBENHAM, S BOOTH
STATUS: CONTINUING | LAST UPDATING: SEPTEMBER 1980	

PAGE NOT USED

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, 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	Completed	Project Leaders
Status Continuing	Last updating	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: 1.1.1,1.1.2,1.1.4,1.2,7.
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: ENEA 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	SCIENTISTS: F. ORIOLO, G. MEI, G. CECCARELLI, G. BITETTI
STATUS: in progress	LAST UPDATING: May 1979	

1.3. BEHAVIOUR AND INFLUENCE OF FUEL-ELEMENTS SPECIFICALLY RELATED TO BLOWDOWN AND ECCS

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	1.3	150 448
Vorhaben/Project Title Untersuchung des Wärmedurchgangs durch den Spalt zwischen Brennstoff und Hüllrohr bei LWR-Brennstäben Heat conductance of the fuel-to-cladding gap of LWR rods, Irradiation tests		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor KRAFTWERK UNION AG Reaktortechnik B 211, Erlangen
Arbeitsbeginn/Initiated 01.06.80	Arbeitsende/Completed 30.09.82 *)	Leiter des Vorhabens/Project Leader M. Gärtner
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating 31.12.80	Bewilligte Mittel/Funds

*) Extension requested

1. General Aim

The temperature of the fuel, and hence the energy stored in it, is the most significant parameter governing rod behaviour under hypothetical loss-of-coolant accident conditions.

2. Particular Objectives

The fuel temperature is not directly measurable in a power reactor and therefore has to be calculated by means of physical models. These models are based on measured data which exhibit a wide scatter range and may thus only be used in a conservative manner. The irradiation experiments in the FRJ-2, to be performed in cooperation with KFA Jülich, will provide data exactly defined by an improved measuring technology and a definite parameter range. These measured data shall permit an improvement of the models and thus a precise estimate of safety margins which are used for fuel rod design with regard to emergency cooling criteria today.

3. Research Program

3.1 Determination of the heat conductance for the gap between the fuel and cladding.

3.2 Investigation of fuel relocation in its various appearances (initial, reversible, irreversible relocation) for varying burnups and powers.

4. Experimental Facilities

Boiling water loop LV9 in the FRJ-2 of KFA Jülich with filter station LV47 for fuel rod gas.

5. Progress to Date

Re. 3.1 Two experiments have been performed to investigate axial gas separation in the fuel rod (test fuel rods TT70 and TT 71). Detailed evaluation of the measured data has been carried out. On account of both experiments being stopped very prematurely, agreement has been reached with the conductor of the experiments, KFA Jülich, that these irradiations should be repeated partly at least. A start has been made on planning the optimum design and irradiation level of the test fuel rods to be used for this purpose. The first experiment, designed to investigate the effective gap size and the influence of fission gas release on gap conductance, has begun successfully with the irradiation of test fuel rod TT72 for a complete FRJ 2 cycle. The fuel rod was run up to its maximum LHGR of 300 W/cm (at the thermocouple locations) following a detailed start-up program. At a later stage in the reactor cycle, a first gas change was implemented in the fuel rod (He→Ne→He). Details of the experimental procedure to be used for the second irradiation cycle have been established.

6. Results

Re. 3.1 In the experiments to detect axial gas separation, there were traces of such separation under the following special conditions:

- large effective gap size (fabricated size 250 µm)
- high xenon content in fillter gas (Xe/He = 40/60 % by volume)
- elevated initial internal pressure (p_i = 30 bar)
- relatively low LHGR (q_{max} < 75 W/cm)

It is necessary to check these results by repeating the experiments as planned.

In the experiments relating to gap conductance performed on test fuel rod TT72, the instruments so far indicate perfect operational behaviour. Fuel centre temperature measurements at low fuel rod pressures using various inert gases to determine their gas extrapolation lengths indicate that temperatures are only significantly dependent on pressure in a pressure range below 10 bar, this dependence itself being more striking with helium than with neon.

7. Next Steps

Re. 3.1 - Further irradiation of test fuel rod TT72 (measurement of fuel centre temperature as a function of filler gas type, filler gas pressure, fabricated gap

size and burn-up)
- Preparation and planning for irradiation of test fuel rod TT 73
- Irradiation of test fuel rod TT73 (specific areas of investigation as for test fuel rod TT72, additional test parameter: pellet surface roughness)
- Evaluation and summarization of experiments in gap conductance on test fuel rods TT72 and TT73
- Preparation for repeating the prematurely discontinued experiments on test fuel rods TT70 and TT71

8. Relation with Other Projects

-

9. References

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10. Degree of Availability

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Berichtszeitraum/Period 01.01.82 - 31.12.82	Klassifikation/Classification 1.3	Kennzeichen/Project Number 06.01.04/19A, 20A, 21A, 22A
Vorhaben/Project Title Development and Verification of a Code-System of Fuel Behavior at Loss of Coolant Accidents Entwicklung und Verifizierung eines Code-Systems zum Brennstabverhalten bei Reaktorstörfällen		Land/Country (PNS 4231.1/3/4/5) Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe GmbH (KfK) Projekt Nukleare Sicherheit (PNS) IRE, INR, IRB
Arbeitsbeginn/Initiated 01.01.1973	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader Dr.Borgwaldt, Dr.Meyder, DI.Malang
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

The aim of this project is the development of a code system SSYST for the transient behavior of Zircaloy-clad fuel rods under LOCA (loss of coolant accident) and SFD (severe fuel damage) conditions. Connected with this activity are the implementation and verification of thermohydraulics-codes for the analysis of the whole system as well as for assessing post-LOCA long-term coolability.

2. Particular Objectives

Development, verification and maintenance of the modular code system SSYST. Modeliing relates to the interaction between the heat transport in the rod, the heat transfer in the gap, the deformation of the cladding, the rod internal pressure and the thermohydraulics phenomena in the subchannel.

Verification and further development of thermohydraulics codes for analysis of the primary system and of the single channels.

3. Research Program

Development work comprises the following areas:

- Development and verification of SSYST modules for single rod analysis.
- Investigation of interactions between the rods and behavior of rod bundles.
- Implementation and verification of advanced primary system codes.
- Verification of some two-phase bundle codes.

4. Computer Codes

Code system SSYST, COMETHE III J, RELAP4/MOD6, RELAP5/MOD1, REFLUX.

5. Work performed

SSYST-2:

On request by NEA Data Bank, some support was given in the implementation, final tests and preparation of the transmittal tape for the released version of SSYST-2. This included the demonstation of the coupling of SSYST-2 with RELAP4/MOD6 as well as with another primary systems code, NORCOOL.

Activities to increase code reliability by introducing data consistency checks into all general purpose and the majority of physical modules of SSYST have been continued. GRS, under subcontract of PNS,made detailed calculations on the experiments REBEKA-3 (KfK) and LOC-3 (PBF, Idaho Falls) as a contribution to the verification of SSYST-2.

SSYST-3:

The new code version SSYST-3, with additional application modules and a streamlined, restructured systems kernel, has been completed. Part of this new version is the preprocessor code PREPRO for complete checks of large SSYST-3 input files in standard or free format. SSYST-3 has been prepared to simplify the future inclusion of additional physical models and a more general execution sequence control.

The documentation of the creep rupture model NORA2 is completed. This model describes creep and rupture behavior of Zircaloy tubes at high temperatures assuming a cylindrical geometry. NORA2 is now also integrated into the SSYST modules STADEF and ZIRKOX, so that clad deformations can be determined quickly and precisely.

Work to integrate the flooding code REFLUX/GRS has been continued.

SFD-modeling:

The MULTRAN code completed in the last year has been fully integrated into SSYST. It models the partitioning of the oxide layer in a cubic and a tetragonal part as observed at temperatures above 1850 K.

A special version of the SIMTRAN code has been developed which calculates the fuel temperature in a case where a fuel rod or a fuel rod simulator is heated up to the melting point of Zircaloy.

To improve understanding of the fuel rod behavior under SFD-conditions and to quantify single effects, calculations of some ESSI experiments performed in the NIELS melting facility were started.

Thermohydraulics codes:

The KfK-version of RELAP4/MOD6/004 was transmitted to GKSS and NEA Data Bank.

Cycle 14 of RELAP5/MOD1 has been put in operation in CYBERNET. That version was used to calculate the CSNI Standard Problem 13 (LOFT L2-5).

The FACOM-version of RELAP5/MOD1/001 that has been taken over from JAERI did not work on IBM-system at KfK. To get a running IBM Code extensive adaption work and checkout testes have to be done in close cooperation with GRS. In order to reach the level of the latest CDC-version of RELAP5/MOD1 the original code updates have been incorporated into the IBM-version.

6. Results obtained

Verification of SSYST-2:

The verification calculations of GRS using SSYST-2 yielded a good agreement with the measured values. The REFLOS module proved to be suited to simulate the thermohydraulics of the REBEKA-3 experiment. The results obtained lie within the scatter band of the

measured values.
The computing times required by SSYST-2 must be considered as extremely favorable; compared to SSYST-1 they are only 32%.
SSYST-3:
A working version of SSYST-3 has been transmitted to GRS, Köln, for field testing. The final version of the preprocessor code PREPRO has the following capabilities: Input for loosely coupled modules is only scanned and passed. Input for all regular fully integrated SSYST-3 modules is accepted in formated, unformated or symbolic form and is converted to standard format. In addition, the data are checked for formal correctness, and the list of required modules for executing the input is evaluated.
The failure criterium of NORA2 is based on the total circumferential elongation (TCE) in the plaine of failure, usually reported for tube burst tests. This failure strain is a function of strain rate and temperature. For the pure phases of Zircaloy α and β TCE increases with increasing strain rate whereas in the α / β two phase region TCE decreases with increasing strain rate. Considering however the local failure strain of the probes, it is found that the local failure strain is independent of the strain rate, in a first approximation. Even the dependence on temperature becomes simpler. This local failure strain is about 200% to 300%. Therefore the sensitivity of TCE with respect to strain rate and temperature must be a consequence of local variations of stress and temperature along the circumference of the probe.
This interpretation can be confirmed with the SSYST module AZI. AZI describes a rod slice in r-Θ geometry. The assumptions for the calculations are: creep law according to NORA2, modification of failure strain as indicated above and initial veriations in wall thickness (+/- 10 μ) and in temperature (+/- 2K). Such variations are always present for real probes. According to this, also in the model TCE is controlled by a localisation of strain caused by imperfections of the probes.
The computer code for modelling the 3D thermoelastic and plastic deformation of a fuel rod cladding between two grid spacers has been successfully tested for the thermoelastic solutions and also for plastic deformations. It was found that the explicit integration of plastic strains is too time consuming. Improved integration techniques are tested now. The documentation of 3D deformation model has begun.
SFD-modeling:
The results of the MULTRAN code agree well with the experimental results as reported by Cathcart and Leistikow (1000 - 1500^{o}C), Urbanic and Leistikow (1550-1800^{o}C), and Baker (above 1940^{o}C).

- 4 -

01.01.82 - 31.12.82

06.01.04/19A, 20A, 21A, 22A
(PNS 4231.1/3/4/5)

Calculations with the special version of the SIMTRAN code showed that the temperature difference in the cladding between surface and oxide/metal interface can reach values up to 50 K if a rod is heated up to the melting point of Zircaloy.
The calculation of the thermal behavior of the fuel rod simulators of the ESSI experiments showed that the steam flow between the Zircaloy tube and the insulation had a strong influence on rod cooling. Experiments performed later in the ESSI test series with an insulation directly attached to the Zircaloy tube confirmed this finding.

Thermohydraulics codes:

The IBM-version of RELAP5/MOD1 has been updated up to cycle 6 and tested.

7. Next Steps

Full integration of REFLUX/GRS as a SSYST module, communicating with other physics modules. Implementation of a generalized execution sequence control in SSYST-3.
Analysis of the experimental data on the interaction between fuel and Zircaloy cladding.
Modeling of the interaction between fuel and Zircaloy cladding with respect to the oxygen profile in the cladding wall.
Starting from 1983 work related to RELAP5 and other advanced thermohydraulics codes will be continued within a new sub-project 06.01.21.

8. Relations with other Projects

This project is part of the general project PNS 06.01 of KfK. Regarding model development and verification it relies on the other parts of this project.

9. Literature

W. Gulden, R. Meyder, H. Borgwaldt:
SSYST-A Code System to Analyze LWR Fuel Rod Behavior under Accident Conditions; Newsletter of the NEA Data Bank No. 28 (September 1982) pp. 73-102.

S. Raff:
Entwicklung eines Deformations- und Versorgungsmodells für Zircaloy im Hochtemperaturbereich zur Anwendung bei Kühlmittelverluststörfalluntersuchungen an Leichtwasserreaktoren, KfK-3184 (November 1982).

10. Availability of the Reports

No restriction on the literature indicated.
Primary reports upon request.

Berichtszeitraum/Period 01.01.82 - 31.12.82	Klassifikation/Classification 1.3	Kennzeichen/Project Number 06.01.06/20A (PNS 4235.1)
Vorhaben/Project Title Untersuchungen zum mechanischen Verhalten von Zircaloy-Hüllmaterial Investigations of the mechanical behavior of Zircaloy cladding material under transient conditions		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe mbH (KfK) Projekt Nukleare Sicherheit (PNS), IMF II
Arbeitsbeginn/Initiated 01.01.1973	Arbeitsende/Completed 31.12.1982	Leiter des Vorhabens/Project Leader Prof. Bocek
Stand der Arbeiten/Status completed	Berichtsdatum/Last Updating	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 Destructive postexamination of in-pile ballooned rods.

3.4 Preparation of verified burst criterions for Zircaloy-4 cladding.

4. Experimental Facilities

Ref. 3.1 Tensile testing will be performed in an INSTRON closed-loop machine.

Ref. 3.2 For burst tests in vacuum tubes will be pressurized in a radiant furnace (TUBA). For integral experiments (steam environment) cladding with internal heaters will be used (FABIOLA).

5. Progress to Date

Ref. 3.1 Completion of creep-tests in a close graduation of temperatures.

Ref. 3.2.2 Completion of FABIOLA-Burst-experiments to verify the developed burst criterions.

Ref. 3.3 Completion of destructive post test examination of irradiated in-pile deformed tubes of the G-serie.

Ref. 3.4 Comparison of creep burst experiments in respect to burst time and burst strain.

6. Results

Ref. 3.1 The creep test serie is needed to know better - in a close graduation of temperatures - the temperature dependence of the stress exponent and the stress and temperature dependence of the apparent activation energy of creep. By that means the life fraction rule (LFR) will be on a level to be installed in the international code SSYST to calculate burst times and burst temperatures during emergency accidents of pressurized light water reactors.

Ref. 3.2.2 The burst data now available from FABIOLA-experiments up to a heating rate of 16 K/sec. had been used on one side to increase the accuracy of the constants to calculate burst strains by means of the modified-Monkman-Grant relationship (MMG). On the other side they had been used to verify in respect to burst temperature and burst time the LFR and to burst strain the MMG very successful [1,2,3].

Ref. 3.3 The evaluation of destructive post test examination of the G-serie (35.000 MWd/t preirradiation and in-pile deformation) had been completed [4]. Differences in mechanical properties compared to test results with cladding containing fresh fuel of A- and B-series and to results from tests with preirradiated cladding (F-serie) could not be detected.

Ref. 3.4 The calculation of burst time and burst strain from creep burst tests of the REBEKA-facility (IRB, KfK) with the Larson-Miller extrapolation method led for the burst time to acceptable results. The burst strain could be calculated in analogy to the modified Larson-Miller extrapolation method with encouraging results [5].

7. Next Steps

After answering all important questions given on the above question this part of the project is completed.

8. Relation to other Projects

PNS 4235.2 (06.01.06/21A)

PNS 4235.3 (06.01.06/22A)

PNS 4235.4 (06.01.06/23A)

9. References

[1] M. Boček
"Lifetime and Failure Strain Prediction for Material Subjected to Non-stationary Tensile Loading Conditions; Applications to Zircaloy-4"
ASTM STP 754 (1982) 329

[2] M. Boček, C. Petersen and L. Schmidt
"The Lifetime and Failure Strain Prediction of Zircaloy-4 Cladding Loaded under LOCA-Similar Temperature Ramp Conditions"
Nuclear Technology (in press)

[3] M. Boček and C. Petersen
"Lebensdauer- und Versagensverformungsvorhersage von Zircaloy-4 bei KVS-ähnlichen Belastungen"
Proceeding of Annual KTG-Meeting '82, (1982), Mannheim, F. R. Germany, 315-318

[4] P. Hofmann et al.
"In-pile-Experimente zum Brennstabverhalten beim Kühlmittelverlustunfall (Versuchsserie G)"
KfK-Report Nr. 3433 (in press)

[5] Unpublished Report

10. Degree of Availability of the Reports

Unrestricted distribution

Berichtszeitraum/Period 01.01.82 - 31.12.82	Klassifikation/Classification 1.3	Kennzeichen/Project Number 06.01.06/21A (PNS 4235.2)
Vorhaben/Project Title Oxidation Behavior of Zircaloy Cladding Tubes during a LOCA of LWR Oxidationsverhalten von Zircaloy-Hüllrohren bei KVS von LWR		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS), IMF II
Arbeitsbeginn/Initiated 01.01.81	Arbeitsende/Completed 31.12.82	Leiter des Vorhabens/Project Leader Dr. S. Leistikow
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

Evaluation of the metallographic hot-cell investigation of fuel rods after in-pile experiments (PNS 4237).

2. Particular Objectives

Oxidation behavior of Zircaloy 4 cladding material during LOCA typical in-pile tests; comparison with out-of-pile results.

3. Research Program

Investigation of the steam oxidation of the external cladding tube surface. Corresponding investigation of the internal surface, which is oxidized in contact with the fuel and by steam access via the rupture opening. Evaluation of the microhardness profiles measured across the cladding tube wall. Microstructural evaluation of the maximum cladding temperature for comparison with temperature measurements and for an estimation of azimuthal and axial temperature differences.

4. Experimental Facilities

See PNS 4237

5. Progress to Date

The results of the investigation of the F series fuel rods (20.000 MWd/t_U) have been published. The evaluation of the G series rods (35.000 MWd/t_U) was finished and compared with the results of low burn-up rods C (2.500 MWd/t_U) and E (8.000 MWd/t_U) and of electrically heated fuel rod simulators.

6. Results

In comparison of all investigated rods and simulators the influences of burn-up and nuclear heating are found to be restricted to only some aspects of minor importance. The oxidation during pre-irradiation can be neglected in comparison with the steam oxidation during the transient. The external oxide scale thickness shows a common scatter band for all rods with the exception of seriously cracked oxide at excessively strained positions and the sporadic occurrence of white patches of thick oxide for some of the pre-irradiated rods. The formation of white oxide is due to early and localized breakaway behaviour induced by the pre-irradiation. Appreciable internal cladding oxidation due to steam access is restricted to the burst position where the steam is mainly consumed. Rods with higher burn-up have formed scales even thicker than the external scales at the vicinity of the burst opening. Altogether the observed oxidation has not determined the ballooning and burst behaviour of the rods.

Compared to the as-received cladding condition a higher microhardness level is reached already after low burn-up. During the LOCA-transient the hardness decreases again and the local values spread to a common scatter band for all rods.

The temperature evaluation of the cladding microstructure has indicated azimuthal differences between 0 and about 100 K of peak temperature for different cross sections of fuel rods and simulators. The method has proved to be reliable and helpful in comparison to the thermocouple measurements during the tests although it has allowed only indirect estimations of burst temperatures.

7. Next Steps

Publication of the results of the G series fuel rods and concluding report.

8. Relation to other Programs

KfK - PNS, KWU, USNRC, JAERI

9. References

1) P. Hofmann, C. Petersen, G. Schanz, H. Zimmermann:
KfK-in-pile Versuche zum Brennstabverhalten in der Aufheizphase eines LOCA; Ergebnisse der zerstörenden Nachuntersuchungen. Bericht über die Versuchsserie F. KfK 3288 (March 1982).

2) Ditto, report about test series G, KfK 3433, in preparation

3) E.H. Karb, M. Prüßmann, L. Sepold, P. Hofmann, G. Schanz:
LWR Fuel Rod Behavior in the FR2 In-pile Tests Simulating the Heatup Phase of a LOCA (Final Report). KfK 3346 (in preparation).

10. Degree of Availability: Open literature

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.82 - 31.12.82	1.3	06.01.07/09A, 10A (PNS 4236)
Vorhaben/Project Title Brennstabverhalten in der Blowdownphase eines Kühlmittelverluststörfalls; Blowdown-Anlage COSIMA FUEL-ROD-BEHAVIOR IN THE BLOWDOWN-PHASE A LOSS-OF-COOLANT ACCIDENT; BLOWDOWN FACILITY COSIMA		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS) IRE/IT
Arbeitsbeginn/Initiated 1.7.1972	Arbeitsende/Completed 1983	Leiter des Vorhabens/Project Leader G.Class, IRE, K.Hain, IT
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

The aim of experiments is to provide information about the failure limits of fuel under incident conditions. The improved knowledge of the fuel element behavior in the blowdown phase of a loss-of-coolant accident is to be used for the development and verification of theoretical models (SSYST).

2. Particular Objectives

The experiments are carried out under blowdown conditions typical of PWR's. The fuel rod behavior is 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 size of 1 and 2F. In each case experiments are carried out at different rod powers and internal pressures.

4. Experimental Facilities, Computer Codes

A loop facility was 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. Work Performed

In the first half of the period under review, the control system of the test facility and the data logging system were revamped so as to match a new computer facility and, consequently, test operation could be resumed only early in the second half of the year. 17 blowdown experiments were run by late 1982, mainly on the pellet relocation problem.

Various experiments (2F LOCA-type experiments with different rod powers and one experiment with LOFT (L2-3)-type transient development) were recalculated with different nodalizations using different options by means of RELAP4/MOD6 (comparison between homogeneous calculation and calculation involving phase separation; application of various DNB correlations and film boiling relations).

The DRUFAN 02 thermodynamics code was implemented on the IBM 3033.

6. Results

6.1 Experiments

The experiments on the pellet relocation problem are intended to represent the influence of the void volume in the SIM fuel rod simulator on relocation of the Al_2O_3 annular pellets.

Two series of tests were run for this purpose:

(a) with enlarged internal gap (graphite heater - annular pellet)

(b) with enlarged external gap (annular pellet - cladding tube).

Test series (a) shows the expected slight effect of the enlarge internal gap, compared with test series (b) and a SSYST sensitivity study. Compared with the sensitivity study referred to above, the enlarged external gap exhibited the expected major influence on peak cladding temperatures in the course of a number of experiments. With initial gap widths of

- 50 μm internal
- 100 μm external,

peak cladding temperatures dropped by 250 K because of pellet relocations in the course of five blowdown experiments. This problem is to be studied in a number of further SSYST calculations and another series of experiments.

6.2 Recalculations by means of RELAP4/MOD6

In the recalculations conducted by means of RELAP4/MOD6 of the experiments mentioned above, a parameter investigation of the following influences was emphasized in particular:

- Nodalization
- data set with 24 volumes, 27 junctions, 39 heat slabs
- data set with 13 volumes, 16 junctions, 21 heat slabs;
- phase separation model
- homogeneous calculation
- phase separation along the measurement section
- phase separation along the plenums.

It was found that the combination of "fine nodalization" and "phase separation over the entire section" led to numerical problems and to the computation being terminated. The setup of the phase separation model (Wilson bubble-rise-and slip model) in the plenums in conjunction with fine nodalization produced for satisfactory results a 2F LOCA experiment, if, in a first approximation, at the beginning of flow reversal (from a positive to a negative core mass flow), the computation was switched from computation with phase separation in the subplenum to homogeneous computation. (Problem: feeding a homogeneous two-phase mixture into a vapor dome).

Under these boundary conditions, the measured values were verified in the following way:

The pressure curve occasionally showed deviations up to some 7 bar, the cladding temperatures were calculated with sufficient accuracy at 50 K. Since the pressure curve shows systematic deviations from the measured pressures in all calculations by means of RELAP4/MOD6, this point needs to be clarified in a continuing study.

6.3 Implementation of DRUFAN 02

The DRUFAN 02 thermodynamic code (development status as of February 13, 1982) was implemented on the IBM 3033 of KfK in two steps:

(1) Implementation of the load module delivered and computation of a first example.

(2) Compilation of the source delivered and computation of a first example.

The first step was completed relatively quickly. However, contrary to all expectations, the following difficulties were encountered in the second step:

- Application of the FORTRAN 77 compiler, V1L24, produced a condition code > 12.
- Two subprograms were missing in the source.

The missing subroutines (CPU check;... job name querying, date ... for compiling restart file) were taken from the corresponding library routines of KfK. The program errors were communicated to the author (GRS, Munich) and corrected there.

7. Plans for Future Work

- Another test series with expanded cladding tubes will be carried out on the pellet relocation problem.
- It will be possible to assess the influence of spacers and the oblique position of the simulator in the channel on the transient development in a series of experiments.
- Various "simple hydraulic tests" (e.g., with the heaters turned off in the test section) have been planned to verify individual phenomena in the recalculation.
- Intercomparison calculations of various experiments are at present being conducted with RELAP4/MOD6 and DRUFAN 02 (perhaps also with RELAP5).
- A final summary of the COSIMA experiments will be conducted in the course of next year.

8. Relation with other Projects

PNS 4231, 4237, 4238, 4239

9. Reference

-

10. Degree of Availability

<table>
<tr><td>Berichtszeitraum/Period
1.1. - 31.12.1982</td><td>Klassifikation/Classification
1.3</td><td>Kennzeichen/Project Number
06.01.08/05A (PNS 4237)</td></tr>
<tr><td colspan="2" rowspan="3">Vorhaben/Project Title
Untersuchungen zum Brennstabverhalten in der 2. Aufheizphase eines Kühlmittelverlust-Störfalles. In-pile-Versuche mit Einzelstäben im DK-Loop des FR2.
Investigations on the 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.</td><td>Land/Country</td></tr>
<tr><td>Fördernde Institution/Sponsor
BMFT</td></tr>
<tr><td>Auftragnehmer/Zuwendgsempf./Contractor
KfK
(Projekt Nukleare Sicherheit)
(Hauptabteilung Ingenieurtechnik)</td></tr>
<tr><td>Arbeitsbeginn/Initiated
1.7.1972</td><td>Arbeitsende/Completed
31.12.1982</td><td>Leiter des Vorhabens/Project Leader
L.Sepold, E.H.Karb</td></tr>
<tr><td>Stand der Arbeiten/Status
completed</td><td>Berichtsdatum/Last Updating</td><td>Bewilligte Mittel/Funds</td></tr>
</table>

1. General Aim

Investigations 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 rod 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

39 nuclear tests with unirradiated as well as with previously irradiated fuel rods. Irradiation in the FR2 reactor, steps of burnup: 0/ 2,500/8,000/20,000/35,000 MWd/t_U. Range of internal pressures: 25 to 125 bar. Eight reference tests with electrically heated fuel rod simulators (BSS). Both nuclear and reference tests were performed with the test rods in the in-pile test section of the FR2 DK loop.

4. Experimental Facilities, Computer Codes

The DK loop was 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.
Upon completion of the test, the specimen was subject to examinations in the neutron radiography facility (NERA) of the FR2 and in the KfK hot cells. WALHYD2D (IKE, Stuttgart), STATI (KfK) and SSYST (KfK) computer codes were used for posttest calculations.

5. Progress to Date

Completion of the entire research program by end of 1982. The final results were documented in reports /1/, /2/.

6. Results

The main results are:

(1) No influence of the nuclear environment on the mechanisms of fuel failure during a LOCA. (2) No influence of burnup on the burst data and on the circumferential strain was found. (3) The pre-irradiated rods resulted in a fuel cracked during irradiation and fragmented during the transient tests due to cladding ballooning. (4) The fuel fragmentation did not influence the cladding deformation process.

7. Next Steps

-

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/ L.Sepold, E.H.Karb, M.Prüßmann, "In-pile-Experimente zum Brennstabverhalten beim Kühlmittelverlust-Störfall, Bericht über die Versuchsserie E, KfK 3345 (1982)

/2/ E.H.Karb et al.; "LWR Fuel Rod Behavior in the FR2 In-pile Tests Simulating the Heatup Phase of a LOCA, Final Report", KfK 3346 (to be published)

Berichtszeitraum/Period 1.1.82-31.12.82	Klassifikation/Classification 1.3	Kennzeichen/Project Number 06.01.09/05A (PNS 4238)
Vorhaben/Project Title Untersuchungen zur Wechselwirkung zwischen aufblähenden Zircaloy-Hüllen und einsetzender Kernnotkühlung (REBEKA-Programm) Studies of the Interaction between Ballooning Zircaloy Claddings and the Emergency Core Cooling (REBEKA-Program)		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe GmbH (KfK) Projekt Nukleare Sicherheit(PNS), IRB
Arbeitsbeginn/Initiated 1973	Arbeitsende/Completed 1983	Leiter des Vorhabens/Project Leader K. Wiehr
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

The general aim of the project is the development of experimental information about the ballooning of Zircaloy claddings during the refill and flooding phases of a loss-of-coolant accident for the verification and further development of the SSYST code system.

2. Particular Objectives

To study the deformation behavior 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 and the influence of ECC on it.
- Studies of the thermal and mechanical interaction between rods
- Generation of information about possible failure propagation.
- Studies of the extent and distribution of cooling channel blockages.

3. Research Program

The experimental program is carried out in a number of consecutive series of parameter experiments starting from basic single rod experiments on separate effects in a steam atmosphere and finishing with bundle flooding experiments of bundle assemblies including 7x7 rods (49 rod bundle), accompanied by theoretical studies.

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. The deformation of the Zircaloy cladding tubes is recorded by X-ray filming. Some 260 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.

5. Progress to Date

- Single rod experiments in He- and steam-atmosphere under flat topped temperature transient test conditions (plateau)
- improvement of the burst criterion relating to the influence of oxidation on the burst stress.
- Completion and evaluation of the 1st 7x7 bundle experiment with flooding.

6. Results

The bundle experiment REBEKA 5 with flooding (7x7 array) was performed. All 49 fuel rod simulators had Zircaloy claddings and internal overpressure. The outer ring of fuel rod simulators was able to balloon in this experiment to enable a maximum mechanical interaction between the Zircaloy claddings. The questions to be investigated were, if an intensive rod to rod interaction, a continuous cosine shape axial power profile and/or the bundle size itself would increase the strains and the coolant channel blockage. The test procedure was the following: internal overpressure 70 bar He at equilibrium bundle temperature of 145 °C (system pressure 4 bar). The heat-up rate from 145 °C to 765 °C was about 7 K/s. During the heat up phase there was a steam velocity of about 2 m/s downstream through the bundle. Flooding with a cold flooding rate of about 3 cm/s was initiated at a maximum cladding temperature of 765 °C.
The deformation started about 15 seconds before start of flooding at a maximum internal overpressure of 88 bar, the cladding temperature in the plateau during the flooding phase was about 800 °C and bursts occurred in the average at about 68 bar and 800 °C early during reflood. The burst interval of the inner 25 claddings was between 25 seconds.
The outer ring of the Zircaloy claddings ballooned under azimuthal temperature distributions and will not be taken into account in the comparison with REBEKA 3 results.
The mean value of maximum burst strains of the inner 5x5 claddings was 52 %, of the inner 3x3 48 %. The maximum channel blockage was 52 %. In spite of the somewhat higher max. burst strain, which is

understood, the coolant channel blockage of 52 % is identical with the REBEKA 3 result. The REBEKA 5 results are in a good agreement with recent knowledge. In spite of the continuous cosine shape axial power profile no increased coolant channel blockage was generated. The mechanical interactions between the rods were not dominant. No influence of the bundle size on the strain and the coolant channel blockage was found.
The REBEKA-bundle tests 1-5 were evaluated in respect to the influence of the flow direction through the bundle on the axial distribution of the locations of maximum strains in the bundle, which determines the max. flow blockage. All 5 experiments show the expected influence of flow direction on the plastic deformation of the Zircaloy claddings.

7. Plans for future work

- Single rod tests with regard to further developing the burst criterion .
- 7x7 rod bundle test without changing the flow direction in the refill- and reflood phase (International Standard Problem ISP 14, reference to NRU-tests).

8. Relations with other Projects

KfK: PNS 4231, 4235, 4236, 4237, 4239. KWU: RS 107, R 36, USNRC: MRBT, NRU.

9. References

Erbacher, F.J.; Neitzel, H.J.; Rosinger, H.; Schmidt, H.; Wiehr, K.: Burst criterion of Zircaloy fuel claddings in a loss-of-coolant accident. Zirconium in the Nuclear Industry. Proc. of the 5th Intern. Conf., Boston, Mass., August 4-7, 1980, STP 754, 1982.
Fiege, A. et al.: Stand und Ergebnisse der Untersuchungen des PNS zum LWR-Brennstabverhalten bei Kühlmittelverluststörfällen. KfK-3422 (Okt. 82)
Wiehr, K. et. al.: Untersuchungen zur Wechselwirkung zwischen aufblähenden Zircaloy-Hüllen und einsetzender Kernnotkühlung. KfK-3250 (Juni 82) S. 4200/90-4200/121

10. Availability of Reports

Available as KfK reports or conference reports.

Berichtszeitraum/Period 01.01.82 - 31.12.82	Klassifikation/Classification 1.3	Kennzeichen/Project Number 06.01.16/11A (PNS 4254)
Vorhaben/Project Title Oxidationsverhalten von Zircaloy-Hüllrohren Oxidation Behavior of Zircaloy Cladding Tubes		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS), IMF II
Arbeitsbeginn/Initiated 01.01.81	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader Dr. S. Leistikow
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

Investigation of Zircaloy cladding behavior under accidental conditions.

2. Particular Objective

Investigation of high temperature oxidation of Zircaloy-4 in steam and its influence on the mechanical properties.

3. Research Program

Isothermal exposures to measure the kinetics of oxygen consumption of Zircaloy-4 tube material under high temperatures. Creep-rupture and burst testing of oxidizing or pre-oxidized capsule specimens.

4. Experimental Facilities

High temperature furnaces for exposure of tube sections to flowing steam at temperatures up to 1800°C. Creep-rupture test facilities for exposure of tube specimen capsules to flowing steam while being internally pressurized.

5. Progress to Date

High-temperature oxidation testing of Zircaloy-4 cladding tube sections in steam at 1350 to 1600°C. Measurement of oxidation kinetics and oxidation-induced cladding deformation. Evaluation by metallography, hardness measurements, and fractography.

Extended exposures up to 25 hrs in the temperature range 600 - 1300°C to establish a comprehensive survey on kinetics of weight gain, oxide/α-Zr(O)-layer growth and scale morphology.

6. Results

Within the temperature range 1350 - 1600°C it is possible to describe the resultant reaction turnover of material by a parabolic time function. Up to total metal consumption the oxide is forming as adherent and consequently protective scales. In the Arrhenius representation of the parabolic rate constants versus the reciprocal absolute temperature the results are fitting to the earlier investigations at temperatures $\leq 1300^{\circ}C$. The discontinuity of the linear function occurring at $1550^{\circ}C$ is to be attributed to the transformation of tetragonal ZrO_2 into the cubic structure. The massive embrittlement of the α-Zr(O) phase, adjacent to the ZrO_2, is indicated by the typical brittle fracture of the oxidized tube material. Microhardness measurements showed the oxygen concentration profiles in the α-Zr(O) and the transformed β-phase. Dimensional measurements allowed to register radial and axial strains occurring under the influence of the progressive oxidation.

A comprehensive survey is now being prepared for publication on the oxidation kinetics of Zircaloy-4 in the temperature range 600 - $1600^{\circ}C$ and during exposure times up to 25 hours. The documentation will be completed by a whole set of metallographic pictures showing the state of the material from very short steam exposures up to total oxidative consumption.

7. Next Steps

Publication of the above mentioned documentation. Temperature-transient long-time exposures.

8. Relations to other Programs

KfK - PNS, KWU, USNRC, JAERI, CEA

9. References

Atef E. Aly
"Oxidation of Zircaloy-4 Tubing in Steam at 1350 to $1600^{\circ}C$."
KfK-3358 (1982)
S. Leistikow, G. Schanz in
"Stand und Ergebnisse der Untersuchungen des PNS zum LWR-Brennstabverhalten bei Kühlmittelverluststörfällen"
KfK-3422 (1982) 54-70

10. Degree of Availability

Open literature

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	1.3	06.01.17/.07A(PNS 4240.1)
Vorhaben/Project Title Untersuchungen zum Störfallverhalten fortgeschrittener Druckwasserreaktoren (FDWR) Investigations on accident behavior of advanced pressurized water reactors (APWR)		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS), IRB
Arbeitsbeginn/Initiated 1980	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader F. J. Erbacher
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

Investigation of the fuel element behavior in a loss-of-coolant accident.

2. Particular Objectives

- Deformation and burst behavior of Zircaloy (Zry) claddings and stainless steel (SS) claddings without and with integral helical fins (single rod burst tests).
- Flooding behavior of closely packed hexagonal rod bundles (bundle flooding tests).
- Flow blockage in a rod bundle due to deformed Zry- and SS-claddings (bundle tests).
- Modeling of the deformation and flooding behavior.

3. Research Program

- Out-of-pile single rod tests in steam using shortened fuel rod simulators.
- Out-of-pile bundle tests with flooding on 61 rod bundles using full-length fuel rod simulators with an axial power profile.
- Theoretical studies.

4. Test Facilities

Test rigs are available for single rod tests in steam and bundle tests with flooding. The deformation of the cladding tubes is recorded by X-ray filming in the single rod tests. A fastcomputer controlled data acquisition system using a PDP 11/03 is available. Some 260 measuring data (temperature, pressure, power, level etc.) of the bundle tests are recorded 10 times per second with a scanning frequency of 100 kHz.

2

5. Progress to Date

- Development of electrically heated fuel rod simulators with cosine shaped axial power profile.
- Development of a testing device to measure the local electrical resistance of the heating element (qualification of the axial power profile).
- Fabrication of prototype fuel rod simulators.
- Modification of the test rigs.

6. Results

Work was concentrated on the development of a suitable experimental technique. Most of this preparatory work has been completed successfully. The test program can be started in 1983.

7. Next Steps

- Fabrication of fuel rod simulators.
- Assembly and instrumentation of the 61-rod test bundle.
- Testing of shortened single rods to investigate the deformation of smooth SS-cladding tubes and those with helical fins.
- Bundle test on a 61-rod bundle with Zircaloy cladding tubes.

8. Relations with other Projects

Cooperation exists within KfK with INR and IMF, externally with KWU-Erlangen and "Lehrstuhl für Raumflugtechnik und Reaktortechnik" of the University Braunschweig.

9. References

10. Availability of Reports

Available as KfK-reports or conference reports.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	1.3	06.01.17/ 09A (PNS 4240.3)
Vorhaben/Project Title Oxidationsverhalten von Edelstahl-Hüllrohren in Wasserdampf Oxidation Behavior of Stainless Steel Cladding Tubes in Steam		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS), IMF II
Arbeitsbeginn/Initiated 01.01.81	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader Dr. S. Leistikow
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

Investigation of accidental behavior of stainless steel cladding materials.

2. Particular Objective

Investigation of the accidental oxidation behavior of austenitic and ferritic steel cladding tubes in steam.

3. Research Program

Isothermal experiments to measure the steam oxidation kinetics of 15Cr15Ni austenitic stainless steel (DIN No. 1.4970) and 12Cr ferritic stainless steel (DIN No. 1.4914) at 600 - 1300 $^{\circ}$C and a duration up to total wall consumption. Comparison with Zircaloy-4 under equal conditions.

4. Experimental Facilities

Laboratory loops for isothermal high temperature exposure of tube sections to slowly flowing steam at atmospheric pressure.

5. Progress to Date

Investigations into the oxidation kinetics of the 15Cr15Ni austenitic steel at 600 - 1300 $^{\circ}$C during 2 - 6 hrs and of 12Cr ferritic steel at 900 - 1300 $^{\circ}$C up to 6 hrs completed.

6. Results

Some additional experiments, in which 15Cr15Ni steel tube specimens at various temperatures were inserted, accomplished the data set by which the Arrhenius equation of the parabolic steam oxidation within the temperature range 600 - 1300 $^{\circ}$C could be calculated: $K_p = 2{,}80 \cdot 10^{11} \exp(-227/RT)$

Kp is the parabolic rate constant, the activation energy Q is given in kJ/mol. Equal steam oxidation experiments, in which 12Cr steel tube and sheet specimens were inserted, were performed between 900 and 1300°C. The maximum duration of exposure was initially 6 hrs (at 900°C) and was reduced with increasing temperature to 2 hrs (at 1300°C), since after 4 hrs at 1200°C and 2 hrs at 1300°C total consumption of the exposed specimens could be registered. Within this whole time/temperature range the weight gain of the materials as function of time of exposure could be described by parabolic functions.

The oxide scales formed in this way were composed of three subscales: an internal spinel phase, a medium one of mixed wustite and magnetite and an external thin one of magnetite. The sheet material oxidized between 1000 and 1100°C showed an overall instability of the metal/oxide bondage by spalling and blister formation, at the other temperatures the adherence of the oxide scale was improved with increasing duration of steam exposure. The increase of external and internal tube diameter by swelling and after total consumption was about 12-15 % each.

7. Next Steps

Further tests of the ferritic 12Cr steel at 600 - 900°C.

8. Relations to other Programs

KfK - PNS, KWU

9. References

S. Leistikow
"Comparison of High Temperature Steam Oxidation Kinetics under LWR Accident Conditions: Zircaloy-4 versus Austenitic Stainless Steel"
Proc. 6th Intern. Conf. "Zirconium in the Nuclear Industry", June 28 to July 1, 1982; Vancouver, BC, Canada.

Z. Zurek
"Isothermal Steam Oxidation of the Ferritic 12Cr Steel (DIN No. 1.4914) at 900 - 1300°C."
KfK 3436 (1982)

10. Degree of Availability

Open literature

Berichtszeitraum/Period 01.01.82 - 31.12.82	Klassifikation/Classification 1.3	Kennzeichen/Project Number 06.01.17/10A (PNS 4240.4)
Vorhaben/Project Title Untersuchungen zum mechanischen Verhalten von Hüllwerkstoffen für FDWR Investigations of the mechanical behavior of cladding material for APWR		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe mbH (KfK), Projekt Nukleare Sicherheit (PNS), IMF-II
Arbeitsbeginn/Initiated 01.01.1981	Arbeitsende/Completed 31.12.82	Leiter des Vorhabens/Project Leader Petersen
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

Investigation of the plastic behavior of various austenitic and ferritic steels during different reactor incidents, especially LOCA-typical temperature- and stress - transients and in a LOCA-typical environment.

2. Particular Objectives

Selection of appropriate cladding materials for Advanced Pressurized Water Reactors (APWR) considering the mechanical properties at emergency conditions.

3. Research Program

3.1 Tensile testing and creep testing of austenitic and ferritic steels at high temperatures without preirradiation.

3.1.1 Influence of precipitation behavior on the plastic properties.

3.1.2 Influence of oxidation on the plastic properties.

3.2 Burst experiments on the selected steel.

3.2.1 Temperature transient tests.

3.2.2 Combined experiments in steam atmosphere.

3.3 Examination of mechanical properties after preirradiation.

3.4 Preparation of verified burst criterions.

4. Experimental Facilities

Ref. 3.1 Tensile and creep testing will be performed in INSTRON "closed-loop" machines in air and vacuum.

Ref. 3.2 For burst tests the FABIOLA facility with internal heater will be used.

5. Progress to Date

The literature search to select alternative cladding material for APWR is completed.

Ref. 3.1 Performance of creep tests on the steel W.Nr. 1.4970 (X10 NiCrMoTiB 1515).

6. Results

The literature search to select austenitic and ferritic, martensitic steels as cladding material for APWR is completed. The data about physical and mechanical properties had been found up to a temperature-range of 800°C maximum [1]. Therefore the necessity exist, to run own tests on the selected steels in the temperature range above 800°C to obtain data about their mechanical properties.

Ref. 3.1 The commencement of operation of a new installed Instron testing machine with a high vacuum, high temperature furnace caused several difficulties, because the power control of the furnace influenced the controller of the testing machine. So, instead of running the test program, the efforts had been concentrated on the elimination of the trouble.

7. Next Steps

Ref. 3.1 Continuation of tensile and creep tests.

Ref. 3.4 Check of the existing failure models in regard to their applicability on the tested materials.

8. Relation to other Projects

PNS 4240.1 (06.01.17/07A)
PNS 4240.2 (06.01.17/08A)
PNS 4240.3 (06.01.17/09A)

9. References

[1] C. Petersen
"Literaturübersicht über mechanische und physikalische Eigenschaften von ausgewählten Stählen bei höheren Temperaturen"
KfK-Report Nr. 3469 (in press)

10. Degree of Availability of the Reports

Unrestricted distribution

145-2-02

1 - 3

TITRE

COMPORTEMENT DES GAINES EN ZIRCALOY PENDANT UN ACCIDENT DE DEPRESSURISATION (PROGRAMME EDGAR-ZY DE LA COMMISSION MIXTE CEA-EDF SURETE EAU ORDINAIRE)

TITRE en anglais

ZIRCALOY FUEL CLADDING BEHAVIOUR DURING A LOSS-OF-COOLANT ACCIDENT (EDGAR-Zy PROGRAM OF THE CEA-EDF JOINT COMMITTEE FOR LWR SAFETY)

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CEA/IPSN
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CEA/DTech/SRMA
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Responsables
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Date de démarrage	1974	*Etat actuel*	En cours
Date d'achèvement	1984	*Dernière mise à jour*	Mars 1983

Scientifiques
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1) OBJECTIF GENERAL

Etude du comportement des gaines en zircaloy au cours d'une perte de réfrigérant qu'il s'agisse d'une "grande" ou d'une "petite" brèche.

Le comportement mécanique des gaines a un impact notable sur les caractéristiques du refroidissement du coeur à la suite d'un LOCA :

- Le gonflement de la gaine soumise à une pression différentielle peut être suffisant pour que celle-ci atteigne des déformations plastiques et éventuellement la rupture, provoquant un rétrécissement de la section de passage du fluide et une oxydation de la gaine sur ses deux faces.

- Les modifications dimensionnelles de la gaine affectent les transferts de chaleur pastille gaine, ainsi que le refroidissement du crayon par le fluide primaire.

2) OBJECTIFS PARTICULIERS

Le programme consiste en des essais d'éclatement à haute température de gaines en zircaloy-4, recristallisées ou détendues, neuves ou irradiées (avec ou sans pastilles d'U02).

Deux objectifs particuliers sont à dégager :

a. La détermination expérimentale de la ductilité diamétrale des gaines de zircaloy-4 dans le but de modélisation et pour la définition d'un critère de rupture ; il s'agit d'études paramétriques en cinétique rapide ou lente.

b. La détermination de l'évolution en fonction du temps de la déformation diamétrale des gaines dans les conditions d'un LOCA grosse brèche, c'est-à-dire avec des températures et des pressions différentielles obéissant à des lois données en fonction du temps.

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3) INSTALLATIONS EXPERIMENTALES

Deux installations ont été réalisées pour des expériences sur une gaine unique, chauffée par effet Joule, sans écran thermique.
"EDGAR Froid" pour les essais sur gaines neuves, "EDGAR Chaud", copie de la première installation, mais en cellule pour les essais sur gaines irradiées.

4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

- En 1980, 60 essais ont été réalisés en cinétique rapide sur des gaines neuves recristallisées en présence de vapeur d'eau (objectif particulier a) ; le domaine de pressions différentielles analysé varie de 25 à 125 bars, tandis que celui des vitesses de montée en température s'étend de 5 à 100°C/s. (réf.1).
  La modélisation correspondante de la déformation de la gaine est achevée (réf. 2).

- En 1981, 50 essais ont été effectués en cinétique rapide sur des gaines neuves détendues en présence de vapeur d'eau dans les mêmes plages de pressions et de vitesses que précédemment. (réf. 3).
  Leur exploitation est terminée (réf. 4).

- En 1980 et 1981, 17 essais ont été réalisés pour l'objectif particulier b) sur des gaines recristallisées neuves, et irradiées sans combustible (crayons carquois) à différentes fluences (réf. 5).
  Il n'apparaît pas d'influence de l'irradiation sur le temps à la rupture.

- En 1981, 8 essais ont été exécutés pour l'objectif particulier b) sur des gaines recristallisées irradiées avec du combustible dans la CAP (réf. 6).
  Le comportement des gaines irradiées (temps à la rupture et déformation) est très comparable à celui observé pour les gaines non irradiées.
  Les caractéristiques initiales des gaines (résistance au fluage, ductilité) semblent donc restaurées dès le début du transitoire de température.

- En 1982, 8 essais ont été effectués pour l'objectif particulier b) sur des gaines en zircaloy 4 détendues, préalablement irradiées sans combustible dans SILOE (réf. 7).
  Les vitesses de fluage observées sont supérieures à celles rencontrées lors des essais sur gaines non irradiées, conduisant à des déformations importantes et à des temps à la rupture plus courts.

- En 1982, 16 essais analytiques à cinétique lente (rampes de température de 0, 2 et 1°C/s) ont été réalisés à pression interne constante (10, 25, 50 ou 75 bars) sur des gaines CEA (Zy4 fort) non irradiées (réf. 8).
  On observe d'une part, une augmentation notable des valeurs d'allongement circonférentiel réparti par rapport à celles obtenues lors des essais à cinétique rapide, et d'autre part, une forte dépendance de la ductilité vis-à-vis de la vitesse de montée en température dans le domaine des faibles vitesses, contrairement à ce qui avait été observé pour des vitesses $\geqslant$ 25°C/s.

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- En 1982, 24 essais d'éclatement avec des rampes de température à cinétique lente (0,2 , 0,5 et 1°C/s), sous des pressions internes constantes de 10, 25, 50 et 75 bars, ont été exécutés sur des gaines en zircaloy 4 détendu non irradiées.
  8 essais isothermes (800°C) ont également été effectués avec des rampes de pression (0,1 , 0,5 , 1 et 2 bars/s) sur ce même type de gaines (réf. 9). Les résultats obtenus sont assez semblables à ceux concernant les gaines CEA.

5) PROCHAINES ETAPES

- Travaux d'interprétation et de modélisation en 1983, comportant la mise sur fichier de la totalité des essais EDGAR froid (gaines Zy4 recristallisées et détendues), la formulation de lois semi-empiriques de déformation des gaines non irradiées (toutes cinétiques), des calculs de reconstitution de transitoires type LOCA grosse brèche réalisés lors d'essais EDGAR chaud et la définition de critères de rupture de gaine.

- 8 essais EDGAR chaud sur des gaines en zircaloy 4 détendu, irradiées avec du combustible dans le réacteur Fessenheim, sont prévus en 1983 (transitoires du type LOCA grosse brèche).
  Ces essais, complémentaires de ceux effectués en 1982 sur des gaines préalablement irradiées dans SILOE, devraient permettre de préciser l'effet de l'irradiation sur la déformation et le temps de rupture.

- Des essais complémentaires EDGAR froid seront également réalisés pour quantifier les effets sur le fluage ultérieur de la gaine d'une incursion dans la phase $\alpha + \beta$ du zircaloy 4 (1r pic de température de gaine lors d'un transitoire type LOCA grosse brèche).

6) RELATIONS AVEC D'AUTRES ETUDES

- Fiche 145-2-01. Développement de modèles et de codes de sûreté sur le comportement du combustible au cours d'accidents (APRP).

- Fiche 170-1-04. Programme PHEBUS. Expérience de dépressurisation et de renoyage d'une grappe combustible PWR.

7) DOCUMENTS DE REFERENCE

(1) Note d'essai DTech. RMA (81) 662. Mai 1981.
"Compte rendu des essais d'éclatement des tubes en zircaloy 4 recristallisé".

(2) Note technique DTech. SECS. SEECRE N° 909. Mai 1982.
"Qualification du modèle de déformation de gaine et du critère de rupture (gaines Zy4 fort recristallisé, type CEA)".

(3) Note d'essai DTech. SRMA (82) 707. Janvier 1982. Diffusion restreinte.
"Compte rendu des essais effectués en vue de la modélisation de la déformation en conditions accidentelles des tubes en zircaloy 4 détendu".

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(4) Note d'essai DTech. SRMA (82) 770. Décembre 1982. Diffusion restreinte. "Essai de modélisation de la déformation plastique en conditions transitoires du zircaloy 4 détendu (vitesses de chauffage comprises entre 5 et 100°C/s)".

(5) Note technique DTech. SECS. SEECRE 902 - SRMA 1134 - LECI 484. Février 1982.
"EDGAR chaud. Résultats des essais d'éclatement sur gaines carquois irradiées (fluence de 0,3 à 2 x 10^{21} n/cm2)".

(6) Note technique DTech. SECS/LECI N° 490. Mai 1982.
"EDGAR chaud. Résultats des essais d'éclatement sur gaines irradiées (fluences de 0,15 à 3,2 x 10^{21} n/cm2)".
Gaines de crayons irradiés dans le coeur I de la CAP.

(7) Note technique DTech. SECS. SELECI N° 513. Décembre 1982.
"Résultats des essais d'éclatement sur gaines irradiées. Gaines de zircaloy 4 détendu irradiées dans SILOE (fluence de 0,2 à 0,8 x 10^{21} n/cm2)".

(8) Note d'essai DTech. SRMA (82) 727. Mai 1982.
"EDGAR froid. Gaines CEA (Zy 4 fort). Résultats des essais analytiques à cinétique lente".

(9) Note d'essai DTech. SRMA (82) 768. Novembre 1982. Diffusion restreinte.
"EDGAR froid. Gaines FRAMATOME. Résultats des essais analytiques à cinétique lente".

8) DEGRE DE DISPONIBILITE

Documents internes non disponibles, sauf accord du CEA et de l'EDF.

170.1.04				1 - 3
TITRE PROGRAMME PHEBUS EXPERIENCE DE DEPRESSURISATION ET DE RENOYAGE D'UNE GRAPPE COMBUSTIBLE PWR				*Pays* FRANCE *Organisme directeur* CEA/DSN
TITRE en anglais PHEBUS PROGRAM DEPRESSURISATION AND REFLOADING EXPERIMENT OF PWR FUEL BUNDLE				*Organisme éxécuteur* DSN/DERS *Responsables* R. DEL NEGRO M. REOCREUX A. TATTEGRAIN
Date de démarrage	01/01/1979	*Etat actuel*	EN COURS	*Scientifiques* Ph. BERNA B. LEGRAND M. MORONI
Date d'achèvement	1985	*Dernière mise à jour*	04/83	

I - OBJECTIF GENERAL

Dans le cadre général des études sur le LOCA, les objectifs principaux du programme PHEBUS, sont les suivants :

a) Au cours d'essais de plus en plus sévères allant jusqu'à la perte d'intégrité de la gaine et la dislocation de l'assemblage et menés dans un environnement réaliste, étudier le comportement thermique et mécanique des gaines et leur cinétique d'oxydation en atmosphère de vapeur d'eau pendant le première phase de l'accident (dépressurisation).

b) L'étude du comportement thermique et métallurgique des crayons au cours du renoyage, sur un assemblage réel, quoique de hauteur réduite. En particulier, on vérifiera sur la tenue des crayons l'influence de l'épaisseur fragilisée et de la température avant trempe.

c) L'étude des intéractions entre crayons, des effets d'assemblage, de la forme et de la position des ballonnements sur les crayons en fonction des différents paramètres pris en compte, comme la puissance linéique, la pression interne, le débit et le moment d'injection.

d) L'étude du terme source que constituent les crayons rompus après l'accident, en complément du programme FLASH.

e) Enfin, la vérification de la conformité aux prévisions du comportement des crayons lorsque les critères imposés par la règlementation américaine 10 CFR 50 sont atteints.

f) Dans le cadre de ces études, viser à tester la capacité des codes intégraux et en particulier des codes combustibles à coupler les traitements des phénomènes sur une séquence accidentelle de type LOCA grosse brèche.

II - OBJECTIFS PARTICULIERS

Trois principaux objectifs peuvent être distingués :

1° - Exploitation et maintenance de l'installation PHEBUS (SES)

Ce premier objectif concerne la mise en service, l'exploitation et la maintenance de l'installation expérimentale PHEBUS décrite ci-après et permettant de réaliser l'objectif général précédent.

2° - Expérimentation (SES)

Ce second objectif concerne la préparation puis la réalisation des essais du projet. Il comprend :

- la définition précise desdits essais,
- l'étude neutronique, thermohydraulique et mécanique des dispositifs d'essais,
- l'étude de l'instrumentation permettant de réaliser les mesures nécessaires à l'interprétation des essais,
- la réalisation proprement dite des essais,
- leur dépouillement.

3° - Interprétation (SEAREL)

Cet objectif comprend trois types d'activités :

a) 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 sont 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.

b) Calculs prévisionnels

Ces calculs sont faits :

- soit pour définir les paramètres de l'essai,
- soit, les paramètres de l'essai étant figés, pour aider à la réalisation de celui-ci (tout essai devra être précédé de ce calcul prévisionnel).

c) Interprétation des essais

Cette interprétation doit comprendre :

- la réactualisation des calculs prévisionnels en fonction des conditions réelles de l'essai,
- la comparaison entre calculs "réactualisés" et résultats expérimentaux,
- l'analyse des résultats de comparaison calcul-expérience,
- la détermination des conséquences de cette analyse sur le programme expérimental,
- la réalisation de nouveaux calculs pour ajuster les modèles sur les résultats expérimentaux.

III - INSTALLATION EXPERIMENTALE ET PROGRAMME

III.1 - Installation expérimentale

L'installation comporte :

- Un réacteur source d'une puissance thermique de 45 MW, capable de produire les flux neutroniques et d'ajuster leur évolution pendant la séquence expérimentale (dépressurisation et action de refroidissement de secours) pour que l'énergie dégagée dans l'assemblage d'essai soit représentative de celle qui existerait dans un réacteur de puissance au cours d'un LOCA. Le coeur nourricier est composé de 36 assemblages à crayons d'U02 faiblement enrichi. Il est contrôlé au moyen de 6 barres de contrôle-sécurité à crayons de hafnium. Son refroidissement est assuré à partir d'une réserve d'eau déminéralisée qui sert de volant thermique et permet 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 bar de surpression et assurant les fonctions de :

    - maintien des conditions nominales de fonctionnement d'un PWR,
    - déclenchement de la dépressurisation (brèche de section et de localisation variables),
    - mise en oeuvre de l'injection de secours, dont on peut faire varier le débit, la température, l'instant et le point d'injection,
    - mesures des paramètres de fonctionnement et prélèvement d'échantillons.

- un ensemble d'acquisition et de 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 des signaux des divers capteurs.

III.2 - Programme

Le programme PHEBUS comporte quatre phases :

- Première phase

Les essais relatifs à cette première partie sont définis dans la grille donée en annexe 1. Cette grille prévoit :

. 4 essais non nucléaires, dont un monocrayon,
. 3 essais nucléaires un crayon, plus un optionnel,
. 7 essais nucléaires 25 crayons.

Elle a pour objectifs :
. la prise en main de l'installation jusqu'à des conditions sévères pour les essais 1 crayon,
. l'étude et la qualification thermohydraulique de la boucle et des codes servant à la prévision et l'interprétation des essais.

Cette partie du programme se termine par la réalisation de l'essai 215 qui est le premier essai de comportement de combustible avec un environnement thermohydraulique réaliste. Cet essai est réalisé avec des crayons combustibles prégonflés.

- Seconde phase

L'objectif prioritaire de cette seconde phase est la vérification expérimentale globale du comportement en pile d'un assemblage de 25 crayons représentatif des assemblages PWR 900. Ceci, dans un certain nombre de scénarios accidentels encadrant les conditions les plus sévères considérées dans le rapport standard de sûreté (rupture "0,5 x RG/BF").

Pour ce faire on prend comme référence la variation de la température de gaine au point chaud en fonction du temps après le début de la dépressurisation.

On ajuste la pression interne des crayons pour obtenir les ruptures des gaines soit au 1er pic ($\sim$920°C) soit dans les zones α ($\sim$ 810°C) ou $\alpha + \beta$ ($\sim$ 890°C).

L'essai est prolongé pour obtenir des oxydations à 1100 et 1200°C de 6 à 17 % de l'épaisseur de la gaine, le mode de refroidissement sera adapté en fonction de la fragilisation des gaines.

Un essai est prévu avec du combustible irradié pour mettre en évidence l'effet de l'irradiation sur le comportement de l'assemblage au cours du scénario accidentel type.

- Troisième phase

Des essais relatifs aux petites brèches conduisant à d'importantes dégradations du combustible sont en cours de définition. Une étude de faisabilité a été faite des études de projet sont en cours.

Cette phase a été limitée à $\sim$ 1800°C, limite de fusion du zircaloy et où l'assemblage gardera encore sa géométrie.

paramètres sont reportés dans le tableau N° 1 ci-après, et d'essais à caractère plus physique dont les résultats permettent de dépouiller les précédents,

. des études d'instrumentation et de réalisation des dispositifs d'essai.

- 1981 : Les études, le suivi de la fabrication, les modifications et le contrôle des dispositifs d'essai ont porté sur les perches qui ont permis de réaliser 5 essais du programme : 210, 211, 212, 102 et 103.

Les dates d'exécution de ces essais sont données dans le tableau 2 ainsi que le numéro des perches utilisées.

Les principaux paramètres de ces essais sont reportés dans le tableau 1 relatif à la première partie de la grille des essais.

De plus un nombre important d'essais permettant une meilleure connaissance des caractéristiques hydrauliques locales de la boucle a été réalisé.

Pour ce qui est de l'instrumentation, les problèmes de corrosion intergranulaire ont conduit à de nombreux essais de vérification puis à la qualification des thermocouples gainés d'acier 347.

La nécessité de limiter la perturbation de la mesure des températures des gaines et assurer une tenue de cette mesure entre 950 et 1200°C a conduit à développer des thermocouples à gaine de zircaloy et à étudier leur fixation aussi bien à l'extérieur (incrustation et soudure B.E.) qu'à l'intérieur (soudure par point) de la gaine.

Divers essais pour améliorer la tenue des capteurs de pression lors des transitoires de température ont été faits.

- Quatrième phase

Celle-ci concerne le comportement du combustible aux hautes températures (> 2000°C) et les possibilités de refroidir le combustible fortement dégradé pouvant même être sous forme de lits de débris.

IV. - ETAT D'AVANCEMENT

IV.1 - Expérimentation

a) Rappel des activités précédant 1982

- 1978 : Divergence de l'installation
  Essais neutroniques à basse puissance.

- 1979 : - Réalisation des travaux de modifications (lyres de dilatation et mesures de débit double phase) et de mise au point de la boucle, avec essais de réception correspondants.

  - Montée en puissance pour la qualification du réacteur et réalisation de :
    . 1 dépressurisation nucléaire sur dispositif monocrayon,
    . 1 dépressurisation nucléaire sur dispositif 25 crayons,
    . 1 dépressurisation non nucléaire du programme.

- 1980 : Durant cette année, les efforts ont porté plus particulièrement sur :
  . la conduite et le dépouillement des essais de démarrage,
  . la préparation et l'exécution des essais 206, 207, 208, 209, 101 et 101 bis du programme proprement dit dont les

b) Activités 1982

Sur le plan de l'exploitation de l'installation on a réalisé sept campagnes d'essais malgré un défaut de fonctionnement des vannes de dépressurisation d'une part, et des soupapes de sécurité d'autre part qui a provoqué un arrêt de quatre mois.

Le tableau N° 1 donne la grille générale des essais et plus particulièrment ceux réalisés en 1982.

Cinq essais ont donné de bons résultats, quatre dans la série 213 pour ajuster les paramètres en vue de la représentation la plus fidèle possible de la courbe de température de gaine pour la phase 2, le cinquième est l'essai 215 P concluant la phase I et qui a conduit à la déformation et la rupture des gaines de crayons portés jusqu'à 1150°C.

Les figures N° 1 à N° 5 donnent pour l'essai 215 P :

- la répartition de l'instrumentation dans la grappe,
- la variation de la température U02, gaine,
- la variation de la pression interne du crayon et montre l'instant de la rupture à t $\sim$ 55 à 60 b suivant les crayons,
- la déformation des gaines au voisinage du point chaud. Trois crayons ne se sont pas déformés car la pression interne initiale y était faible 1 et 5 b, tous les autres présentent des ruptures avec des déformations importantes des gaines des crayons centraux les plus chauds.
- l'amplitude des déplacements des pastilles et la déformation axiale des gaines,
- la forme de la déformation au voisinage de la rupture.

Les résultats de ces essais ont été présentés lors d'un congrès à Chicago en Août Septembre 1982 (Réf. au § VII).

IV.2 - Instrumentation

La mise au point et la qualification des thermocouples à gaine de zircaloy ont été poursuivies, elles ont conduit à la fabrication quasi-industrielle de thermocouples de grande longueur pour les dispositifs d'essai.

Des développements sont prévus pour la mise au point d'autres type de gainages pour une tenue en température et vapeur d'eau au delà de 1200 - 1400°C, ces études sont menées par le DTech/SECS à SACLAY.

Pour ce qui est des capteurs de pression de fluide, un capteur a été mis au point ; celui-ci présente lors des essais de qualification une dérive thermique moindre que les capteurs correspondants du commerce. Toujours en liaison avec le DTech/SECS des études et des réalisations de capteurs de pression fluide et crayons adaptés aux conditions des essais de sûreté sont prévues en 1983.

Les autres études d'instrumentation ont porté sur :

- l'utilisation de capteurs de mesure double phase dans la perche (moulinets, drag-body, etc...),
- le développement des techniques de protection des matériaux pour éviter la corrosion par la vapeur à haute température.

IV.2 - Interprétation

a) Rappel des activités précédant 1982

Celles-ci ont essentiellement porté sur la mise en place des moyens d'interprétation.

Cette mise en place a consisté en :

. La prise en main du code thermohydraulique et combustible RELAP 4 MOD 6 et des codes combustible CUPIDON, FRAP T2 et T4 suivie de la mise en place d'une technique de couplage de ces codes pour faire des études de sensibilité. Des calculs préliminaires prévisionnels et réactualisés ont été faits pour chacun des essais réalisés.

. La structuration des méthodes de travail pour l'interprétation de chaque essai et la recherche de l'origine des désaccords calculs expérience constatés.

. L'amélioration de la modélisation de la boucle pour la prévision et l'interprétation des essais.

. La définition et l'interprétation d'essais support pour cette modélisation.

b) Activités 1982

Les principaux résultats et les faits marquants pour 1982 sont résumés ci-après :

- Mise en évidence de l'intérêt de paramètres jugés jusqu'ici secondaires pour gouverner la thermohydraulique dans PHEBUS à travers l'analyse de 8 essais 25 crayons, (dont cinq réalisés en 82) tel que différence de températures entre branches chaude et froide, dépressurisation de l'eau de refroidissement du tube de force.

- Mise en évidence de l'insuffisance du code RELAP IV Mod 6 pour le calcul du débit dans le coeur, lorsque celui-ci est faible (effet de différence) donc de son insuffisance pour la prédiction des montées importantes de température dans la grappe.

- Observation qu'un retour brutal des gaines à Tsat se produit au cours de la dépressurisation dans toutes les expériences 25 crayons double brèches, sauf une (essai 213 B), et même dans une expérience dite hydraulique caractérisée par l'absence de chauffage nucléaire (essai 210).

- Aptitude à réaliser et à reproduire malgré le refroidissement prématuré une montée de température adiabatique prototypique de calcul d'évolution CP1 (essais 213A et 215P).

- Démonstration de la capacité de PHEBUS à réaliser un transitoire qui conduit à déformer de manière importante et à rompre des gaines de crayons de type PWR (essais 215P).

- Approche de la maitrise de remplissage au cours de la montée adiabatique pour amener l'eau au pied des cryaons avant le déclenchement du renoyage. Les premiers essais 83 devraient permettre d'éliminer les quelques brefs refroidissements intempestifs de la grappe qui apparaissent actuellement au cours du remplissage.

- Démonstration de la possibilité de réaliser une trempe progressive, donc prototypique des gaines en utilisant une valeur de débit modérée au cour du renoyage et en limitant les effets d'inertie des branches froide et chaude par un scénario approprié des brèches.

- Réalisation d'une étude de synthèse sur les causes et les remèdes possibles du refroidissement prématuré.

- Réflexions, à partir de l'analyse des résultats expérimentaux et des résultats calculés, sur les différentes manières d'obtenir l'enchaînement des différentes étapes d'un transitoire de température gaine à deux bosses du type évaluation CP1 :

  . un premier pic de température élevée qui conduit les gaines à la phase $\alpha + \beta$ vers 925°C ;

- un creux de température pour ramener les gaines à la phase α aux alentours de 750°C. De récents essais Edgar ont montré que le premier pic avait une nette influence à travers ce creux sur les déformations avant rupture ;
- une montée adiabatique régulière aux environs de 10°C/s avec remplissage pour amener l'eau juste au pied des crayons afin de démarrer le renoyage dans de bonnes conditions ;
- un ventuel plateau de température pour accentuer l'oxydation des gaines ;
- un renoyage à vitesse de trempe progressive.

- Présentation - poster d'un papier de synthèse sur les résultats et le programme des essais dans PHEBUS au congrès de CHICAGO (21/08 - 03/09).

- Présentation plus complète des activités PHEBUS en séminaire international le 09/11/82.

- Etudes préliminaires phase II :

  - définition de la grille des essais phase II ;
  - recherche des caractéristiques adéquates du ressort et du plénum pour avoir des déformations des crayons similaires à celles estimées en situation réacteur ;
  - appréciation des modifications à apporter au transitoire PHEBUS pour synchroniser les évolutions de température et d'écart de pression au travers la gaine suivant la chronologie évaluée pour CP1, en particulier allongement de la durée de dépressurisation de 18 à 22 s environ.

- Mise au point d'une méthode utilisant des calculs inverses par FRAP T4 pour reconstituer au niveau d'un crayon l'évolution de puissance au cours de la dépressurisation et l'évolution du coefficient de transfert thermique gaine-réfrigérant en aval du front de trempe pendant le renoyage.

- Etudes de sensiblité pour cerner en particulier l'effet de découpage des strutures, de la représentation en 4 volumes de la ligne d'injection de l'ERTF (Eau de Refroidissement du Tube de Force), de l'écart initial de température entre branche froide et branche chaude, du changement du modèle d'échange thermique à travers le jeu combustible gaine (Mac Donald and Broughton et Ross and Stoute) autour des conditions des essais 211, 212, 215P.

- Tous les calculs prévisionnels (essais 213A, B, E et 215P) et réactualisés (essais 210, 211, 212, 215P) ont été réalisés pour la thermohydraulique boucle par RELAP IV Mod 6, et, pour le comportement des crayons combustible, par RELAP IV Mod 6, sauf pour l'essai hydraulique 210, analysé par FRAP T4 et CUPIDON.

- L'alimentation du code FRAP T4 par les conditions thermohydrauliques expérimentales au lieu et place des conditions calculées par RELAP IV Mod 6 a été mise au point.

V - PROCHAINES ETAPES

V.1 - Expérimentation

1983

Un ou deux essais, 213 H et 215 R sont prévus pour finir d'ajuster la courbe d'évolution de température de gaine sur celle prévue pour les réacteurs de puissance CP1.

Trois essais (216, 217, 218) seront exécutés dans le cadre de la deuxième phase du programme. Ils sont destinés à :

- analyser le mécanisme de la rupture de la gaine au premier pic, au moment du minimum de température, ou lors de la montée adiabatique avec des crayons pressurisés,

- provoquer des taux d'oxydation progressifs jusqu'à 16 % de l'épaisseur de la gaine,

- mettre en évidence les effets du mode de refroidissement lent ou rapide.

Un essai à une température de $\sim$ 1600°C est à l'étude pour s'assurer de la tenue de l'instrumentation au delà de 1400°C, pour mettre en évidence les problèmes de fonctionnement et de sûreté liés à la phase III (coeur sévèrement dégradé) pour laquelle il pourra servir d'essai de démonstration.

Le développement de l'instrumentation spécifique sera poursuivi pour la préparation de cet essai et s'assurer de la tenue de celle-ci au voisinage des limites des critères ; il portera sur :

- la détermination de l'effet des hautes températures (θ_g > 1400°C) sur les thermocouples ; études d'isolants, matériaux de gaines,
- la réalisation et le test de prototypes en eau et vapeur dans des conditions réalistes en boucle hors pile ou dans des fours.

- l'étude et la réalisation de capteurs de pression fluide et crayons adaptés aux conditions rencontrées dans les essais de sûreté : capteurs à jauges de contraintes miniaturisées pour les crayons et adaptés pour les mesures dans des fluides à 900°C, capteur inductif dont la dérive en fonction de la température en transitoire soit réduite et connue.

1984

Poursuite du programme LOCA grosses brèches.

1985

Arrêt de l'installation pour modification en vu du programme PHEBUS C.S.D qui sera ultérieurement poursuivi en parallèle, si nécessaire, avec le programme LOCA grosses brèches portant sur du combustible irradié éventuellement.

V.2 - Interprétation

V.2.1 - Recherche sélective et calculs prévisionnels de différentes solutions pour la thermohydraulique de la boucle PHEBUS en vue de réaliser un transitoire de température à deux bosses de type évaluation CP1 sur les gaines, sans remouillage prématuré. Fin de cette partie prévue en mars. Calculs réactualisés.

V.2.2 - Analyse et interprétation des résultats obtenus par la réalisation des essais phase I :

- Choix de scénario de référence pour la phase II, au niveau de la boucle.
- Exploitation générale des résultats des essais phase I.
  En particulier : évaluation des fuites thermiques hors grappe à partir des montées adiabatiques réalisées, corrélativement détermination de la variation du coefficient de couplage sur la puissance pendant le transitoire.

- Mise au point du jeu standard de données, RELAP IV Mod 6.

- Fin mise au point et ajustement de modèles indépendants de RELAP IV Mod 6, pour la description du remplissage et du renoyage.

- Suivi essais ERSEC-PHEBUS-SYSTEME et analyse des résultats.

V.2.3 - Suivi des essais phase II :

- Etablissement du programme général de chaque essai.

- Définition de la pression interne des crayons en fonction de l'objectif de l'essai.

- Calculs prévisionnels avec les codes RELAP IV Mod 6, FRAP T4 et CUPIDON.

- Synthèse des enseignements à tirer des essais vis-à-vis de chaque aspect sûreté du comportement des crayons combustible.

- Rédaction des rapports d'interprétation partiels (au fur et à mesure de l'avancement du programme) et des rapports globaux.

- Fourniture des éléments pour reconstituer la forme du bouchage créé par la déformation à la rupture aux Unités chargées de l'étude des possibilités de refroidir de tels ensembles.

VI - RELATIONS AVEC D'AUTRES ETUDES

- Programme OMEGA et ERSEC dont ERSEC PHEBUS	145.1.03 145.1.04 145.1.09 145.1.11 170.1.05
- Comportement des éléments combustibles :	
. essais EDGAR,	145.2.02
. études d'instrumentation	170.1.06
. étude du comportement métallurgique de l'élément combustible PWR en régime accidentel : PIE.	145.2.02

VII - DOCUMENTS DE REFERENCE

. Principales publications avant 1982.

- 48 12 40 : RS N°1 à N°7/81, Rapports de dépouillement divers 1 et 25 crayons ainsi que calculs prévisionnels et résultats de spectrométrie et dosimétrie.

* - Plaquette PHEBUS - PHEBUS experimetal program. General Presentation and objectives - PHEBUS I - 1981

- NOTE PHEBUS N° 81/04
"Jeu Standard 2503 - Calcul de thermohydraulique et du combustible des expériences PHEBUS en configuration 25 crayons à l'aide du code RELAP IV Mod 6".
MM. ADROGUER - PIGNARD - Mme BERNE.

- NOTE TECHNIQUE SEAREL N° 81/40
" Calcul réactualisé PHEBUS - Expérience 207 - Perche 2502.
MM. ADROGUER, PIGNARD.

- NOTE TECHNIQUE SEAREL N° 81/56
  "APPLICATION DU CODE PSCHITT A LA TRANSPOSITION AIGUILLES ELECTRIQUES"
  M. MAILLAT.

. Publications 1982 (intégralité)

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48 20 10
N° 13/82
"ANALYSE DES CONSEQUENCES DU DEPASSEMENT DE LA PRESSION DE CALCUL SUR LES COMPOSANTS DE LA BOUCLE".
M. VEROT

48 20 10
N° 14/82
"COMPTE RENDU SUR LE MAUVAIS FONCTIONNEMENT DES SOUPAPES H.P LORS DE L'ESSAI 213 LE 10 FEVRIER 1982".
MM. CLEMENT - DEL NEGRO - MORONI

48 64 21
N° 20/82
"OUVERTURE DES VANNES DE DEPRESSURISATION SUR SEUIL MAXIMUM DE PRESSION DANS LA CELLULE EN PILE".
MM. DEL NEGRO - TABUS

48 03 20
N° 21/82
"ESTIMATION DE LA TEMPERATURE DU CORPS DES SOUPAPES SOEP 01 ET 02".
M. CLEMENT

48 50 53
N° 22/82
"DOSSIER DES APPAREILS A PRESSION DE VAPEUR SOUSPAES DE SECURITE SOEP 01 - SOEP 02".
M. BREYSSE

48 50 53

N° 23/82

"ESSAI DE FONCTIONNEMENT IN SITU DES SOUPAPES".

M. MORONI

48 11 20

N° 26/82

"ETUDE DE FAISABILITE DU PALLIER DE TEMPERATURE PENDANT LA PHASE ADIABATIQUE".

M. GONNIER

48 11 20

N° 40/82

"ETUDE DE FAISABILITE DE L'ESSAI 215 BIS (OBTENTION DE HAUTS NIVEAUX DE TEMPERATURE SUR LES GAINES)".

M. GONNIER

48 90 60

N° 47/82

"ETALONNAGE DES MOULINETS PHEBUS M1 ET M2 SUR IN 110".

M. BEYLY

48 14 20

N° 55/82

"ESSAI HYDRAULIQUE SUPPORT MESURE DE LA PERTE DE CHARGE ENTRE BRANCHE CHAUDE ET BRANCHE FROIDE EN 25 CRAYONS"

Melle COURT

48 13 40

N° 56/82

"ESSAI HYDRAULIQUE SUPPORT MESURE DU COEFFICIENT DE PERTE DE CHARGE DE VAEP 14".

Melle COURT

48 14 20

N° 57/82

"ESSAI HYDRAULIQUE SUPPORT MESURE DE LA PERTE DE CHARGE ENTRE BRANCHE CHAUDE ET BRANCHE FROIDE EN CONFIGURATION MONOCRAYON"

Melle COURT

48 50 53

N° 58/82

"CR ESSAI IN SITU DU FONCTIONNEMENT A CHAUD DES SOUPAPES SOEP 01 ET 02"

MM. CAISSO - GRAEFF

48 03 20

N° 60/82

"CR SUR LE DEFAUT DE FONCTIONNEMENT DES THERMOCOUPLES DE SURVEILLANCE DU TUBE DE FORCE"

MM. DEL NEGRO - MORONI

48 13 80

N° 62/82

"ETUDE DE FAISABILITE DE MESURE DOUBLE PHASE DANS LES PERCHES PHEBUS"

M. ROMAIN

48 13 80

N° 64/82

"TECHNOLOGIE D'IMPLANTATION DE LA MESURE DOUBLE PHASE EN PARTIE BASSE DE PERCHE PHEBUS"

M. ROMAIN

N°

REGLES GENERALES D'EXPLOITATION

MISE A JOUR GENERALE EN DECEMBRE 82

NOTES TECHNIQUE SEAREL

N° 82/57
"BILAN DES CONNAISSANCES AU DSN/SRS SUR LE COMPORTEMENT DES ELEMENTS COMBUSTIBLES D'UN REACTEUR PWR AU COURS D'UN LOCA".
M. PERRET

N° 82/62
"CONTRIBUTION DE FAISABILITE DES ESSAIS PHEBUS PHASE II.
EVALUATION DE LA PUISSANCE DEGAGEE PAR LA REACTION ZIRCALOY-EAU (ELABORATION D'UN MODULE DE CALCUL). RESISTANCE DES THERMOCOUPLES GAINES-ZIRCALOY".
M. PERRET

N° 82/68
"CALCUL PREVISIONNEL PHEBUS - EXPERIENCE 213 A".
M. SENEMEAUD

---

* - International Meeting on Thermal Nuclear Safety, Chicago 29/08 - 2/09/82

* - R. DEL NEGRO, M. REOCREUX, J. PELCE, B. LEGRAND, Ph. BERNA
PHEBUS PROGRAM - First Results on PWR Fuel Behaviour in Loca Conditions.

* - Revue Générale Nucléaire N° 82/04 - J. DUCO, R. DEL NEGRO, J. PELCE M. REOCREUX (DSN) - M. CHAGROT (DTECH) et C. JANVIER (DMG)
Comportement du Combustible en Situation Accidentelle - Le Programme PHEBUS.

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N° 82/07
"CALCUL PREVISIONNEL PHEBUS - EXPERIENCE 212 - PERCHE 2502"
Mme BERNE - M. PIGNARD

N° 82/08
"ESSAI N° 206 - DEPRESSURISATION SIMPLE BRECHE BOUCLE CHAUDE A PUISSANCE NULLE"
M. PELOU

N° 82/09
"BILAN DE L'INSTRUMENTATION SUR LA PERCHE 0103 APRES LES ESSAIS 102 ET 103"
M. BARTHELEMY

N° 82/10
"DEFINITION DES EXAMENS APRES ESSAI SUR L'ELEMENT COMBUSTIBLE DE LA PERCHE PHEBUS POUR L'INTERPRETATION DES ESSAIS 102 ET 103"
M. PERRET

N° 82/11
"DEGRE DE FIABILITE ATTENDU DES RESULTATS PRESENTES DANS LE RAPPORT PRELIMINAIRE DE DEPOUILLEMENT (RPD)".
M. SCOTT DE MARTINVILLE

N° 82/12
"DEFINITION DE L'INSTRUMENTATION MINIMALE D'UNE PERCHE PHEBUS PERMETTANT SA REUTILISATION".
GROUPE SEAREL

N° 82/13
"ERSEC-PHEBUS PRESENTATION DES CALCULS DE 4 ESSAIS DE RENOYAGE".
MM. RODRIGUEZ - SENEMEAUD - Mme POUBLAN

N° 82/14
"PROJET DE NOTE D'INTERPRETATION PHEBUS".
M. ADROGUER

N° 82/15 et 82/16
"BESOINS EN MESURE POUR L'ESSAI PHEBUS 216".
M. ADROGUER

N° 82/17
"ANALYSE DE LA PHASE DE REMPLISSAGE RENOYAGE DE L'ESSAI PHEBUS 103"
MM. SENEMEAUD - DUBOIS - Mme POUBLAN

N° 82/18
"CALCUL DE L'EVOLUTION DU COUPLAGE ENERGETIQUE AU COURS DU TRANSITOIRE"
M. HUEBER

N° 82/19
"RAPPORT PRELIMINAIRE DE DEPOUILLEMENT ESSAI PHEBUS 213 A"
B. CLEMENT

N° 82/20
"FICHE TECHNIQUE 82/14 SEAREL
PROPOSITION D'EXAMENS APRES IRRADIATION DE LA PERCHE PHEBUS 2504"
M. ADROGUER - Mme BERNE

N° 82/21
"CALCULS PREVISIONNELS EXPERIENCE 215 P"
MM. ADROGUER - PIGNARD

N° 82/22
"ETUDE DE LA SENSIBILITE DANS RELAP 4 AUX CHOIX DES CORRELATIONS DE DEBIT CRITIQUE D'ECHANGE THERMIQUE ET DE FLUX CRITIQUE. APPLICATION A LA SIMULATION DE PHEBUS".
MM. PERRET - PIGNARD

N° 82/23

"RAPPORT PRELIMINAIRE DE DEPOUILLEMENT ESSAI 103"

MM. HUEBER - SIMONNEAU

N° 82/24

"REFROIDISSEMENT PREMATURE DU COMBUSTIBLE DANS LES ESSAIS PHEBUS DOUBLE BRECHE SUR 25 CRAYONS : ANALYSE DES CAUSES ET REMEDES POSSIBLES"

MM. SCOTT DE MARTINVILLE

N° 82/25

"RAPPORT PRELIMINAIRE DE DEPOUILLEMENT ESSAI 212"

Melle COURT - MM. MESTRE - SIMONNEAU

N° 82/26

"RAPPORT PRELIMINAIRE DE DEPOUILLEMENT ESSAI 210".

Melle COURT

N° 82/27

"RAPPORT DE DEPOUILLEMENT ESSAI 210"

Melle COURT

VIII - DEGRE DE DISPONIBILITE DES DOCUMENTS PUBLIES

Les documents des listes précédentes marqués d'un astérisque (*) sont diffusables tels quels.

TABLEAU 1
GRILLE GENERALE DES ESSAIS - PREMIERE PARTIE

NUMERO DE L'ESSAI	PARAMETRE CRAYON			PARAMETRE DOUBLE		PARAMETRE INJECTION			DATE DE REALISATION
	NOMBRE	PRESSION	P.LIN.MAX	LOCAL. BRECHE	POS. VAEP 14	DEBIT	LOCAL.	RETARD	
		M PA	KW/M			L/H			
101	1	0,1	0	BC + BF	F	160	BF	0	24.06.80
101 BIS									03.12.80
102	1	0,1	57	BC + BF	F	160	BF	0	17.12.80 14.04.81
103	1	0,1	57	BC + BF	F	160	x	θ_G - 1000°C	28.10.81
104	1	0,1	57	BC + BF	F	160	x	θ_G - 1200°C	
105	1	-	R	-	-	-	-	-	-
206	25	0,1	0	BC	0	2600		0	24.04.80
207	25	0,1	27	BC	0	2600		0	03.06.80
208	25	0,1	0	BF	0	2600	BF	0	22.05.80
209	25	0,1	27	BF	0	2600	BF	0	10.06.80
210	25	0,1	0	BC + BF	F	2600	BF	0	04.06.81
211	25	0,1	27	BC + BF	F	2600	BF	0	09.09.81
212	25	0,1	40	BC + BF	F	2600	BF	0	26.11.81
213	25	0,1	40	BC + BF	F	2600	BF	x	10.02.82
213 A	25	0,1	40	BC + BF	F	ACCUS	BF	-	24.06.82
213 B	25	0,1	57	BC + BF	F	1100	BF	+	21.10.82
213 E	25	0,1	47	BC + BF	F	1600	BF	+	18.11.82
213 C	25	0,1	47	BC + BF	F	1600	BF	+	25.11.82
213 F	25	0,1	57	BC + BF	F	1600	BF	+	17.12.82
215 P	25	4,0	48	BC + BF	F	1600	BF	+	08.07.82

TABLEAU 2

PROGRAMME REALISE

LISTE DES ESSAIS ET DES PERCHES UTILISEES

NUMERO DE L'ESSAI	PARAMETRES CRAYON			DATE DE REALISATION	N° DE PERCHE
	NOMBRE	PRESSION	P. LIN. MAX		
		M Pa	KW/m		
101	1	0,1	0	24/06/80	H 0101
101 bis				03/12/80	0101
102	1	0,1	57	17/12/80	0101
				14/04/81	0103
103	1	0,1	57	28/10/81	0103
104	1	0,1	57	-	-
105	1	-	R	-	-
206	25	0,1	0	24/04/80	H 2501
207	25	0,1	27	03/06/80	2501
208	25	0,1	0	22/05/80	2501
209	25	0,1	27	10/06/80	2501
210	25	0,1	0	04/06/81	H 2502
211	25	0,1	27	09/09/81	2502
212	25	0,1	40	26/11/81	2502
213	25	0,1	40	10/02/82	2503
213 A	25	0,1	40	24/06/82	2503
213 B	25	0,1	57	21/10/82	2503
213 E	25	0,1	47	18/11/82	2503
213 C	25	0,1	47	25/11/82	2503
213 F	25	0,1	57	17/12/82	2503
215 P	25	4,0	48	08/07/82	2504

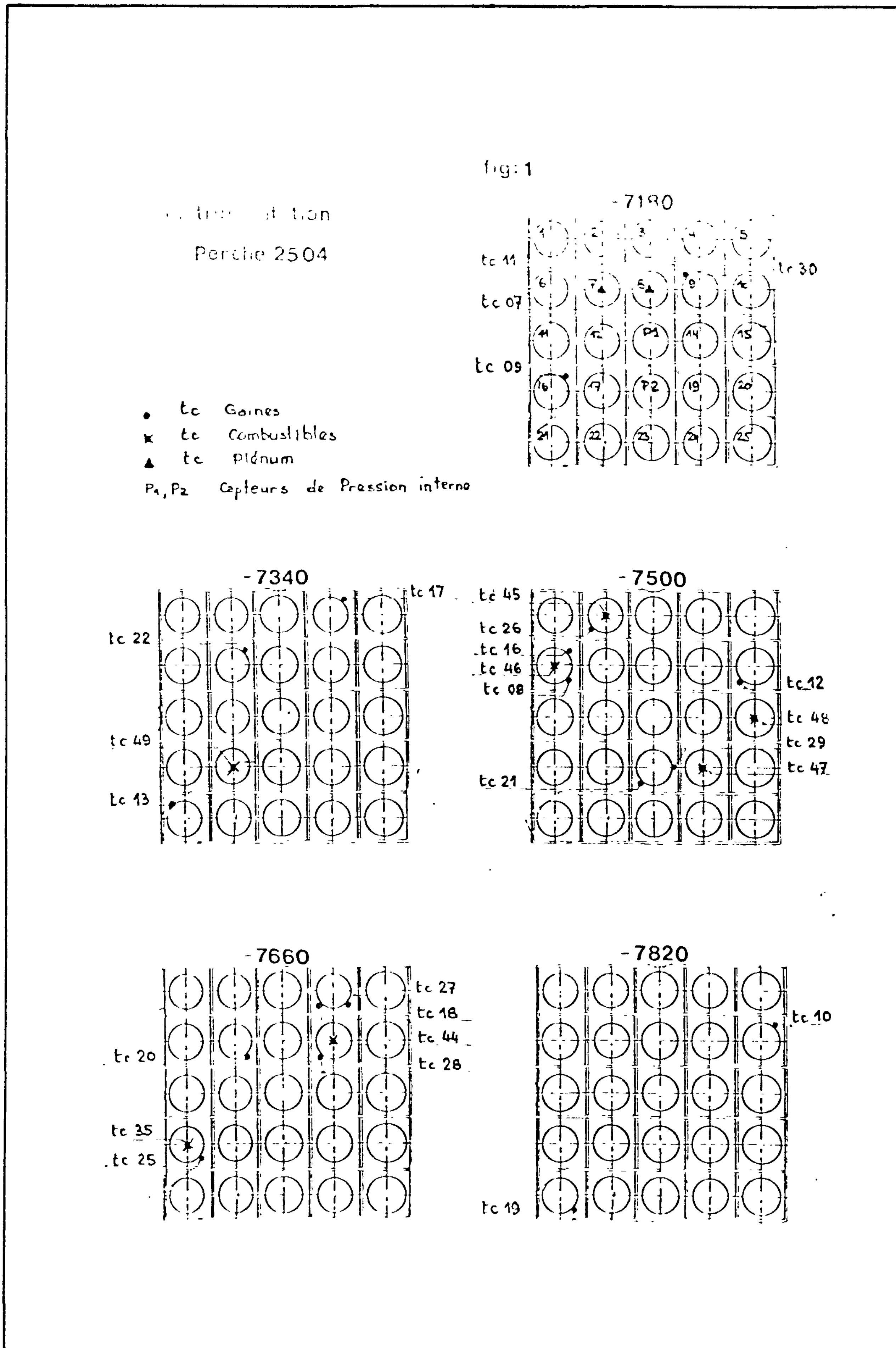


fig: 1
Perche 2504
tc Gaines
tc Combustibles
tc Plénum
P1, P2 Capteurs de Pression interne
-7180
tc 11
tc 30
tc 07
tc 09
P1
P2
-7340
tc 17
tc 22
tc 49
tc 13
-7500
tc 45
tc 26
tc 16
tc 46
tc 08
tc 12
tc 48
tc 29
tc 47
tc 21
-7660
tc 27
tc 18
tc 44
tc 28
tc 20
tc 35
tc 25
-7820
tc 10
tc 19

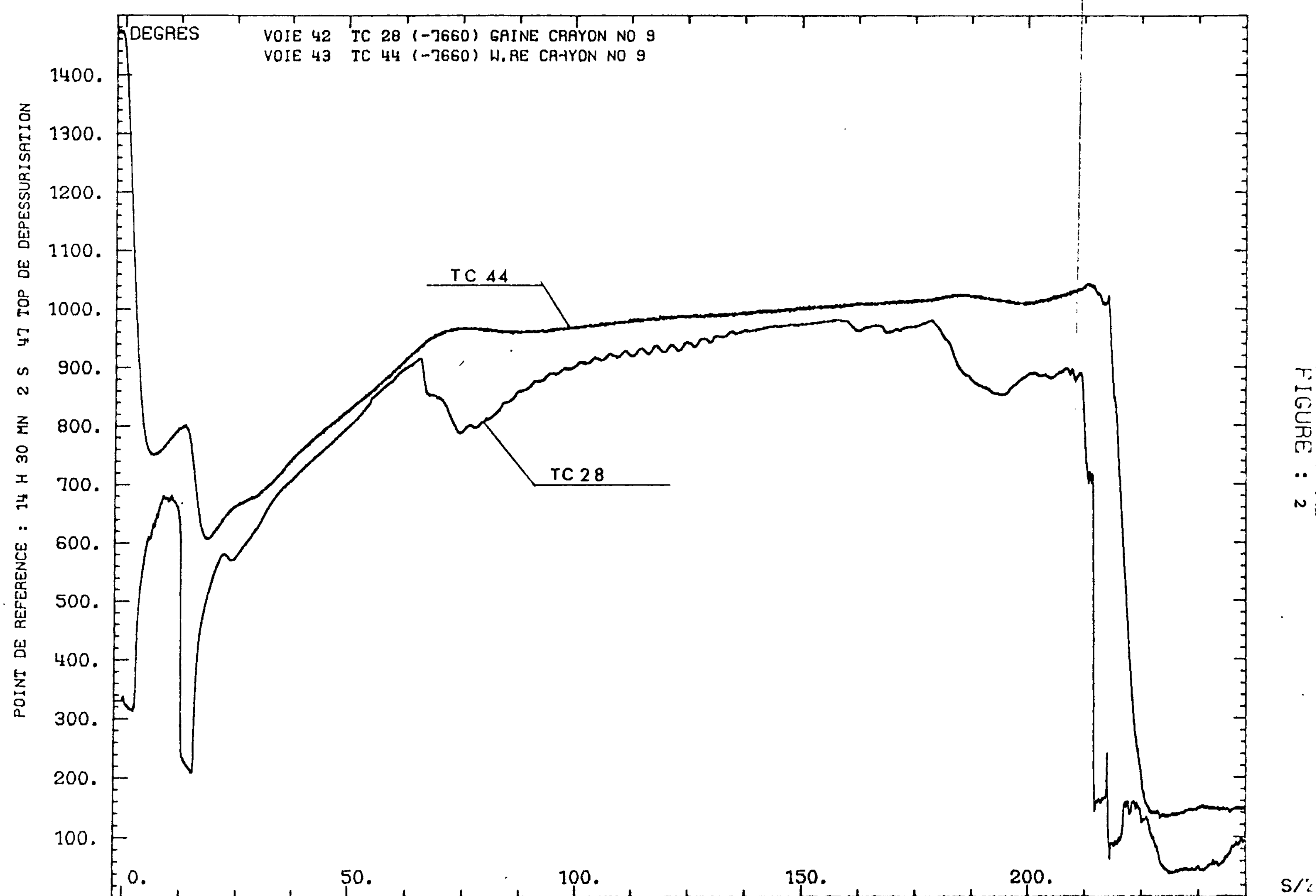

DEGRES
VOIE 42 TC 28 (-7660) GAINE CRAYON NO 9
VOIE 43 TC 44 (-7660) W.RE CRAYON NO 9
POINT DE REFERENCE : 14 H 30 MN 2 S 47 TOP DE DEPESSURISATION
1400.
1300.
1200.
1100.
1000.
900.
800.
700.
600.
500.
400.
300.
200.
100.
0.
50.
100.
150.
200.
TC 44
TC 28

FIGURE : 2

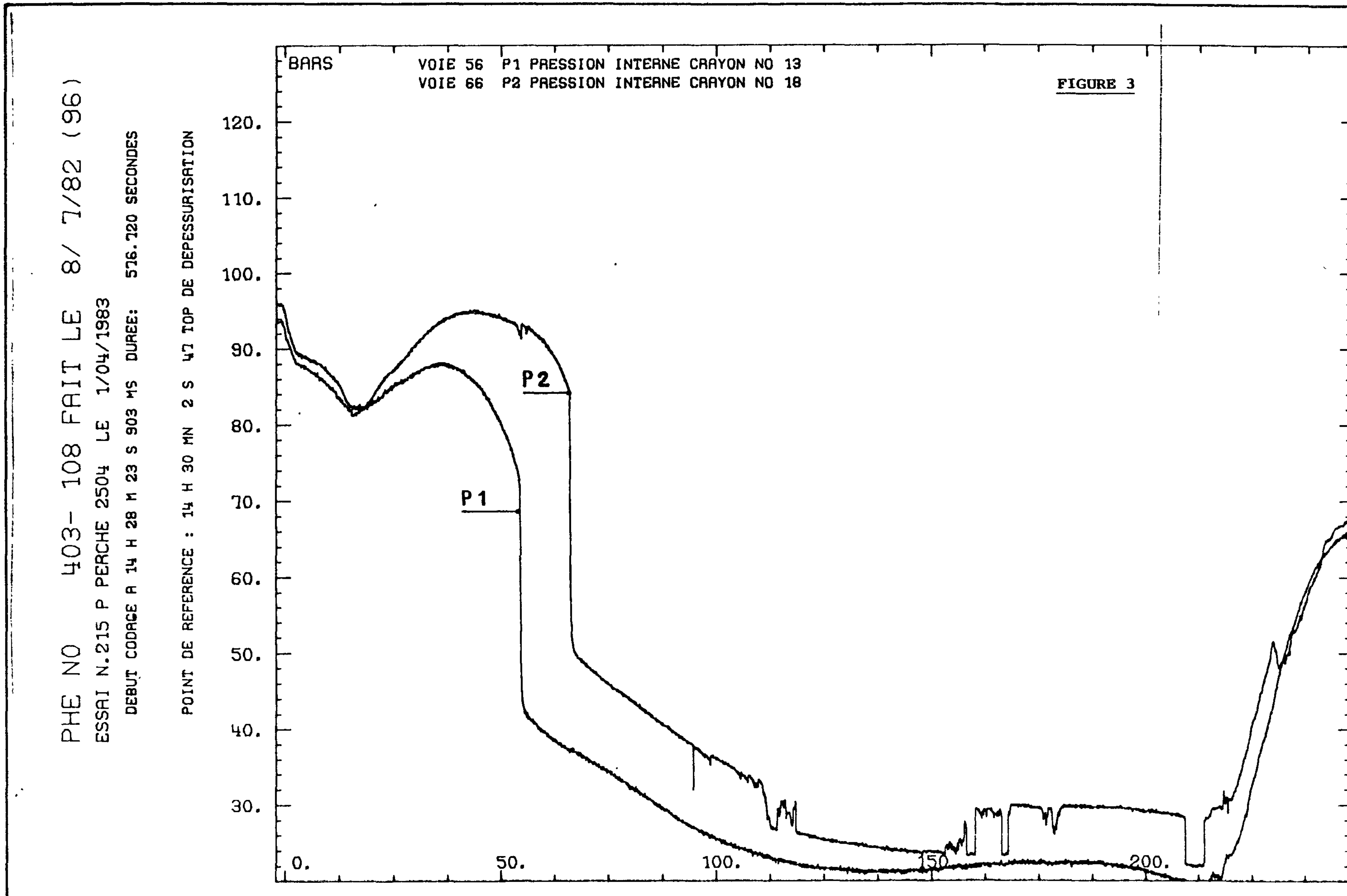

PHE NO 403- 108 FAIT LE 8/ 7/82 (96)
ESSAI N.215 P PERCHE 2504 LE 1/04/1983
DEBUT CODAGE A 14 H 28 M 23 S 903 MS DUREE: 576.720 SECONDES
POINT DE REFERENCE : 14 H 30 MN 2 S 47 TOP DE DEPESSURISATION
BARS
VOIE 56 P1 PRESSION INTERNE CRAYON NO 13
VOIE 66 P2 PRESSION INTERNE CRAYON NO 18
FIGURE 3
120.
110.
100.
90.
80.
70.
60.
50.
40.
30.
0.
50.
100.
150
200.
P 1
P 2

FIGURE 4

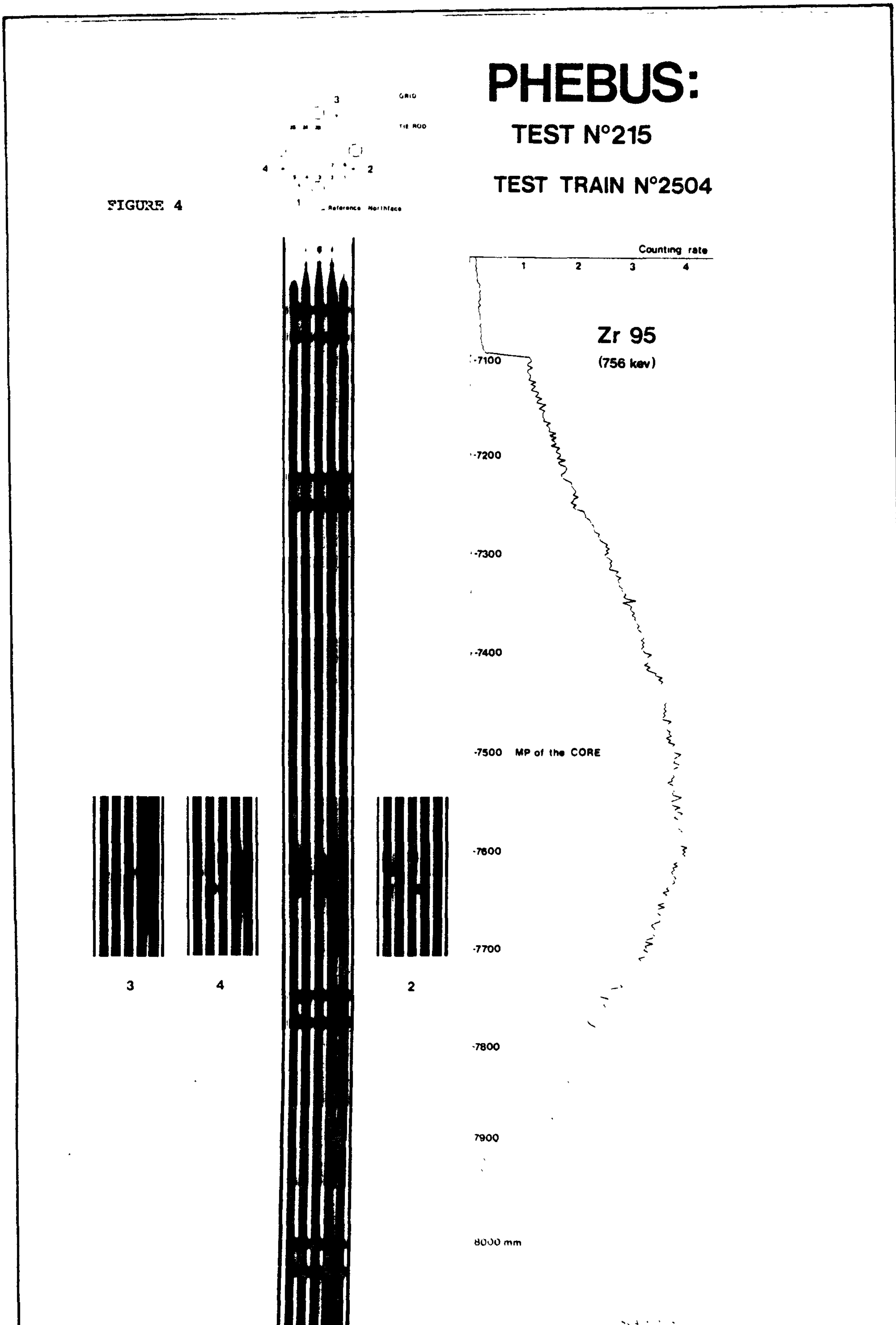
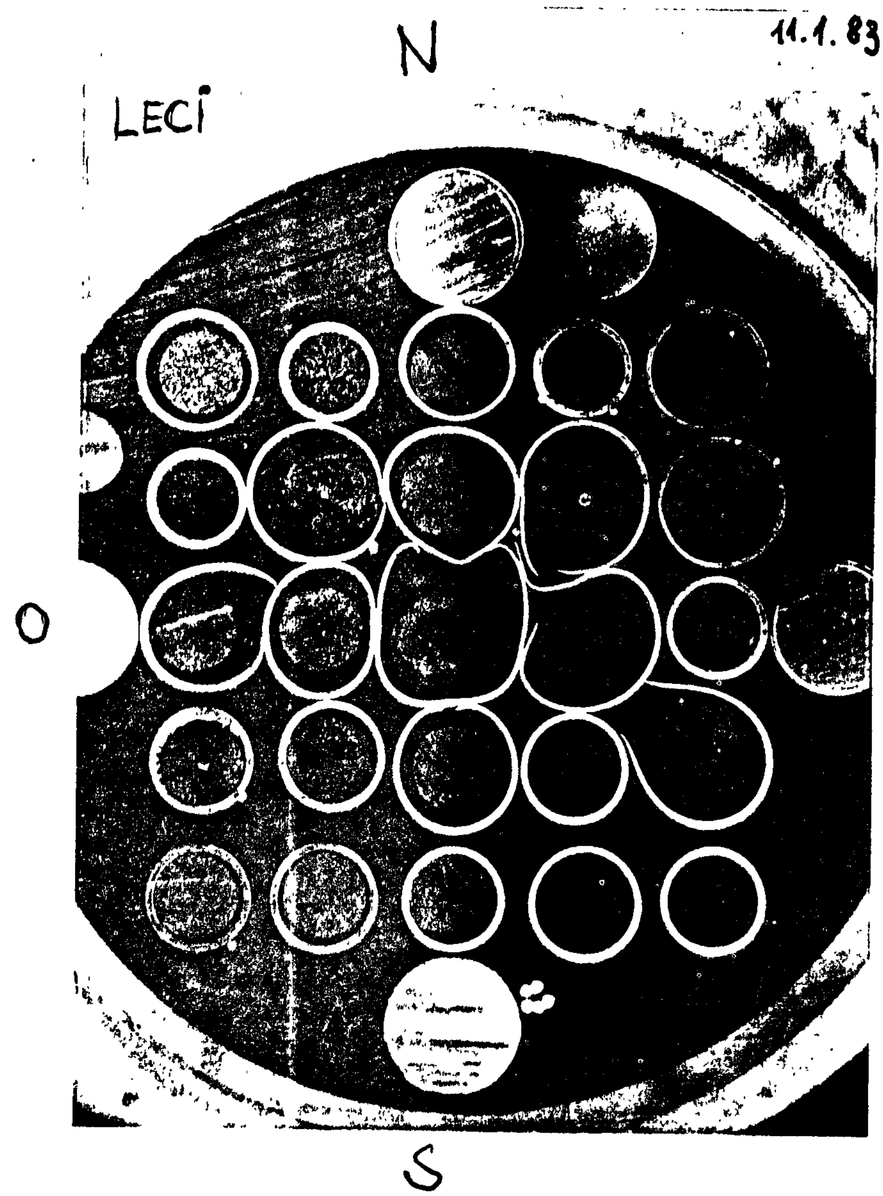

PHEBUS:
TEST N°215
TEST TRAIN N°2504
GRID
TIE ROD
Reference Northface
Counting rate
1
2
3
4
Zr 95
(756 kev)
-7100
-7200
-7300
-7400
-7500 MP of the CORE
-7600
-7700
-7800
7900
8000 mm
3
4
2

FIGURE N° 5



Perche 2504 Cote - 7650

170.2.01			1 - 3
TITRE PROGRAMME PHEBUS CSD ETUDE DE LA DEGRADATION DU COMBUSTIBLE D'UN REACTEUR PWR SOUMIS A DES CONDITIONS ACCIDENTELLES AU DELA DES CRITERES DE SURETE			*Pays* FRANCE *Organisme directeur* IPSN *Organisme éxécuteur* IPSN/DERS
TITRE en anglais PHEBUS SFD PROGRAM STUDY OF SEVERE FUEL DAMAGE UNDER ACCIDENTAL TRANSMENTS OVER SAFETY CRITERIA			*Responsables* M. TATTEGRAIN M. REOCREUX
Date de démarrage 01/01/82		*Etat actuel* Etudes de Projet	*Scientifiques* M. ARNAUD M. BERNA
Date d'achèvement 31/12/89		*Dernière mise à jour* Première Fiche Etude	

I. - OBJECTIF GENERAL

L'étude de la dégradation du coeur d'un réacteur soumis à des conditions accidentelles sévères, au délà des critères retenus pour le dimensionnement, a le double objectif suivant :

1/ - examiner les problèmes liés à la maîtrise de tels accidents,

2/ - évaluer les conséquences radiologiques de ces accidents.

L'étude comporte deux volets :

- le comportement du combustible,
- le relâchement et le dépôt des produits de fission.

Elle implique la mise au point de modèles physiques, établis sur la base d'expériences phénoménologiques hors pile, validés sur des expériences globales en pile, et transposés au cas réacteur.

Le programme PHEBUS C.S.D a pour objectif le déroulement d'expériences en pile, de caractère global, destinées à valider des modèles physiques décrivant le comportement du combustible pour des niveaux de températures supérieurs à 1200°C, jusqu'au point de fusion de l'oxyde d'uranium.

L'étude du relâchement et du dépôt des produits de fissions concerne les expériences FLASH, conduites par DMECN/DRG à Grenoble.

II. - OBJECTIFS PARTICULIERS DU PROGRAMME PHEBUS

Quatre grands objectifs peuvent être distingués, à savoir :

a) - (DERS - SES)

Modification de l'installation PHEBUS permettant de soumettre une grappe de 21 crayons à des conditions représentatives d'accidents réacteurs types. Compte tenu des caractéristiques actuelles de l'installation, la représentativité de la totalité de la séquence accidentelle n'est pas recherchée. On limite la représentativité "en temps" à la période correpondant à un niveau de température supérieur à 1200°C. De même, la grappe d'essai n'est représentative que d'une petite zone du coeur réacteur. Dans cette fenêtre espace-temps représentative, on cherche à réaliser un environnement de la grappe aussi proche que possible de celui qui règne dans un coeur en situation accidentelle.

Cet environnement est déterminé à l'aide des codes de calculs disponibles à ce jour et qu'il s'agit, in-finé, d'améliorer.

Cet objectif implique le développement d'un programme R et D axé sur la tenue des matériaux et de l'instrumentation à haute température et en milieu oxydant.

b) - (SEAREL - SES)

Définition d'une grille d'essais et réalisation de ces essais dans des conditions définies par le Comité Programme PHEBUS. Préparation par le calcul de ces essais et dépouillement.

c) - (SEAREL)

Mise au point de moyens de calcul nécessaires pour :

- l'élaboration précise du programme,
- la prévision et l'interprétation des phénomènes,
- leur transposition au réacteur.

d) - (SEAREL)

Définition et suivi d'expériences hors-pile destinées à la mise au point de modèles physiques utilisés dans les codes de prévisions et d'interprétation. Ces études sont conduites principalement par DTECH et DTCE/STT.

III. - INSTALLATION EXPERIMENTALE ET PROGRAMME

III.1 - L'installation est composée de 3 parties :

a) un réacteur source du type piscine, d'une puissance thermique de 40 MW,

b) une boucle d'essais, utilisée pour des essais du type LOCA, constituée d'une cellule disposée dans l'axe vertical du coeur et d'un circuit externe de refroidissement assurant une circulation d'eau préssurisée aux conditions nominales d'un réacteur PWR,

c) une boucle expérimentale (boucle CSD), objet des modifications de l'installation indiquées au paragraphe précédent.

Cette boucle comprend 3 parties :

- une perche contenant le combustible d'essais, disposée dans la la cellule en pile et contenant l'ensemble des dispositifs destinés à l'alimentation en vapeur surchauffée de la grappe d'essais,

- un appareillage hors pile pour alimentation en eau, hydrogène et gaz neutre de la perche,

- une ligne de sortie des gaz (vapeur d'eau, hydrogène, gaz de fissions) qui traverse la piscine du réacteur et conduit à l'appareillage de régulation en pression et de mesures expérimentales.

Les modifications envisagées permettent l'utilisation de l'installation, soit pour des expériences du type LOCA (phases I et II du programme), soit pour des expériences "CSD" (phases III et IV).

Par ailleurs, l'installation comprend un dispositif de manutention permettant l'introduction de la perche expérimentale dans la cellule, son stockage sur site après essai et son transfert vers les laboratoires d'examens post-mortem.

III.2 - Le programme expérimental n'est défini à ce jour que pour la phase III du programme caractérisée par des températures de gaine inférieures à 1850°C. Sept expériences sont prévues, portant sur du combustible vierge, à partir de septembre 1985. Le rythme expérimental prévu est de 4 expériences/an.

Les essais planifiés à ce jour correspondent à quatre séquences accidentelles prenant en compte l'indisponibilité des injections de secours dans les cas suivants :

- transitoire LOCA grosse brèche,
- transitoire LOCA petite brèche,
- perte prolongée de l'alimentation en eau du générateur de vapeur,
- coeur découvert quelques jours après l'arrêt du réacteur.

IV. - ETAT D'AVANCEMENT

IV.1 - L'étude de faisabilité phase III a montré la possibilité d'adaptation de l'installation à la réalisation des essais de phase III, conformément aux conditions de la matrice d'essai expérimentale, excepté en ce qui concerne :

- la pression expérimentale limitée à 0,5 MPa
- la température de la vapeur au bas de la grappe limitée à 1000°C.

IV.2 - Une maquette hors pile (ARTEMIS), grandeur nature, de la cellule et de la perche d'essai a été définie, de manière à vérifier le bon fonctionnement de l'ensemble des dispositifs destinés à produire le débit de vapeur surchauffée en bas de grappe.

IV.3 - Une recherche sur les revêtements protecteurs a été entreprise dans le but de protéger de l'oxydation les structures supérieures de la perche et les gaines de thermocouple.

IV.4 - Un code de calcul décrivant la thermique de la grappe d'essai et des structures qui l'entourent est en cours de développement.

IV.5 - Les premiers calculs de neutronique montrent la possibilité d'utiliser des crayons d'essais avec un enrichissement voisin de celui adopté pour les essais LOCA.

La configuration actuelle du coeur nourricier PHEBUS n'apporte aucune limitation particulière au déroulement des essais du type CSD.

IV.6 - La matrice des essais de phase III a été définie en utilisant le code BOILK dans la description des phases accidentelles mentionnées en 3.2.

V. - PROCHAINES ETAPES

V.1 - Programme PHEBUS phase III

V.1.1 - Les essais sur maquette hors pile ARTEMIS sont prévus pour Juillet 1983.

V.1.2 - Les études sur les modifications de l'installation doivent conduire à l'approvisionnement de matériels à long délai de livraison en décembre 1983

L'arrêt de l'installation PHEBUS pour modifications est prévu pour Septembre 1984

V.1.3 - Le premier essai de phase III est prévu pour Septembre 1985.

V.2 - Programme PHEBUS phase IV

V.2.1 - Aucun code de calcul ne permet à ce jour de définir une grille d'essais pour la phase IV du programme. Une ébauche de grille sera définie fin 1983 sur les bases de l'extrapolation des conditions d'essais de phase III.

V.2.2 - Fin 1983 le projet définira les limitations technologiques à la réalisation des essais de phase IV d'une part, le programme R et D à entreprendre dans ce domaine d'autre part.

V.2.3 - Dans ces conditions le premier essai de phase III ne peut avoir lieu avant fin 1986.

V.3 - Interprétation des expériences

La stratégie à suivre pour développer un code de calcul pour interprétation des expériences de phase III et IV est à définir pour la fin du premier semestre 1983 et le développement du code en 1984-1985.

VI. - DOCUMENTS DE REFERENCE

1982 - "Définition des paramètres de la pré-étude de faisabilité relative aux phase III et IV du programme PHEBUS"
NOTE TECHNIQUE SES 05/82 - M. ARNAUD

"Pré-étude de faisabilité sur la phase III du programme PHEBUS".
NOTE TECHNIQUE SES 10/82 - M. ARNAUD, M. DEL NEGRO, M. DUTRAIVE.

1983 - "Phase III du programme PHEBUS - Maquette hors-pile pourtests de perches simplifiées".
NOTE TECHNIQUE SES 17/82.

"Etude de séquences accidentelles avec dégradation du coeur sur un réacteur PWR".
NOTE TECHNIQUE SAER/83 - M. EVRARD.

"French severe fuel domage program at the PHEBUS facility".
NOTE TECHNIQUE SAER 83/350 - M. DUCO.

	CLASSIFICATION : 1.3
TITLE (ORIGINAL LANGUAGE) : Ribagnamento di foderi per caduta di film liquido e condizioni di flooding.	COUNTRY : ITALY
	SPONSOR : E N E A
TITLE (ENGLISH LANGUAGE) : Falling film shrouds rewetting and flooding conditions	ORGANISATION : E N E A
	PROJECT LEADER : G. E. Farello

INITIATED :	COMPLETED :	SCIENTISTS :
March 1978		M. FURRER
STATUS : In progress	LAST UPDATING : Oct. 1981	

1) General Aim

Measure of the rewetting velocity and of the sputtering and flooding characteristics.

2) Particular Objectives

Types of flow (rivulet, minimum thickness, sputtering zone, ect.) and droplet characteristics measurements in dipendence of flowrate subcooling, initial wall temperature. Impingement laws of droplets on heated walls (precursory cooling). Flooding conditions and steam droplets thermic interaction.

3) Experimental Facility

"Ad hoc" test section with visualization devices.

4) Project Status

A set of measurements has been completed.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION :
Falling film shrouds rewetting and flooding conditions.	1.3

5) Next Steps

Flooding conditions: counter-current critical flow of steam and water; steam upwards influence on the quench-front velocity; behaviour droplets in the sputtering region.

7) Reference Documents

1) M. CUMO, G.E. FARELLO, M. FURRER
Experimental Remarks on "Sputtering" Phenomena and Droplets. Generation in Falling Film Rewetting. CNEN-RT/ING (80) 2.

2) M. CUMO, G.E. FARELLO, M. FURRER
Observation on the Impingement of Droplets on Heated Walls in Emercency Core Cooling. CNEN-RTE/ING (82) 12.

8) Contact Person : Ing. G.E.FARELLO
E.N.E.A. (Casaccia) - TERM/ISP
C.P. 2400, 00100 R O M A

9) Additional Information

Personnel involved: 1 researcher, 1 technician
Budget: 50 million Lit (up to 1983, excluding salaries).

	CLASSIFICATION : 1.3
TITLE (ORIGINAL LANGUAGE) : Studio del fenomeno del ribagnamento di superfici ad alta temperatura con particolare riferimento alla formazione di rivoli.	COUNTRY : ITALY SPONSOR :
TITLE (ENGLISH LANGUAGE) : Studies on the rewetting of high temperature surfaces, with reference to the tendency to form rivulets	ORGANISATION : University PALERMO(+) PROJECT LEADER : E. OLIVERI
INITIATED : 1981 / COMPLETED :	SCIENTISTS : F. CASTIGLIA, S. TAIBI, G. VELLA
STATUS : In progress / LAST UPDATING : May 1982	

(+) ISTITUTO DI APPLICAZIONI E IMPIANTI NUCLEARI

DESCRIPTION: Our previous research program devoted to the rewetting of high temperature surfaces has brought about good results in the evaluation of dimensionless velocity of an uniform falling water film [1, 2, 3, 4].
The aim of the present work is:
- set up an improved theoretical model in order to account for the possible arising of rivulets;
- find suitable methods for eliminating or, at least, limiting the arising of rivulets.

As for the first objective a numerical two-dimensional analysis has been carried out and some interesting results have been already gained.

REFERENCES:

[1] E. OLIVERI-F. CASTIGLIA-S. TAIBI-G. VELLA: '' Relazioni finali sulla 1a, 2a e 3a fase dei lavori relativi al contratto NUCLITAL-UNIVERSITA' DI PALERMO su: Refrigerazione di emergenza dei noccioli dei reattori BWR '' 30/6/1979 - PALERMO -

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Studies on the rewetting of high tempera ture surfaces, with reference to the tendency to form rivulets	1.3

[2] F. CASTIGLIA-E. OLIVERI-S. TAIBI-G. VELLA: '' Sulla valutazione della velocita' di ribagnamento di superfici ad elevata temperatura '' 34° Congresso Nazionale ATI - Palermo, Ottobre 1979

[3] E. OLIVERI-S.R. TAIBI-G. VELLA-F. CASTIGLIA: '' Un'analisi numerica bidimensionale del ribagnamento di superfici calde '' Ingegneria nucleare, n. 1, 21-30 - 1981 -

[4] E. OLIVERI-F. CASTIGLIA-S. TAIBI-G. VELLA: '' A new correlation for quench front velocity '' Int. J. of Heat and Mass Transfer (in press)

BUDGET: 10.800.000 Lit

PERSONNEL INVOLVED: 4 Researchers .

Università - Istituto di Applicazioni e Impianti Nucleari
Parco d'Orleans, I- 90128 Palermo

	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 :	SCIENTISTS : G. OLIVETI, A. SABATO, M. CUCUMO
STATUS : In progress / LAST UPDATING : JUNE 1982	

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

Experiments have been performed on a tubular test section having an inside diameter of 10 mm and a heated length of 3000 mm instrumented with 5 thermocouples located at various axial positions. The mode of cooling has been that of bottom

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TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Rewetting of Fuel Rods during E.C.C.S.	1.3

flooding. Mass flow rates between 6 and 40 $g/cm^2 s$, and initial rod temperatures of 500 and 600 °C have been used. The experimental wall temperature profiles have been analized, in order to obtain the rewetting times and the heat transfer coefficients.

5. - Next steps

Experiments in annular geometry with a cosine axial heat flux will follow.

6. - Relation to other project and codes: none

7. - References documents

A. Sabato
Tesi di Laurea: Modelli di ribagnamento per le barrette di combustibile di LWR nel caso di incidente
Jun, 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 counter current flow of liquid and gas in a flooding regime
Atto del Dipartimento 28, May 1979, presented also: European two phase flow group Meeting at Ispra, June 1979

V. Marinelli - G. Oliveti - A. Sabato
Studio sperimentale dello scambio termico parete-fluido per un tubo simulante una barra di combustibile di reattore nucleare ad acqua durante il raffreddamento di emergenza mediante allagamento dal basso,
Atto del Dipartimento 50, June 1981

8. - Additional information

Staff: 4 scientists, 4 technicians.

Università della Calabria, Dip. di Meccanica, I-87030 Arcavacata di Rende, Cosenza.

	CLASSIFICATION : 1.3
TITLE (ORIGINAL LANGUAGE) : Analisi dello scambio termico in condizioni incidentali nel nocciolo PWR in dipendenza del livello del refrigerante.	COUNTRY : ITALY SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : Uncovered Heat Transfer and Thermal Non-Equilibrium	ORGANISATION : ENEA PROJECT LEADER : G. PALAZZI

INITIATED :	COMPLETED :	SCIENTISTS :
1980	1982	A. ANNUNZIATO V. LOMBARDI
STATUS : in progress	LAST UPDATING : April 1982	

1. General Aim

The small LOCA (as TMI-2 accident) can reduce the water inventory in the core and it can cause a situation in wich the coolant flowrate is so low that the liquid level may be below the active length of the core and a particular type of post dry-out heat transfer dominates the temperature regime in the uncovered region. Two-phase flow regimes, liquid swelling as well as liquid entrainment in this region have been investigated.

2. Particular Objectives

Post dry-out heat transfer has been investigated with particular focus on the entrained droplets population and a droplet size distribution law has been derived by means of the coolant temperature measurements.
Moreover transition boiling heat transfer has been examinated in order to link the boiling region and the post C.H.F. region.
A strong swelling phenomenon was observed and therefore void fraction profile investigation has been performed to interpretate the liquid level height.

3. Experimental Facility and Programme

The research has been performed employing a simple loop(called STPA). The test section is made on Inconel Tube and has an inside diameter of 12.6 mm and a uniformly heated length of 3600 mm. Wall thermocouples are placed axially every

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Uncovered Heat Transfer and Thermal Non-Equilibrium	1.3

100 mm; bulk thermocouples, of special design, every 200 mm. The mass flowrate ranges from 4 to 10 Kg/m^2s, the surface heat flux from 9 to 25 kW/m^2, the inlet subcooling from 10 to 80°C; the pressure is at the atmospheric value.

4. Project Status

Critical qualities at the dry-out point and post dry-out heat transfer coefficients have been systematically measured and compared with the available correlations (for different ranges of variables) obtaining indications of the respective applicability.

5. Next Steps

Other about 30 test are necessary to complete the first experimental campaign.

6. Reference Documents

a) A. Annunziato, V. Lombardi, R. Mazzucco, A. Pasqualini, M. Sica, "Strumentazione della sezione di prova mediante brasatura ad induzione per esperienze di Scambio Termico in Parete Asciutta", CNEN(81)16/VBD-8/AA14-2.
b) V. Lombardi "Circuito per Esperienze di Scambio Termico in Parete Asciutta (STPA), CNEN(81)23/VBD-11/AA14-3

7. Degree of Availability

Free.
Contact Person: G. Palazzi - TERM-RISIL,ENEA,CSN-Casaccia, CP 2400, I-00100 Roma

8. Additional Information

Staff: 2.5 man·year
Budget: ≃30 ML

	CLASSIFICATION : 1.1.2/1.2/1.3/7.2
TITLE (ORIGINAL LANGUAGE) : ATTIVITA' DI RICERCA FINALIZZATA NEL CAMPO DELLA TERMOIDRAULICA DEI TRANSITORI CONSEGUENTI A LOCA PER IL REATTORE PWR ENEL UNIFICATO	COUNTRY : ITALY SPONSOR : ENEL
TITLE (ENGLISH LANGUAGE) : FYNALIZED RESEARCH ACTIVITY IN THE FIELD OF THERMO-HYDRAULIC TRANSIENTS FOLLOWING LOCAS FOR THE ENEL UNIFIED PROJECT PWR REACTOR	ORGANISATION : ENEL PROJECT LEADER : G. TREBBI
INITIATED : 1980 / COMPLETED :	SCIENTISTS : L. BELLA, F. DONATINI
STATUS : IN PROGRESS / LAST UPDATING : May 1982	

1. General aim

The research has the purpose to develop an extensive know-how about the thermal-hydraulic aspects of LOCA'S transients for the ENEL reference PWR plant. The following aspects will be investigated:

a) Primary system thermal-hydraulic behaviour
b) Fuel assembly thermal behaviour (refill - reflood phase)
c) Containment behaviour/ECCS performance.

In the next years such three aspects could be splitted up in three distinct research activities.

2. Particular objectives

Qualification of RELAP 4 codes chain for LOCA'S analyses applied to PWR ENEL reactor (EM/Best estimate calculations), extended to the refill-reflood transients. The analyses will cover a great range of breaks (large DE-small) and ECCS performance, under various hypotheses. Comparisons with experimental results will be made with OECD/CSNI Standard Problems program.

The containment will be analized with CONTEMPT LT and COMPARE MOD1 codes and the effect of various emergency systems will be investigated.

	CLASSIFICATION : 1.3
TITLE (ORIGINAL LANGUAGE) : Beräkning av den maximala kapslingstemperaturen med förbättrad strålningsmodell	COUNTRY : Sweden SPONSOR : The Swedish Nuclear Inspectorate
TITLE (ENGLISH LANGUAGE) : A calculation of the peak clad temperature with improved grey-body factors	ORGANISATION : ETU Energiteknisk Utveckling AB PROJECT LEADER : Peter Bergquist
INITIATED : 1982-07-21 | COMPLETED : 1983-07-31	SCIENTISTS : Peter Bergquist
STATUS : In progress | LAST UPDATING : June 1983	

1. A single-node grey-body factor matrix which is used in licensing calculations assumes that the reflected radiation is evenly distributed around the entire rod circumference (The radiation can be "reflected through a rod"). Single-node grey-body factors overestimates the radiation in a BWR-cluster.

2. The project should result in two calculations of the peak clad temperature. One with a single-node grey-body factor matrix used and the other with a multinode matrix.

3. The MOXY-code is used. The code is modified for calculation of a cluster with octant-symmetry.

4.1 The project is completed except for the final report.

4.2 The new model increased the peak clad temperature with 67 ^{o}F. (2179 ^{o}F to 2246 ^{o}F)

8. Peter Bergquist, ETU, Box 4022, 102 61 STOCKHOLM, Sweden

9. An essential progress is the use of multinode grey-body factors for calculation of view-factors. These view-factors could be compared to others, for instance those calculated with an anisotropic factor (used in the TRAC-code)

	CLASSIFICATION : 1.3
TITLE (ORIGINAL LANGUAGE) : Deltagande i CSNI-PWG2 - "Task Group on Fuel Behaviour under Design Basis Accident Conditions" i Paris 22-23 mars 1983	COUNTRY : Sweden
	SPONSOR : Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE) : Participation in CSNI-PWG2 - "Task Group on Fuel Behaviour under Design Basis Accident Conditions" in Paris March 22-23, 1983	ORGANISATION : Studsvik Energiteknik AB
	PROJECT LEADER : K Malén

INITIATED : 1983-04-27	COMPLETED :	SCIENTISTS : K Malén
STATUS : Completed	LAST UPDATING : June 1983	

1. General Aim: Fuel Behaviour under Design Basis Accident Conditions.

2. Particular Objectives: Participation in CSNI-PWG2 Task Group meeting in Paris March 22-23, 1983

4. Project status: Completed

7. Reference Documents: Malén K, Möte: CSNI - "Principal working group 2 on transients and breaks - Task group on fuel behaviour under design basis accident conditions i Paris 22-23 mars 1983. STUDSVIK/NF(P)-83/23 (In Swedish)

8. Degree of availability CSNI-member

	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

INITIATED:	COMPLETED:	SCIENTISTS:
March 1978		K G Pearson M K Denham
STATUS: In progress	LAST UPDATING: August 1980	

Background

Experimental studies of heat transfer and hydraulics during the reflood stage of a PWR LOCA are being carried out in the Heat Transfer Laboratory of the UKAEA Atomic Energy Establishment at Winfrith in collaboration with the development and validation as part of the UKAEA Programme.

Objectives

The objectives are to provide high quality benchmark data relating to the reflooding of undistorted and ballooned clusters which can be used for validation and improvement of computer codes used in the assessment of reactor safety and an improved understanding of the physical mechanisms involved.

Test Facilities

The experimental Programme employs two major test facilities.

REFLEX (Reflood Experiments) Electrically heated internally flood tubes

THETIS (Thermal Hydraulic Emergency Cooling Test Installation)
Electrically heated full length fuel cluster simulation for studies at pressures up to 70 atmospheres

Programme

(a) Single Tube Studies of Reflood Heat Transfer/Hydraulics - REFLEX

(i) Undistorted Subchannel - Completed

(ii) Ballooned Subchannel - 1980

(iii) Effects of Zircaloy Oxide (EEC supported) - 1981

(b) <u>Cluster Experiments - THETIS</u>

(i) <u>Blocked Cluster Test</u>

7 x 7 Ballooned Cluster (90% local blockage) to demonstrate coolability (12.2mm pins) - 1981

(ii) <u>ECCs Studies</u>

8 x 8 Undistorted Cluster (9.5mm pins) - 1982

(iii) <u>Ballooned Cluster Test</u>

8 x 8 Ballooned Cluster (9.5mm pins) (EEC supported) - 1983

	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: J H Gittus (SNL)
INITIATED: March 1978 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: July 1980	

Background

In a loss of coolant accident following a large breach the rapid circuit depressurisation can lead to high clad temperatures and pin distension and so to cooling channel blockage. The slow depressurisation associated with a small breach can lead to partial core uncovery, adiabatic heat up of the pin and clad temperatures at which chemical reaction, gas release and even component melting may occur.

Objectives

These are to calculate the core response to various types of LOCA (different positions and size of break) using the transient core specific code MABEL-2. This entails interfacing with a total circuit thermal hydraulic code eg RELAP and a performance model eg SLEUTH which fixes the steady state condition of the fuel pin at the moment of the transient. Factors such as pin rating history, proximity of cold control tubes etc. will be taken into account. Validation work on MABEL-2 and CANSWEL-2 will continue.

Present position

The clad deformation code CANSWEL-2 has been integrated into the MABEL-2 so that azimuthal variations in temperature, clad thickness, fuel-clad gap size affecting the behaviour of the centre pin of a 3 X 3 array can now be treated. (This work is in collaboration with AEE Winfrith). The CANSWEL-2 code has been developed to the stage when it can model the bulging of cladding as it expands between neighbouring pins.

Facilities

The computers installed at Springfields, Harwell and Risley are being used for this work.

Reference Documents

BOWRING R W, COOPER C A, GITTUS J H, HASTE T J.
A Code to analyse cladding deformation in a loss-of-coolant accident : status. IAEA Specialists Meeting on Water Reactor Fuel Element Performance Computer Modelling, Blackpool, UK, March 1980.

	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: R W BOWRING
STATUS: IN PROGRESS / LAST UPDATING: JULY 1980	

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 Spring 1981
" " MABEL-3

FACILITIES

MABEL is programmed for the IBM 3033 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 (1978).

2. R W Bowring, C A Cooper, J H G Hus and T J Haste. "MABEL-2: A code to analyse cladding deformation in a Loss-Of-Coolant Accident: Status February 1980" AEEW-M1766 (1980).

	CLASSIFICATION: 1.3
TITLE (ORIGINAL LANGUAGE): Thermal and hydraulic performance of partially uncovered cores	COUNTRY: United Kingdom
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Winfrith)
	PROJECT LEADER: J Fell

INITIATED:	COMPLETED:	SCIENTISTS:
March 1978		K G Pearson M K Denham
STATUS: In progress	LAST UPDATING: August 1980	

Background

Experimental studies of the thermal and hydraulic performance of partially uncovered cores relevant to PWR small breach accidents (eg TMI) are being carried out in the Heat Transfer Laboratory of the UKAEA Atomic Energy Establishment at Winfrith in collaboration with code development and validation as part of the UKAEA Programme.

Objectives

The objectives are to provide high quality benchmark data relating to the thermal and hydraulic performance of partially uncovered cores which can be used for validation and improvement of computer codes used in the assessment of reactor safety and an improved understanding of the physical mechanisms involved.

Test Facilities

THETIS (Thermal Hydraulic Emergency Cooling Test Installation)
Electrically heated full length fuel cluster simulation for studies at pressures up to 70 atmospheres.

Programme

Cluster Level Swell Experiments - THETIS

(i) 61 Pin Cluster (SGHWR pins 12.2mm diameter) - Completed

(ii) Ditto - improved instrumentation - 1980

(iii) 7 x 7 Ballooned Cluster (PWR Array 12.2mm pins) 1981

(iv) 8 x 8 Undistorted Cluster (PWR Array 9.5mm pins) 1982

	CLASSIFICATION: 1.3
TITLE (ORIGINAL LANGUAGE): Clad deformation and multi-rod interaction effects in a LOCA	COUNTRY: United Kingdom
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Springfields)
	PROJECT LEADER: J H Gittus (SNL) J B Sayers (Harwell)

INITIATED:	COMPLETED:	SCIENTISTS:
March 1978		
STATUS: In progress	LAST UPDATING: September 1980	

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

Previous studies of the deformation behaviour of single rods[1,2] will be continued, mainly to produce data for, and verification of, the transient code MABEL. The effects of interaction with adjacent, deforming rods, and with control rod guide tubes will be determined using 4 X 4 assemblies of short rods, and 6 X 6 assemblies o full-length rods. A rig, MULTIROD, is being designed for PWR rods, about 750 mm long, in up to 9 X 9 arrays to test interaction during heating and reflooding.

The feasibility of a single rod LOCT (loss of coolant test) in the DIDO reactor high pressure water loop has been established,[3] in which rod behaviour can be compared with out-of-reactor simulation. The heat generation, heat loss, and rise in temperature are designed to be similar to that which might be experienced by an inter-grid span of one fuel rod in a PWR assembly. Both radiative and convective cooling can be available. The present work is the preparation of costed studies for construction and for the test series so that a decision can be made early in 1981. If the results of the technical assessment, safety submission and costing are acceptable the first experiment could take place in 1983/4.

Facilities

PROPAT rig (single rod tests) FLEET and MULTIROD rigs (multi-rod tests) at Springfield High Pressure Water Loop in Harwell DIDO reactor.

Clad deformation and multi-rod interaction effects in a LOCA

Reference Documents

1. ROSE K M, MANN C A and HINDLE E D
The axial distribution of deformation in the cladding of PWR fuel rods in a loss-of-coolant accident.
Nuclear Technology 46 (2) 220-227 (1979)

2. HINDLE E D and MANN C A
The experimental study of the deformation of Zircaloy PWR fuel element cladding under mainly convective cooling.
Fifth International Conference on Zirconium in the Nuclear Industry, Boston, Mas, USA. August 1980.

3. MARLOW B, LANG C and CRABBE C J
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 programme, Petten, June 1979.

2. CORE MELTDOWN

Berichtszeitraum/Period 1.1. - 31.12.1982	Klassifikation/Classification 2	Kennzeichen/Project Number RS 474
Vorhaben/Project Title Analytical Activities Analysis of Core Meltdown		Land/Country FRG Fördernde Institution/Sponsor Auftragnehmer/Zuwendgsempf./Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH
Arbeitsbeginn/Initiated 1.5.1980	Arbeitsende/Completed 31.12.1982	Leiter des Vorhabens/Project Leader Dr. Friederichs/Dr. Scharfe
Stand der Arbeiten/Status completed	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

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.

In particular:

2.1 Review and analysis of research reports concerning the meltdown process, the interaction between molten core material and reactor concrete and the integral description of the accident.

2.2 Verification of computer codes, concerning the different stages of a Hypothetical meltdown accident.

2.3 Review of the meltdown process and behavior of molten material after hypothetical accidents (LMFBR).

2.4 Review of Fuel-Coolant-Interactions.

2.5 Review of behavior and cooling of fuel debris beds.

2.6 Review of behavior of a molten pool (LMFBR).

2.7 Review of interactions between core melt and core-catcher materials or concrete.

2.8 Review of behavior of radioactive inventory inside the plant (LMFBR).

3. Research Program

3.1
- Assessment of the results of research projects concerning the different stages of a hypothetical meltdown accident.
- Development of models for special phenomena.

- Identification of uncertainties or problems concerning the meltdown accident that should be eliminated or solved.

3.2 • Acquisition, review and assessment of computer codes as well as computed results for verification of the statements made in advanced risk studies.
- Post-test analyses of selected experiments.
- Improvement of computer codes and computer models.

3.3 Review of activities concerning meltdown and the behavior of molten materials in the core.

3.4 Critical assessment of experiments and analytical methods to investigate the fuel coolant interaction; investigation of the applicability to reactor conditions.

3.5 Review of experimental and analytical activities concerning fuel debris which are carried out in foreign countries and as RS-projects. Application of this knowledge to LMFBRs. Extension of the investigation to temperatures above dryout and below melting.

3.6 Critical assessment of experiments with simulant materials and reactor materials to investigate the heat transfer from molten pools.

3.7 Transfer of experiences which have been gained for LWRs to LMFBRs. Review and assessment of experiments which simulant materials and reactor materials.

3.8 Calculation of the inventory of radioactive materials, of the plate-out and transport behavior of Plutonium and fission products. Evaluation of release-pathways and of the amount of released radioactivities from the plant (LMFBR).

4. Experimental Facilities, Computer Codes

-

5. Progress to Date

for 3.3 With regard to an integral risk assessment the knowledge about
to 3.7 the behavioar of the core materials after the core disruption has been completed.

6. Results

for 3.3 to 3.6 LMFBRs have very favourable properties for the retention of core materials inside the reactor vessel after a core disruption. If those structures which could be covered with core materials are designed appropiately, it can be excluded in most cases that the reactor vessel will be penetrated. Retention structures which are cooled from the bottom by sodium are a very favourable design. Molten core materials cannot melt through such structures. Some uncertainties still consist in the properties of steel at very high temperatures. Some other uncertainties are of a more phenomenological nature:

- if a jet of liquid core materials can reach a retention structure, this might result in an extreme thermal load at the stagnation point,
- the penetration length of liquid core materials inside cold channels can be estimated only roughly. This causes uncertainties about the progression of the accident,
- fuel coolant interactions with considerable mechanical work potential are very unlikely under reactor conditions, but they cannot be excluded completely.

for 3.7 In the fields of fuel-concrete interactions and sodium fires considerable progress has led to a certain understanding of these phenomena. For the problem of sodium-concrete interactions conservative assumptions still have to be made at the present state of knowledge /1/.

7. Next steps: The project ist completed.

8. Relations to other projects: -

9. References

/1/ G. Bönigke, P. Bogorinski, K.-H. Martens: Chemische und thermische Wechselwirkungen bei hypothetischen Kernschmelzunfällen in schnellen natriumgekühlten Reaktoren GRS-A-734 (Aug.82)

10. Degree of Availability of the reports

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

	CLASSIFICATION : 2 (7, 14)
TITLE (ORIGINAL LANGUAGE) : Projekt FILTRA Filtrerad tryckavlastning av reaktorinneslutning Projektstudie av typkonstruktion	COUNTRY : SWEDEN
	SPONSOR :
TITLE (ENGLISH LANGUAGE) : Project FILTRA Swedish Vent-filter conceptual design study	ORGANISATION : SNPI, NBESD, utilities*
	PROJECT LEADER : Kjell Johansson and Lars Nilsson

INITIATED : 1980 Febr 1	COMPLETED : Nov 1982	SCIENTISTS :
STATUS : Completed	LAST UPDATING : June 1983	

OBJECTIVE: Study risk reduction potentially obtainable from application to reactor containments of vent-filter conceptual designs. The primary objective is to reduce land contamination from reeases due to large core melt and containment overpressurization accidents.

APPROACH: Theoretical investigations of containment events plus some experimental work on crushed rock stonebed and sand filters.

PROGRESS: Final report in Nov 1982

INTENDED USE OF RESULTS: The Swedish government requires a vent-filter system to be installed at Barsebäck and be operative before Sept 1, 1986 (3). Modifications of the other ten existing Swedish operating LWR reactors containments, if assessed to entail subtantail risk reduction, can be required with a view to achieve effective risk reducing meansures before end of 1989.

SHORT DESCRIPTION OF PROJECT: The project is to be executed in three stages, which comprise:

* Swedish Nuclear Power Inspectorate and National Board for Energy Source Development, respectively.

Stage 1: A. Prelimimary design criteria.

B. Preliminary decisions about pressure relief and filtration equipment.

Stage 2: C. More detailed design criteria.

D. Type design.

Stage 3: E. Complementary, verifying and concluding items.

The first stage started early in 1980 and will continue till the beginning of 1981. The results of the first stage of work was reported in (2). The second and third stages of work were reported in Nov 1982. Experiments have been performed on the retention of aerosols and iodine in crushed rock beds and also on the pressure drops and steam condensation in such beds.

REFERENCE DOCUMENTS:

1 FILTRA report 2 Progress March 1981
Filtered Atmospheric Venting of LWR containments
Available from:
Studsvik Library
S-611 82 NYKÖPING
SWEDEN Telex: 64070 stu bib s

2 Summary of the present Swedish licensing and regulatory position on prevention and mitigation of radioactive releases in the case of severe accidents.
Swedish Nuclear Power Inspectorate 1982-01-22
Box 27106
S-102 52 STOCKHOLM
SWEDEN Telex: 11961 sweatom s

3 Kjell Johansson, Lars Nilsson, Åke Persson
FILTRA Design Considerations for implementing a containment vent-filter plant at Barsebäck, Sweden. Paper intended for presentation at ANS Chicago conf. Aug 29 - Sep 2, 1982

4 FILTRA Final Report nov 1982
Available from:
Studsvik Library
S-611 82 NYKÖPING
SWEDEN Telex: 64070 Stu bib s

2.1. MOLTEN MATERIAL BEHAVIOUR

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82-31.12.82	2.1	06.01.16/07A (PNS 4250)
Vorhaben/Project Title Langzeitkühlung im stark beschädigten Core (COLD) Long Term Coolability of a Heavily Damaged Core (COLD)		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated 01.01.1981	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader DI. G. Hofmann
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

In order to better understand the possible courses of severe accidents with core degradation the coolability of debris is investigated. It is believed that such improved understanding is of benefit to plant design as well as to risk analysis and accident management.

2. Particular Objectives

Under certain accident conditions a debris consisting of degraded fuel rods and structural material can be generated. The reavailability of coolant and heat sink does not necessarily mean that debris temperatures are under control. The long term coolability of such a debris bed is investigated to learn if bed temperatures stay below the melting temperatures of the components and no further change of the geometry takes place even if the cooling boundary conditions become worse after an initial rewetting is completed.

3. Research Program

It does not seem possible at the present time to relate certain accident paths with geometrical characterizations of the degraded core material for every time during the accident. But since also the knowledge about the coolability limits and about the dryout mechanisms in idealized beds is still incomplete the program tries to close these gaps without having the full information about the geometry of real core debris. The program consists of dryout experiments (out-of-pile) in deep beds with water as coolant in order to

- investigate the influence of the bed height (up to 1 m) on the dryout heat flux
- investigate the influence of the hydraulic boundary conditions (mainly bottom inlet flow by natural convection through a permeable

support plate)
- clarify the dryout mechanisms by measuring the dryout locations and the axial pressure and saturation distributions in the bed.
- verify existing models and improve them where necessary and possible.

4. Experimental Facilities, Computer Codes

For induction heating of the particulate beds a radio frequency generator with 60 kW and 250 kHz at the heating coil is available. The data acquisition system which is used reads up to 40 channels from steady state or slowly transient experiments.

5. Progress to Date

The more important results of the feasibility study were summarized in Ref. 1. The dependence of the dryout heat flux from the bed depth was investigated with a series of 111 additional dryout tests. In these tests also the location of the first dry spot was measured; a computer model for predicting this location was developed and compared with the experimental data. Bottom inlet flow experiments were conducted also; the results were compared to an existing model.
An additional outer circuit of the RF generator for heating of more than 1 m deep beds was installed and tested.

6. Results

The dryout heat flux was found to be independent of bed depth for top-fed beds of 3 mm-spheres when they were deeper than 250 mm, though an influence up to 400 mm was reported earlier due to the misleading interpretation of a very small data base. It was found that even with 3 mm-particles long-term boiling causes minor geometrical bed changes with an increased dryout heat flux as a consequence.
The evaluation of the top-fed experiments resulted in an improved understanding of dryout as the end of a hydraulically transient process characterized by a time-dependent saturation change in the bed which precedes the appearance of the first dry spot. The dryout location was measured for the first time as a function of the power step beyond dryout. A quasi-steady-state computer model was developed to describe this process and to predict the location of the first dry spot in the bed. Model and experiment are in reasonable agreement for particle beds with small capillary forces /2/. An analysis of the differential

equation describing the saturation distribution in the 1-dim. model with capillary forces has shown the possibility to apply the dryout location model to small diameter particle beds also.
A comparison between the model of Ref. 3 with experimental data from this program for bottom inlet flow by natural convection revealed a remarkable discrepancy. The dryout heat flux and the bottom inlet mass flux are calculated by far too low. It is believed that this is due to the way in which the axial pressure gradient is calculated in the model and that the channeled flow assumption without interfacial momentum exchange is not adequate for the lower portion of the bed.

7. Next steps

Since the prediction of the pressure difference within the bed seems to be a weak point of existing dryout models, future experiments with up to 1 meter deep beds are planned to explore this point; they will be set up to investigate mainly combined top-fed and bottom-fed situations and will include efforts to measure the saturation distributions.

8. Relation to the other Projects

PNS 06.01.08, PSB 01.02.15. Information exchange with similar investigations at SANDIA, ANL, UCLA and Westinghouse; followup of the severe fuel damage experiments in SUPER SARA, PBF and ACRR.

9. References

/1/ G. Hofmann, H. Schneider; Langzeitkühlung im stark beschädigten Core, PNS-Jahresbericht 1981, KfK 3250 (1982)

/2/ G. Hofmann; On the location and mechanisms of dryout in top-fed and bottom-fed particulate beds. Proceedings of the Int. Information Exchange Mtg. on Post Accident Debris Cooling, Karlsruhe (July 1982)

/3/ R.J. Lipinski; Bottom-Fed Deep Debris Bed; Adv. Reactor Safety Research Quarterly Rep., Jan.-Mrch. 81, NUREG/CR 2238 (1982)

10. Availability of Reports

unrestricted, PNS-Project

	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: - Oct.1981	SCIENTISTS: Mr.T.A.Dullforce
STATUS: Continuing / LAST UPDATING: May 1982	

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.

	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: - Oct.1981	SCIENTISTS: Mr.T.A.Dullforce
STATUS: Continuing / LAST UPDATING: May 1982	

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.

39.507

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.

		CLASSIFICATION : 2.1
TITLE (ORIGINAL LANGUAGE) : Småskaliga studier av fissionsaerosoler		COUNTRY : Sweden SPONSOR : RKS, SKI, Nuclear Power Insp
TITLE (ENGLISH LANGUAGE) : Small-scale studies of fission product aerosols		ORGANISATION : Chalmers Technical University PROJECT LEADER : Jan Rydberg
INITIATED : May 1982	COMPLETED :	SCIENTISTS : Jan Rydberg Oliver Lindqvist
STATUS : In progress	LAST UPDATING : June 1983	

1. General Aim: A large international program is underway to use the Marviken full-scale facility for studying the transport of aerosols and fission products in the reactor primary system after a core-melt accident. The chemistry of the aerosol source may be an important factor, that cannot be studied parametrically in the full-scale experiments, where source and environmental conditions have to be grossly simplified. In order to provide guidance for the simplifications, small-scale aerosol generation experiments are being carried out, where the source composition and the chemical environment are varied.

6. Relation to Other Projects and Codes: Marviken-V International Aerosol Transport Test Program

8. Degree of Availability: No restrictions.

		CLASSIFICATION : 2.1
TITLE (ORIGINAL LANGUAGE) : MARVIKEN AEROSOL TRANSPORT TESTS		COUNTRY : Canada, Finland, France, Italy, Japan, Netherlands, Sweden, UK, USA.
		SPONSOR : International project
TITLE (ENGLISH LANGUAGE) : MARVIKEN AEROSOL TRANSPORT TESTS		ORGANISATION : STUDSVIK ENERGITEKNIK AB
		PROJECT LEADER : Jan Collén
INITIATED : February 1982	COMPLETED : August 1985	SCIENTISTS :
STATUS : In progress	LAST UPDATING : June 1983	

General aim

Investigation of the behaviour of material released by a nuclear reactor in a core melt accident.

Particular objectives

The primary purpose of these tests is to create a large scale data base on the behaviour of vapors and aerosols produced from overheated core materials within typical LWR primary systems for risk-dominant scenarios. The program has a second objective which is to provide a large scale demonstration of the behaviour of aerosols in primary systems.

Experimental facilities and programme

The pressure vessel of the Marviken Station in which will be located a simulated reactor vessel designed for the test. Simulated pressurizer and relief tank will also be included in the test facility.

The program comprises tests with fissium simulants as well as corium simulants. The first tests will include only fissium and at a later stadge tests with both corium and fissium aerosols will be performed.

Project status

Experimental program in progress.

Next step

Perform the tests according to the experimental program.

Additional information

Additional information about this test programme can be obtained from STUDSVIK ENERGITEKNIK AB, The Marviken Project, S-611 82 Nyköping, Sweden.

2.2. FUEL/COOLANT INTERACTION

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.82 - 30.11.82	2.2	150 371
Vorhaben/Project Title Entwicklung eines Modells zur thermischen Detonation Development of a Thermal Detonation Model		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Universität Stuttgart Institut für Kernenergetik u. Energiesysteme
Arbeitsbeginn/Initiated 15.3.79	Arbeitsende/Completed 30.11.82	Leiter des Vorhabens/Project Leader Prof.H.Unger/DP M. Bürger
Stand der Arbeiten/Status ----	Berichtsdatum/Last Updating ----	Bewilligte Mittel/Funds

1. General Aim

Within the frame of the research project investigating core meltdown problems of light water reactors theoretical calculations on hypothetical vapor explosions in light water reactors are performed. The theoretical activities initiated shall lead to a better comprehension on conditions, course and extend of vapor explosions, which may possibly occur during a hypothetical core meltdown accident.

2. Particular Objectives

The aim of the research project can be subdivided into the following tasks:

- Development of a calculational model in order to describe the self-sustained propagation of a detonation wave. It is found that a large scale thermal detonation wave may be a necessary condition to get significant vapor explosions.
- Extension of the model to transient events in order to investigate the pressure rise until the self-sustained pressure propagation will be reached (detonation wave).
- Development of calculational models (if not done by other works) to describe the fragmentation process and the fine mixing between the fuel and the coolant. The results may be used as input data for the detonation model.
- Investigations on vapor film instability at low pressure pulses in order to get the minimum pressure pulses which may lead to a vapor explosion.

It is intended to perform best-estimate calculations with experimentally obtained particle distributions. The relevant input data will be continuously completed.

3. Research Program

3.1 Development of a Model and a Computer Code to Describe the Self-Sustained Propagation of a Detonation Wave

Modeling of the jump conditions at the shock front in a fuel coolant-dispersion dependent on the initial conditions. Calculation of the pressure propagation, velocities, fragmentation process and heat transfer in the relaxation zone. Determination of the initial conditions which might lead to a self-sustained detonation wave.

3.2 Extension of the Model to Transient Events

Modeling of arbitrary trigger pulses in order to investigate the behavior of unsteady state detonation waves.

3.3 Development of Fragmentation Models

Theoretical investigation of fragmentation models as far as necessary. Description of the fine mixing process between fuel and coolant. Collection of data obtained from experiments as input for best-estimate calculations.

3.4 Investigations on Vapor Film Instability at Low Pressure Pulses

The aim is to get the minimum pressure pulses which may lead to vapor explosions.

3.5 Special Calculations with Reactor-Relevant Materials

The calculations will be made with water as coolant and mainly with Sn, Al and corium as fuels.

3.6 Investigation of Inhomogeneities in the Coarse Premixture

The effect of inhomogeneities shall be analysed by use of the unsteady state model.

3.7 Description of Heat Transfer from the Fragments to the Coolant

A transient, but not instantaneous heat transfer shall be considered.

3.8 Comparison of Theory and Experiment

Escalating cases within vapor explosion experiments shall be calculated by use of the unsteady state model. The comparison shall be performed for the OECD standard experiments of Fry and Robinson.

4. Experimental Facilities, Computer Codes

In order to calculate the steady state thermal detonation wave for a given set of initial conditions a computer code has been developed. Additionally an unsteady state detonation code has been completed.

5. Progress to Date

An unsteady state model has been developed which describes the propagation and growth behavior of pressure waves within coarse premixtures of melt and coolant. The model can be used to describe the effect of arbitrary trigger pulses. The effect of inhomogeneties in the coarse premixture can be analysed as well.

6. Results

A comparison of theory and experiment was performed for the OECD standard experiment of Fry and Robinson, using the unsteady state model. The results of the model were in good agreement with the experimental findings.

7. Next Steps

Project has been completed.

8. Relation with Other Projects

To describe the fragmentation process in cooperation with Ispra additional theoretical calculations are done. Some of the obtained results are expected to be useful for the detonation model.

9. References

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10. Degree of Availability of the Reports

Gesellschaft für Reaktorsicherheit (GRS) mbH, Köln

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	2.2	150 493
Vorhaben/Project Title Berechnung zur Dampfexplosion im Reaktordruckbehälter Vapour Explosion Calculation in a Reactor Pressure Vessel		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Zuwendgsempf./Contractor: KRAFTWERK UNION AG Reaktortechnik R 233, Erlangen
Arbeitsbeginn/Initiated 01.10.80	Arbeitsende/Completed 30.06.83	Leiter des Vorhabens/Project Leader Dr. Zeitner
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31.12.82	Bewilligte Mittel/Funds

1. General Aim

An analysis is to be carried out to examine whether the dynamic loads on the reactor pressure vessel associated with a vapour explosion could lead to a failure of the containment. The integrity of the containment is in any case ensured if the RPV is not damaged or is only slightly damaged. If there is considerable damage to the reactor pressure vessel or if the vapour explosion takes place after a melt-through of the reactor pressure vessel in the containment basement (after penetration of the lateral concrete wall upon contact with the sump water), the analysis must establish whether the integrity of the containment will be endangered by flying fragments of the steel or concrete structures.

2. Particular Objectives

The recommended examinations will concentrate largely on the following points:

- estimation of the maximum quantity of melt which can be expected to come into contact with water
- calculation of the heat transferred from the metallic melt to the adjacent water
- determination of the load on the reactor pressure vessel, its support structure and the adjacent concrete structures as a result of pressure build-up in the transiently heated water
- determination of structural failure criteria and of maximum sustainable loads
- comparison of the maximum sustainable load with the load resulting from a vapour explosion.

3. Research Program

3.1 Study of the effects of vapour explosions

01.01.82 - 31.12.82 150 493

3.2 Evaluation of a hypothetical vapour explosion

3.3 Evaluation of the large-scale experiments at SANDIA

4. Experimental Facilities

Two calculation models have been used, of which the first (model 1) registers the loads in the upper or lower half of the reactor pressure vessel and the second (model 2) calculates, for a given quantity of heat Q, the duration of Q (t_Q), the reacting quantity of water m_o and the pressure curve on the wall of the reactor pressure vessel.

5. Progress to Date

Re. 3.1 The time history of heat transfer from melt to water has been verified relative to a number of fragmentation classes and to a trigger wave setting off a vapour explosion and its time history.

The computer coding for the thermomechanical model, from the trigger wave setting off the vapour explosion through to the mechanical load on the RPV in all its major parts including dome plate, closure head and RPV suspension, has been completed.

Re. 3.2 Two different water condition models have been integrated in the thermomechanical model. While the first extrapolates in the supercritical regions on the basis of the limiting curve, the second requires reference values for the states of equilibrium under consideration.

As the discharged heat flux is a significant variable, the thermomechanical model has been supplemented with a variable temperature model which supplies the instantaneous heat flux at any moment in the time history computation.

To enable improved evaluation of containment integrity, the mechanical RPV model has been verified and non-linear materials and component behaviour have been simulated.

6. Results

Re. 3.1 Parameter studies with the thermomechanical computer model confirm not only the influence of the mechanical properties of the RPV but also the predominant role of the initial melt/water composition, where the melt particle size in particular is a determining factor for the violence of the vapour explosion and hence for the load on the RPV.

Re. 3.2 In the subcritical region there was a high degree of correlation between the two water models, whereas there are considerable differences in the supercritical region as the distance from the critical point increases.

If single-class fragmentation is specified (final fragmentation = initial fragmentation), the heat transfer yielded by the integrated thermal conductance model is unrealistically high. It was possible to clearly demonstrate the necesstiy for improved model simulation, particularly in respect of mechanical non-linearities for the purpose of containment integrity demonstration.

7. Next Steps

The RPV load limits are to be determined by selective variation of the chief determining factors and the statement of the limit loads is to be verified by applying the heat flux to the computer model in order to integrate Euler's equations of motion. Then the computer model is to be applied to the high pressure path. Stability limits relative to a water vapour explosion under the RPV are to be established.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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	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

Continue 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: September 1980	

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 wa 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 plac cannot foreseeably occur in the PWR. Conditions under which local failure to cool may occur may be achieved by coolant ch 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.

Previews 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.

2.3. EFFECTS OF MOLTEN MATERIAL ON STRUCTURES

Berichtszeitraum/Period 01.01.82 - 31.12.82	Klassifikation/Classification 2.3	Kennzeichen/Project Number 150 377
Vorhaben/Project Title Ergänzende Untersuchungen zum Verhalten von Reaktorbeton in der 4. Phase eines hypothetischen Kernschmelzunfalls Supplementary Investigations on the Behaviour of Reactor Concrete During the 4th Phase of a Hypothetical Core Meltdown Accident		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik B 222, Erlangen
Arbeitsbeginn/Initiated 01.04.79	Arbeitsende/Completed 30.09.82	Leiter des Vorhabens/Project Leader Hr. Kaspar
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating 31.12.82	Bewilligte Mittel/Funds

1. General Aim

Great significance is attached to a detailed analysis of the events of the 4th phase of a hypothetical core meltdown accident. The core melt is in direct contact with the concrete foundation during this phase.

2. Particular Objectives

Among other thins answers to the following questions are required:

- Is the resistance of concrete to thermal erosion and thermoshock impaired by subjection to exterior constraining forces (building load)?
- Will additional steam be released from the concrete within the containment interior during this phase?

Experiments shall be performed to fill this gap in our knowledge. These experiments will determine both the behaviour of concrete subjected to thermoshock and interior stresses an the partial pressure of water vapour from concrete within a temperature range of up to 600 °C.

3. Research Program

3.1 Concrete behaviour subjected to thermoshock an interior stresses

3.2 Determination of the partial vapour pressure of concrete as a function of temperature.

4. Experimental Facilities

Measurement of water vapur partial pressure of concrete takes place in a research autoclave.

5. Progress to Date

Re. 3.2 The partial vapour pressure of reactor concretes with different aggregates (silicates and carbonates), different types of cement (Portland cement and blast-furnaceslag cement), different water/cement ratios and different preparatory treatments has been measured as a function of temperatures. In addition, the partial vapour pressure of Philippsburg reactor plant concrete and of concrete after 6 years storage was investigated.

6. Results

Re. 3.1 The test results show that there is no increase in melt front propagation rate as a result of possible spallation under combined thermal and mechanical loading of concrete (external compressive and tensile stresses up to 12 N/mm^2 and 1.5 N/mm^2 respectively) as compared to purely thermal loading.

Re. 3.2 The partial vapour pressure in all the concretes investigated lies within the range of the p-T diagram which is parameterized as a function of the specific volume of the water vapour, where the specific volume in the case of the concret is determindes by the pore volume of the concrete and the volume of water released. Both the pore volume and the quantity of released water are temperature-dependent and thus determine that the partial pressure characteristic of water vapour from concrete is a function of temperature. In addition, all concretes exhibit a water vapour pressure drop on account of capillary depression (discharge of water from small pores).

7. Next Steps

Work has been completed. A final report will be compiled.

01.01.82 - 31.12.82 150 377

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	2.3	150 379

Vorhaben/Project Title	
Detaillierung von KAVERN und Programmentwicklung zur Gasabströmung aus der Schildgrube Further development of KAVERN and Code Development on Gas Generation from the Containment Basement during Concrete Decomposition	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Zuwendgsempf./Contractor: KRAFTWERK UNION AG Reaktortechnik R 923, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.04.79	31.12.82	Dr. K. Hassmann
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	31.12.82	

The present project encompasses two main topics in the entire accident sequence which will lead to substantially improved results not only for pressure time history in the containment atmosphere but also for the period of the concrete penetration phase.

2. Particular Objectives

2.1 Consideration of the separation of the metallic and oxidic constituents of the melt influences the shape of the cavity. Therefore, a more realistic consideration of the molten iron from the reinforcement in the concrete foundation, the different heat source distribution in metal or oxide, and the gas release (H_2, CO_2, H_2O) is necessary. For the energy and mass balances, the heat transfer between the metallic and oxidic constituents must be known.

2.2 Knowledge is required on the streaming of steam, CO_2 and hydrogen released during the concrete destruction into the containment atmosphere out of the primary circuit as well as through openings in the biological shield, for several reasons.

- Hydrogen burns as soon as it enters the containment atmosphere if its temperature is above the self-ignition point.
- The thermodynamic state of the gas flow must be known for investigation of the mixing of the released gases with the containment atmosphere.
- The surface temperatures of the structures are necessary to determine the deposition of fission products in the primary circuit and in the reactor cavity.

- 2 -

01.01.82 - 31.12.82 150 379

3. Research Program

3.1 Further development of the KAVERN computer code.

3.2 Code development for investigation of the gas flow released during core/concrete interaction.

4. Experimental Facilities

4.1 A first version of the KAVERN computer code was developed and tested for project RS 183.

4.2 No computer code is available for calculating the gas flow released during the concrete destruction at the present time. However, parts of other computer codes are available and shall be used.

5. Progress to Date

Re. 3.1 Test runs with the KAVERN computer code have been completed and all instabilities which arose have been eliminated. The models to calculate the energy discharged via the surface have been improved. A model capable of simulating the long-term phase has been produced. Long-term production runs have also been carried out. In particular, the influence of the silicate concrete with its 3.1 % CO_2 release and of carbonate aggregates with a 33.9 % CO_2 release has been investigated. Containment calculations have been performed with the data computed for energy and mass transfer.

Re. 3.2 Longer computations have been carried out with the STROMI computer code.

6. Results

Re. 3.1 The results obtained with the WECHSL computer code tally very closely with results from the KAVERN code.
The choice of mineral aggregate has a significant influence on penetration behaviour. With the carbonate concrete, the melt shows a tendency to penetrate in a vertical direction; with the silicate concrete, however there is an approximately uniform penetration both horizontally and vertically. With regard to the build-up in the containment, the influence of early ingress of sump water with silicate concrete is more our less counterbalanced by the higher gas release rate with

carbonate concrete in the first few days.

Re. 3.2 The longer computation runs, for which in some cases simplified boundary conditions were chosen, yield plausible results. They suggest that the gas outlet temperatures prior to the ingress of sump water will be in the region of 400 - 500 °C.

7. Next Steps

Work has been completed. A final report will be compiled.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	2.3	150 524

Vorhaben/Project Title	
Untersuchungen zu Containmentintegrität bei Kernschmelzen unter hohem Primärkreisinnendruck Containment Integrity after Core Meltdown	Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor KRAFTWERK UNION AG Reaktortechnik R 921, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.01.81	30.06.83	Dr. K. Hassmann
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	31.12.82	

1. General Aim

Containment load during core meltdown sequences initiated by transient or small LOCA which determine the level of risk in phase B of the German Risk Study (GRS) is to be examinated in greater detail. The Project is also intended to establish whether other events in the course of the core meltdown sequence can have any effect on containment integrity.

2. Particular Objectives

On the basis of the results of the GRS, priority is assigned to hypothetical accident sequences at high system pressures. The chief areas for investigation are the buildup of pressure and temperature in the containment caused by the transmission of energy and mass, and the resultant stresses in the containment shell during selected core meltdown sequences.

3. Research Program

3.1 Investigation of the accident sequence until the onset of primary system blowdown

3.2 Investigation of the accident sequence during blowdown (computer model for blowdown, hold-up capacity, RPV suspension, strength of concrete structure)

3.3 Investigation of the accident sequence after blowdown

3.4 Analysis of containment integrity throughout the accident sequence (conditions in the containment, comparison with MARCH, strength of the containment shell)

3.5 Compilation of results

4. Experimental Facilities

The computer programs developed as part of the core meltdown analysis (e.g. COCMEL, LEVEL, RELAP) are used for this project.

5. Progress to Date

Re. 3.1 The time history of evaporation in the primary and secondary systems until the onset of core meltdown has been calculated for all scheduled incident sequences using the LEVEL program.

Re. 3.2 A model has been developed to simulate discharge through the RPV leak cross-section and the pressure/time history in adjacent compartments.

Re. 3.4 With the aid of the MARCH computer program developed by Battelle Columbus, the time history of thermodynamical conditions in the containment until containment failure has been calculated for loss of all AC power as initiating event with following failure of the secondary heat removal system.

6. Results

Re. 3.1 The course of the subsequent accident sequence (start of core heat-up) and rates of escape into the containment have been established.

Re. 3.4 The MARCH computations for incident sequence a) (loss of all AC power) demonstrated that evaporation in the steam generators is complete 90 minutes after the beginning of the accident. Failure of the RPV is expected to ensue after 230 minutes.

7. Next Steps

Re. 3.2 Program ASTRO is to bė applied to various break sizes in the RPV to establish the pressure/time history in several groups of compartments and the resultant loads on the RPV suspension.

Re. 3.3 Factors such as the degree of fragmentation are to be parameterized as a means of investigating the coolability of the melt in the containment basement, the emptying of the accumulators, contact with the sump water and the resultant pressure build-up in the containment.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1982 - 31.12.1982	2.3	06.01.11/23A (PNS 4314)
Vorhaben/Project Title KONSTITUTION UND REAKTIONSVERHALTEN VON LWR MATERIALIEN BEIM CORESCHMELZEN CONSTITUTIONS AND REACTION BEHAVIOR OF LWR MATERIALS AT CORE MELTING CONDITIONS		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor KERNFORSCHUNGSZENTRUM KARLSRUHE (KfK), PROJEKT NUKLEARE SICHERHEIT (PNS)
Arbeitsbeginn/Initiated 1.1.1974	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader H.Holleck/A.Skokan
Stand der Arbeiten/Status CONTINUING	Berichtsdatum/Last Updating DECEMBER 1982	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 in 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 temperatures 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

5.1 Constitutional investigations in the U-Zr-O system at temperatures between 1500 and 2000°C.

5.2 Thermogravimetric and thermal analysis on simulated core melts with fission product elements and basaltic concrete at temperatures up to 2100°C.

5.3 Examination of a light water reactor fuel rod segment after a 3.9% burn-up; investigation of the fuel and fission product behaviour before and after a simulated transient (1200°C/30 min/Ar) by microprobe analyses of the fuel and of some fission products, partly solved in the fuel and partly precipitated submicroscopically.

6. Results

Reg. 5.1 The eutectical composition in the quasibinary join UO_2-Zr(O) is more rich in UO_2 than previously assumed (15 instead of 5 mole %). In the vicinity of the composition $ZrO_{1.o}$, the cubic suboxide has been found at 2000°C, containing a few percent of U.

Reg. 5.2 During the interaction of the metallic corium melt with basaltic concrete, considerable losses by vaporization are observed at temperatures above 1650°C. The interaction of the oxide corium melt with basaltic concrete is characterized by considerable vaporization at higher temperatures above 1750°C. The fission products are partly carried along with the vaporizing melts.

Reg. 5.3 Within the range of accuracy, no release of Cs, Mo and Zr was found. Nd was slightly released. The release of Xe after isothermal annealing was 8%, after the temperature transient 52% (related to the concentration of generated Xe).

7. Next Steps

Continuing of the constitutional investigations in the U-Zr-O system and of the investigations regarding the fission product release from burnt-up LWR fuel rods with varied heating-up conditions.

8. Relation with Other Projects

PNS 4311, 4315, 4317

9. References

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10. Degree of Availability of the Reports

GRS - FB

Berichtszeitraum/Period 1.1.1982 - 31.12.1982	Klassifikation/Classification 2.3	Kennzeichen/Project Number 06.01.11/24A (PNS 4317)
Vorhaben/Project Title MATERIALKUNDLICHE UNTERSUCHUNGEN IM RAHMEN DER BETAEXPERIMENTE MATERIAL INVESTIGATIONS IN THE FRAMEWORK OF THE BETAEXPERIMENTS		Land/Country Fördernde Institution/Sponsor Auftragnehmer/Zuwendgsempf./Contractor KERNFORSCHUNGSZENTRUM KARLSRUHE (KfK) PROJEKT NUKLEARE SICHERHEIT (PNS)
Arbeitsbeginn/Initiated 1.1.1979	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader A. Skokan
Stand der Arbeiten/Status CONTINUING	Berichtsdatum/Last Updating DECEMBER 1982	Bewilligte Mittel/Funds -

1. General Aim

Theoretical and experimental investigations on chemical interactions between steel melts or thermite melts and concrete in the frame of the BETA experiments.

2. Particular Objectives

- Determination of the experimental aims with particular regard to material problems.
- Metallographic and ceramographic examinations of thermite melt samples.

3. Research Program

- Examination of samples drawn partly from the melt during the BETA experiments, partly from the cut up crucibles after the experiments.
- Investigation of special problems resulting from the continuing research program.
- Investigations of parameters with take an important influence upon the course and consequences of the interaction between core melt and concrete.

4. Experimental Facilities

Laboratory high temperature furnaces, differential thermal analysis, metallography and ceramography, X-ray diffraction, electron microprobe analysis, scanning electron microscope and chemical analysis.

5. Progress to Date

Extended theoretical assessment of thermodynamical data for the reactions having considerable influence on the interaction between core melt and high-density concrete (with iron oxide), the oxidation of the metal components by CO_2 being included. Compilation of the reaction mechanisms.

Fabrication of crucibles from iron oxide yielding heavy-density concrete and reaction and melting tests with a metallic core melt at high temperatures. Preliminary experimental tests of the gas analyses.

6. Results

Thermochemical data of all relevant chemical reactions that are required for the WECHSL code have been achieved and compiled. Reactions between Zr, Cr, Fe and H_2O, CO_2, Fe_3O_4 were considered.

Solidus temperatures werde measured for concrete types containing magnetite (Fe_3O_4) and hematite (α-Fe_2O_3) aggregates. Compared to normal concrete with basaltic or limestone aggregates, these iron oxide concrete types seem to be attacked by the melt more slowly, and the hydrogen production is considerably lowered. The heat ablation has been calculated.

7. Next Steps

- Examination of thermite melt samples.
- Further experimental investigations of the interaction between high-density concrete and a metallic core melt.

In future the results will be reported in PNS 4314.

8. Relation with Other Projects

PNS 4314, 4323, 4325, 4331.

9. References

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10. Degree of Availability of the Reports

GRS-FB.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.1982 - 31.12.1982	2.3	06.01.12/20A (PNS 4325)

Vorhaben/Project Title	Land/Country
Ausbreitung von Stahlschmelzen in Beton - Bestimmung der Schmelzfrontausbreitung. Erosion of Concrete by Steel Melts - Investigation of the Melt Front Velocity.	
	Fördernde Institution/Sponsor BMFT
	Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe GmbH (KfK) Projekt Nukleare Sicherheit (PNS)

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1979	1985	Dr. S. Hagen
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General aim

The investigation of the interaction between molten steel and concrete will provide the basis for the development and verification of computer programs describing the behavior of core melt in contact with concrete.

2. Specific objectives

The work reported will determine the temperature distribution, the moisture behavior and penetration of the melt into the concrete.

3. Research program

Measuring methods will be developed, tested and used for the following tasks:

- temperature distribution
- propagation of dehydration
- propagation of the melt front.

4. Experimental facilities and computer codes

- The temperature distribution will be measured with NiCrNi thermocouples. An essential presupposition regarding the use of the thermocouples is the development of suitable positioning methods. For the transition region between concrete and melt high temperature thermocouples have to be tested which allow the measurement of the melt temperature for at least a few minutes.
- For the determination of dehydration behavior detectors are being developed based on the moisture content dependency of concrete electric resistance.
- The propagation of the melt is determined by failure of the NiCrNi thermocouples.
- The data acquisition system and the code for the evaluation of the measured data was developed and tested.
- An arrangement for the photogrammetric measurement was tested and the codes for the evaluation of the data developed.

5. Progress to date

In the reporting period the testing of measuring procedures for the determination of the temperature distribution and the melt front propagation was continued. The usual thermite melt experiments in US and BETA geometries were performed.
The applicability of NiCrNi thermocouples with 0.5 mm Inconel sheaths has been tested. The cross section of this type is a factor 4 smaller than a thermocouple of 1 mm. This has a distinct advantage with respect to the higher thermal conductivity of the thermocouple in comparison to the concrete.

In experiment M14, W-Re thermocouples were tested. These TCs had an 8 mm diameter protective quartz tube as sheath. The W-Re thermocouples were positioned in the concrete at four different distances from the cavern floor. The test should provide a statement about the life time of these thermocouples.

The applicability of light guides for the determination of the melt front propagation has been tested. The front end of the light guides are embedded at different depths in the concrete. The arrival of the melt front is signaled by the increasing brightness at the front end.

6. Results

The tests with 0.5 mm thermocouples within US and BETA crucibles have shown that thin sheathed thermocouples can be successfully used in large concrete units.

The W-Re thermocouples used had an life time after the arrival of the melt front of less than 5 minutes. The 0.5 mm NiCrNi thermocouples failed immediately upon the arrival of the melt with a temperature rise time in the range of a second. Due to their larger mass the W-Re thermocouples had a response time approximately ten minutes greater.

A comparison of melt front penetration measurements between light guides and thermocouples has shown that the guide tubes can also be used for the determination of melt front propagation. The uncertainty in time determination is larger for the light guides, but the independence from induction fields interaction recommends this method as a back up solution.

01.01.1982 - 31.12.1982 06.01.12/20A (PNS 4325)

7. Work to be done

In the acceptance tests of the BETA-facility the capability of the measuring methods in strong induction fields must be tested.

8. Relation with other projects

This work is performed as part of the Core Melting Program of BMFT.
In addition to the different groups at KfK, KWU and TU Hanover also participate.

9. References

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10. Degree of availability of the reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	2.3	06.01.12/21A, PNS 4323
Vorhaben/Project Title Experimente zur Wechselwirkung zwischen Stahlschmelzen und Beton Experiments of interaction of steel melts and concrete		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK), Projekt Nukleare Sicherheit; Hauptabteilung Ingenieurtechnik
Arbeitsbeginn/Initiated 1.1.1977	Arbeitsende/Completed 31.12.1984	Leiter des Vorhabens/Project Leader D. Perinić
Stand der Arbeiten/Status	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

1. General Aim

The experiments on the interaction of steel melts and concrete are to allow verification and further development of the computer codes relating to the influences exerted by the core melt on the concrete of the reactor.

2. Particular Objectives

Research and development work performed with KfK/IT on the interaction of melt and concrete. Development of test facilities, methods of measurement and procedures of evaluation. Conduct of tests, documentation and evaluation of measured results.

3. Research Program

3.1 Concept, planning, construction and operation of the BETA test facility for experiments on the interaction of steel melts and concrete.

3.2 Development and acquisition of the measuring technology for the facility.

3.3 Planning, preparation, implementation and evaluation of experiments.

3.4 Coordination of construction measures including the licensing procedure.

4. Experimental Facilities

- Thermite melting facility: 15 kg thermite at the maximum.
- Experimental facility for testing the measuring technology: 600 kg thermite at the maximum.

- BETA experimental facility: 300 kg steel at the maximum, 600 kg thermite at the maximum.

5. Process to Date

Work relating to 3.1, 3.2, 3.3 and 3.4 has been carried on.

6. Results

Ref.3.1 The thermite casting facility was subjected to a second testing involving 1000 kg thermite. All components of the facility have fulfilled their functions and can be installed in the BETA facility without modification.
Assembly work in the process engineering zone of the BETA facility was started according to schedule. The first part of assembly work of the cooling system in the pump area was completed. The cooling tower has been erected. The frame of the facility and the vertical lifting platform in the BETA hall have been assembled and preliminary acceptance has been granted.
The first invitation of bids for the exhaust gas system was stopped and the concept modified. One offer has been received for the new concept. The respective modifications of the steel frame have been ordered.

Ref.3.2 Work on planning and development, respectively, of the measurement, control, monitoring and recording technology was continued and intensified. Most of it will be completed by the end of the year.
Preparations for installing the measuring technology in the BETA control room have started. This includes instrumentation of the control panel for the control room, fabrication of electronic wafers for amplifiers, and construction of the equipment for crucible rupture monitoring.
The immersion probe system was granted preliminary acceptance in the workshop of the manufacturer on September 20, 1982.
It has been delivered to KfK. The supports for the maintenance positions of the immersion probe system in the BETA hall have been mounted.

The 10 t balance was modified by the manufacturer in order to attain the required accuracy of measurement. The acceptance was made using weights (10 t) of the Bureau of Standards in Karlsruhe on June 9, 1982.

Ref.3.3 During the period of reporting eight BETA preliminary tests were performed with a view to develop the measuring technology and the crucible technology, five of them true to scale.

In test M102 the curcuble and cap were made true to BETA scale. 600 kg of thermite powder were ignited in the crucible. The exhaust gas was evacuated via a 10 m long lined pipe. The experiment served to test the crucible reinforcement, crucible monitoring system, cap pressure measurement system, periscope cooler with ratio pyrometer, two periscope dummies, refractory lining, and gas measurement system.
Beginning with concrete crucible B1 all crucibles are made with a steel mold guaranteeing true to scale fabrication so that the crucibles fit the crucible support. For crucible lining the more temperature resistant furan resin was chosen to embed glass fiber mats, as an alternative to polyester resin, and tested in an axperiment involving 100 kg of thermite melt (M17). Due to the good test results obtained this resin will be used in future for BETA crucible lining.

Ref.3.4 After the partial construction permit had beed granted for the shell constructions, construction work started in schedule within the 7th calendar week of 1982. Because of unforeseeable difficulties with the ground water level for the two exavations (inductor, capacitor batteries) all subsequent deadlines had to be slightly postponed. But since then construction work has been within the fixed deadlines so that the first part of assembly work in process technology including cooling tower, recirculation cooling system, cooling tower circuit and steel construction, was completed on schedule.
Both the final construction permit and the construction permit under water laws for operating the cooling tower had still been granted within the first half-year. Acceptance in time of the hall crane by the Technical Inspectorate (TÜV) made it

possible to use the crane for process engineering assembly work in the hall, as intended. Completion of the BETA building complex (experimental hall, supply building, control room) took place at the end of the period of reporting.

7. Next Steps

Continuation of work on 3.1, 3.2, 3.3 and 3.4.

8. Relation with Other Projects

06.01.13

9. References

D.Perinić, A.Mack: Entwicklung der Versuchsanlage BETA zur Untersuchung der Schmelze/Beton-Wechselwirkung; Jahrestagung Kerntechnik, Mannheim, May 4-6, 1982.

10. Degree of Availability of the Reports

Unrestricted.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01. - 31.12.1982	2.3	06.01.12/22A (PNS4331)
Vorhaben/Project Title: Hydrodynamical and Thermal Models for the Interaction of a Core Melt with Concrete Hydrodynamische und thermische Modelle zur Wechselwirkung einer Kernschmelze mit Beton		Land/Country Fördernde Institution/Sponsor: BMFT Auftragnehmer/Zuwendgsempf./Contractor: Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS), Institut für Reaktorbauelemente (IRB)
Arbeitsbeginn/Initiated: 01.01.1978	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader: Dr. M. Reimann
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

- Quantification of the chronological development of a core meltdown 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 meltdown 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.
- Quantification of the mixing processes and the chemical reactions in the molten pool.
- Assessment of the transferability of the BETA simulation experiments on a core meltdown accident.

3. Research Program

- Development of physical models.
- Code development for the core melt/concrete interaction in the fourth phase of a core melt accident.
- Verification of the WECHSL code by comparison with results of the BETA test facility.
- Definition of the general aims in the BETA test program.

4. Experimental Facilities

--- Computer Programs FILM, BEZENT, WECHSL, SYMROT, DEHDIS

Code Systems: MARCH

5. Program to date

- Computations for the Swedish BWR-plant Barsebäck.
- Extensions of WECHSL for high density concrete.
- Sensitivity study for the influences of different parameters.
- Assessment of the long-term behavior of a coremelt.

6. Results

For the Swedish BWR-plant Barsebäck, computations have been carried out in the framework of the FILTRA project. After failure of the RPV, the core melt slumps into a dry concrete cavity and contacts a steel door. With a simplified version of SYMROT, crust formation, transient heat conduction, and the melt-through of this steel door could be treated. The door's failure occurred after some seconds. After the slumping of the core melt into the suppression pool, the concrete erosion was calculated with WECHSL by assuming water on the top of the melt. At the bounds of the metal melt, massive crusts are formed. Consequently, the melt propagates mainly in the radial direction.
The WECHSL code was extended to cover also the decomposition of high density concrete containing a considerable weight fraction of Fe_2O_3 or Fe_3O_4. These aggregates lead to a rapid oxidation of the metallic constituents of the melt and moreover, they have the potential to reoxidize the burnable gases hydrogen and carbon monoxide to steam and carbon dioxide.
In the 4th phase of a core melt accident, the temperatures of the molten layers drop down in an interaction period of about 10^4 s close to the solidification temperature of the metal layer, and crust formation starts. In the frame of a sensitivity study, the crusts are assumed at one hand to be gas-tight after having grown to a certain thickness and on the other hand, the crusts remain completely gas permeable. Thus, the heat transport from the melt bulk to the inner bound of the crust is varied between natural and gas-driven convection.
The total volume flux of the gases released from the concrete and, consequently, the pressure built-up in the containments, showed only a weak dependence from the heat transfer model. The computed cavity shapes show for siliceous concrete and gas-driven convection a relatively uniform erosion in all directions. In the natural convection case, metal crusts are formed at the bottom and the erosion runs

preferably in the radial direction. For calcareous concrete, the radial erosion is limited, especially in the natural convection case, by oxide crust formation.
Unter the assumption of gas-tight crusts, abated melt bulks occur aft after some hours of interaction. However, this configuration is not stable and tends towards periodical remelting and freezing. About after a week, a stable condition may be reached. The melt layer is then completely frozen, the oxide layer is completely enclosed by a crust and has a high bulk temperature. This results in erosion rates between 0.1 (vertical) and 0.5 (radial) mm/min.

7. Next Steps
- Further development of the physical models.
- Computations of simulation experiments and of core melt accidents.

8. Relations with other Projects
06.01.12/23A (PNS 4334), 06.01.12/20A (PNS 4323)

9. References
Reimann, M., Stiefel, S.: Hydrodynamische und thermische Modelle zur Wechselwirkung einer Kernschmelze mit Beton. PNS-Jahresbericht 1981, KfK 3250, Juni 1982, S. 4300/114-4300/127.
Alsmeyer, H., Müller, U., Reimann, M.: Wärmeübergang von Flüssigkeiten an abschmelzende oder sublimierende Körper. Chem.-Ing.-Techn. 54, (1982), S. 172-173.
Reimann, M., Alsmeyer, H.: Hydrodynamics and Heat Transfer Processes of Dry Ice Slabs Sublimating in Liquid Pools, Proc. of 7th Int. Conf. Heat Transfer, München, Sept. 6-10, 1982, Hemisphere New York, 1982, Vol. 5, S. 167-173.
Hassmann, K., Schwarzott, W., Reimann, M.: Gas Release and Containment History During Melt-Concrete Interaction, 2nd Int. Workshop on the Behavior of Hydrogen During Hypothetical Accidents in Water Reactors, Albuquerque, N.M., Oct. 4-7, 1982.

10. Degree of the Availability of the Reports
unlimited

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.-31.12.1982	2.3	06.01.12/23A (PNS 4334)
Vorhaben/Project Title Modellentwicklungen zur analytischen Beschreibung von Kernschmelzenunfällen Development of models for the analytical determination of core meltdown accidents		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated 1.1.1979	Arbeitsende/Completed 31.12.1982	Leiter des Vorhabens/Project Leader Dr. H. Alsmeyer
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

- Quantification of the chronological development of a core meltdown 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 meltdown 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 on a core meltdown accident.

3. Research Program

- Development of physical models on the coremelt-concrete interaction in the fourth phase of a core meltdown accident.
- Model experiments on the heat transfer and fluid behavior.
- Service for the data acquisition system for the BETA-facility.

4. Experimental Facilities

Laboratory equipment for model experiments.

5. Progress to Date

- Experiments on heat transfer and fluid flow in simulated core melts with existing crusts.

- Model experiments on the investigation of crust formation and stability in two component melts with decay heat simulation.

6. Results

The formation and stability of crusts resulting from the freezing of the metallic or oxidic components of the core melt are very important with respect to the long time behavior of the core materials in contact with the reactor basement. The reduction of heat transfer associated with the formation of crusts as well as the reduced gas release and the possible decrease of hydrogen formation will influence the containment overpressurization and the attac of the concrete.
Model experiments were carried out to investigate the heat transfer from a frozen crust to the lateral concrete wall. Following the present knowledge a gap exists between the decomposing concrete and solidified melt, which is filled by the silicious melt and percolated by the gases released from the concrete. Heat transfer from core material to concrete through the gap is largely enhanced by the agitation of the gas bubbles. The experiments determined the heat transfer as function of gap width, gas flow rate, surface tension, temperature difference and Prandtl-number of the fluid, where the gas was injected at the bottom of the vertical gap. The results could be presented in dimensionless form, thus allowing application to reactor materials. Other experiments have been carried out to study the heat transfer from the liquid melt to the solidified crust of melt material near the concrete surface. On the base of the experiments described below it is assumed that crusts over horizontal surfaces have holes to release gases escaping from the concrete. However, vertical walls are covered by a gas tight crust. The model experiments, first carried out with water to simulate a low viscous, low Pr-number melt, show that the mean heat transfer from the melt to the crust surface is in a large range independent of the number of gasing holes and depends only on the volume flux of the gas. For horizontal crusts, the heat transfer coefficient is proportional to $Re^{0.4}$, where Re is formed with the superficial gas velocity, and is independent of the driving temperature difference. Heat transfer to a vertical crust which is, for the conditions mentioned above, in contact to a gas agitated fluid, practically agrees with the heat transfer to the horizontal

crust. This signifies that the heat transfer is dominated by the stirring of the gas bubbles in the pool whereas the release of the bubbles from the surface is of minor importance to the mean transfer. Present experiments with highly viscous fluids investigate the influence of Pr-number on heat transfer to allow application of the experimental results to the different components of a core melt. Further experiments studied the phenomenology of the formation and stability of crusts in two component melts using horizontal layers of simulant materials. The layers of concrete -metallic melt- oxidic melt were simulated by dry ice - water - oil with internal heat simulation in the oil layer. Heating rate was adjusted in such a way that solidification of the melt could occur due to heat removal to the sublimating dry ice. For high heat generation in the oil the percolation of the released gas through the melt is unaffected by the thin ice crust at the bottom of the water layer, because the crust possesses a large number of gas releasing holes. Consequently, the melt is well stirred and the temperatures in the water and oil layer are nearly equal. The experiment shows that this is a stable situation over long periods. Reducing the internal heat generation, crust thickness may increase considerably and supress the gas release through the crust. Therefore, heat transfer is controlled by free convection due to temperature induced density gradients in the liquid pool materials. To maintain the heat removal under the different condition, a much higher temperature in the oil layer does occur, thus allowing for heat removal to the bottom by conduction and thermal convection only. In an intermediate heating range, the experiments show periodic melting and freezing as well as locally inhomogeneous crust formation. To apply these phenomena to the reactor situation, quantitative models have to be developed.

7. <u>Next steps</u>

Further investigations on heat transfer and model description for formation and stability of crusts including more complex geometries.

8. <u>Relations with Other Projects</u>

06.01.12/12A (PNS 4331); 06.01.13/04 (PNS 4323)

9. <u>References</u>

7th Int. Heat Transfer Conf., Munich (1982), Vol.4,p.167; Chem.-Ing. Tech. 54(1982) p. 172

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

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.1982 - 31.12.1982	2.3	06.01.16/08A/09A (PNS 4251)
Vorhaben/Project Title Out-of-pile Bündelexperimente für die Untersuchung schwerer Brennelementschäden. Out-of-pile Bundle Experiments Investigating Severe Fuel Damage.		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe GmbH (KfK) Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated 1.1.1981	Arbeitsende/Completed 31.12.1982	Leiter des Vorhabens/Project Leader Dr. S. Hagen, K. Hain
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General aim

Small break LOCAs and special transients occuring with the additional failure of the emergency core cooling system may lead to severe fuel damage without necessarly escalating to an uncontrolled core meltdown. PNS has sponsored several experimental investigations on the most important phenomena influencing fuel specific damage, the coolability of a damaged core, and the release of fission products from damaged fuel.

2. Specific objectives

Damage mechanism in the temperature range to 2000 °C will be investigated in out-of-pile bundle experiments. Boundary conditions (geometrical arrangement, heat production, quenching, pressure relations) will be as realistic as possible. Many small break scenarios can be postulated with no unique failure sequence. Therefore, the tests will be performed with transients between 0.1 and 4 °C/sec to bound the extent of damage.

3. Research program

Within the parameters given in section 2 the following will be investigated:
- Influence of the exothermal zirconium-steam reaction on the temperature time behavior (temperature escalation).
- Dissolution of UO_2 by zircaloy.
- Fragmentation of fuel rods by quenching.
- Oxidation behavior of molten material.
- Influence of absorber materials and absorber guide tubes.
- Comparison with inpile experiments (PBF,Super Sara,ACRR, Siloe (TMI-Flash)).

4. Experimental facilities and computer codes

The NIELS facility will be used for the initial experiments without quenching and pressure and for testing the instrumentation. The final facility (CORA) is in construction. Modeling of fuel behavior will be done with SSYST. Oxidation behavior will be similated by the SIMTRAN and MOP programs.

5. Progress to date

CORA facility development:

The technical details of the CORA facility were further developed and a schematic of the whole facility produced. In order to use the rotunda of FR2 with the existing infrastructure as a location for the CORA facility, it was proven for safety reasons that the FR2 facility could expect no reaction from the experiment operation and proposed transients. The fuel rod simulator concept was tested at high temperature. The first parts of the facility have been ordered.

Pretests:

The influence of the exothermal reaction energy on the temperature rise was investigated with two types of experiments: (1) A series of 6 experiments with a single fuel rod simulator surrounded by a zircaloy shroud and 10 cm of ceramic fiber insulation (ZrO_2+Al_2O_3) was performed in steam. The initial temperature rise was varied between 0.3 and 5 ^oC/sec by varying the electric power increase. (2) A 3x3 bundle of fuel rod simulators was heated in steam. The bundle was surrounded by a zircaloy shroud and a 6 mm thick zircaloy fiber insulation.

6. Results

CORA facility development:

The requirement of reaction free experiment operation within FR2 lead to the concept of a "safety enclosure". That is, the facility components which are relevant from a safety standpoint were enclosed in a 210 m^3 pressure tight containment. The high temperature shield, quench funnel, surge condenser, and the two vent condensers were specifically enclosed.

The effectiveness of the chosen "safety enclosure" with respect to the following topics was substantiated by the expert opion of the Frauenhofer Institute for Fuel and Explosive Materials:

1. Postulated steam explosions, caused by downward falling melt.
2. Effect of a hypothetical hydrogen explosion inside the experiment apparatus. (The hydrogen originates from the zircaloy/steam reaction of the cladding.)

Pretests:

All 6 single rod experiments showed a distinct temperature escalation. The maximum temperature gradient achieved was about 6 ^oC/sec. In all cases the maximum temperature remained below 2100 °C. The reaction was limited by run off of the molten zircaloy out of the reaction zone. Also, the formation of a hydrogen layer on the surface of the oxide may have influenced the limitation of the escalation.

In the bundle experiment the greatest temperature escalation was observed on the central rod of the bundle. The maximum temperature achieved was around 2200 °C. The melt originating in the reaction zone later froze in the lower, cooler region of the bundle in a compact mass solidly surrounding the rods.

7. Work to be done

Work on the temperature escalation will be continued in single rod and bundle experiments.
Comparison of results in steam and Ar/O_2 mixtures with improved axial temperature measurements will provide information on the possible influence of a hydrogen layer.
Design, construction and ordering of components for the CORA facility is continuing.

8. Relation with other projects

The Severe Fuel Damage investigations are performed as part of PNS in various institutes (IT, IMF I, IMF II, IRB, PNS/PL). The out-of-pile experiments are done in close cooperation with the inpile experiments of USNRC (PBF and ACRR programs), EURATOM-Ispra (Super Sara), and CEA (Phebus program and TMI-Flash at the SILOE).

9. References

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10. Degree of availability of the reports

-

	CLASSIFICATION : 2.3
TITLE (ORIGINAL LANGUAGE) : NUMERICAL SOLUTION OF CERTAIN CORIUM FLOW PROBLEMS	COUNTRY : SWEDEN SPONSOR : S K I
TITLE (ENGLISH LANGUAGE) :	ORGANISATION : CHALMERS TEKNISKA HÖGSKOLA PROJECT LEADER : BRYAN McHUGH

INITIATED : NOVEMBER 1982	COMPLETED : Phase 1 NOVEMBER 1984	SCIENTISTS : GEORGE FILEAS
STATUS : IN PROGRESS	LAST UPDATING : JUNE 1983	

1. General Aim — To develop numerical methods suitable for handling complex flow situations where multi-phase freezing phenomena, internal heat generation and non-Newtonian behavior can be present.

2. Particular Objectives. To calculate the blockage risk when a core melt flows through penetrations - either in the reactor vessel or in pipes into the wet well of a BWR plant.

3. Experimental Facilities. None envisaged at this stage of the work.

4. Project Status.

   1. Progress to Date: A finite volume method has been coded for a two dimensional calculation of corium flow down a short vertical pipe.
   2. Essential Results: The method has been shown to be numerically stable with acceptable running times on a ND-50 minicomputer.

      Instabilities in the freezing surface have been observed - which agrees with other work - but which have not yet been given a satisfactory physical explanation.

5. Next Steps — To extend the programme method (FINITE VOLUME) to three dimensions on a VAX 750 which will become available in the Fall of 1983.

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6. Relations to Other Projects and Codes.

It is hoped to be able to follow the German experimental work at Karlsruhe in order to be able to attempt the numerical simulation of some of the empirical situations - as a very important way of correcting the numerical method and validating the results.

7. Reference Documents.

FFOCUS - A computer Program for Calculating Freezing of Corium Flowing through a Control Rod Penetration	G Fileas June 1981 R-81/06
A simple explanation of non-Newtonian concepts and their application to FFOCUS	G Fileas December 1982 R-82/08 ISSN 0280-7572
On Numerical Methods in Non-Newtonian Flows	G Fileas December 1982 R-82/07 ISSN 0280-7572
Computation of Freezing Flow Phenomena - Progress Report: Nov 1982 - April 1983	G Fileas May 1983 R-83/12

8. Degree of Availability

No restriction

3. EXTERNAL INFLUENCES

3.1. SEISMIC EFFECTS

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.82 - 31.3.82	3.1	150444
Vorhaben/Project Title Untersuchungen zum nichtlinearen Verhalten von erdbebenbeanspruchten Stahlbetonkonstruktionen Investigations on the Non-Linear Behaviour of Reinforced Concrete Structures under Seismic Loads		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor TH Darmstadt Institut für Massivbau
Arbeitsbeginn/Initiated 1.11.79	Arbeitsende/Completed 31.3.82	Leiter des Vorhabens/Project Leader Prof. Dr.-Ing. König
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

1. General Aims

Calculation methods to consider quite exactly the nonlinear behaviour of reinforced-concrete structures under seismic loading are so extensive and sensitive, that it seems not practicable to use them on problems of engineering practice. Therefore sufficiently checked approximation methods are of great importance.

Approximation methods used at present, which try to consider the energy-dissipation caused by plastic deformations by means of either introduced ductility retios or modified damping ratios shall be checked by an experimently proved calculation and shall be improved, if necessary.

2. Particular Objectives

2.1 Experimental checking and in case improvement of a calculation method to consider the nonlinear behavior of reinforced concrete structures under seismic loading.

2.2 Analytical checking of approximation methods used at present through the above mentioned calculation method. In case development of improved approximation methods.

1.1.82 - 31.4.82 150444

3. Research Programm

3.1 Development and assembly of an experimental facility

3.2 Preliminary tests to check the experimantal facility

3.3 Development of test-models: cantilever beams and one bay/one story frames

3.4 First testing series

3.5 Analysis of the first testing series; in case improvement of the calculation method

3.6 Second testing series with extended range of parameters

3.7 Analysis of the second testing series

3.8 Checking of approximation methods used at present, in case improvements

4. Experimental Facility, Computer Codes

Ref. 3.1 Computer-aided experimental facility consisting of the components

a) Computer
b) Actuator
c) Shaking table
d) Measuring tools

Function:
Conversion of acceleration time-histories into control impulses for the actuator and registration and analysis of the results of measurement (accelerations, displacements, strains) by the computer.

Ref. 3.3 R/C cantilever beams and simple R/C-frames are provided as test-models

Ref. 3.4 The experiments on the earthquake simulator are controlled by the computer-program ITFC (Iterative Transfer Function Compensation)

Ref. 3.5 For recomputation of the experiments the computer program SAKE is used. For a detailed interpretation of the experimental data a group of routines are developed.

Ref. 3.8 For calculations on basis of approximation methods SAP IV will be used

5. Progress to Date

Ref. 3.5 Interpretation of experiments of the first series. Modification of the calculation method (step by step method) by improving the material models

Ref. 3.6 Development of test models for investigations with an extended range of parameters

6. . Results

Ref. 3.5 Derivation of parameters to describe the dynamic properties of R/C members (Stiffness degradation, energy absorption due to material and friction damping and plastic deformations); improvement of the member model
Modification of the calculation method

7. Next Steps (within the Project 150444 A9)

Ref. 3.6 Further improvement of the calculation method to
3.7
consider the nonlinear plastic properties of R/C members and structures.

Ref. 3.8 Analytical and experimental checking of approximation methods used at present

8. Relation with Other Projects
   Project 150444 A9

9. References
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10. Degree of Availability of the Reports
    GRS - Forschungsbetreuung

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
15.10.82 - 31.12.82	3.1	150444 A
Vorhaben/Project Title Untersuchungen zum nichtlinearen Verhalten von erdbebenbeanspruchten Stahlbetonkonstruktionen Investigations on the Non-Linear Behaviour of Reinforced Concrete Structures under Seismic Loads		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: TH Darmstadt Institut für Massivbau
Arbeitsbeginn/Initiated 15.10.82	Arbeitsende/Completed 31.12.82	Leiter des Vorhabens/Project Leader Prof. Dr.-Ing. König
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

1. General Aims

Calculation methods to consider the nonlinear dynamic response of reinforced concrete structures using the technique of direct integration and solving the equations of motion in incremental steps are best suited, however, they are also very extensive and sensitive. The nonlinear response characterics of the member models should be experimentally checked and the phenomenon of energy absorption in dynamically loaded R/C members should be clarified.

An experimentally proved calculation also provides for means of evaluation and discussion of approximation methods used at present.

2. Particular Objectives

2.1 Improvement of the nonlinear models for R/C members covering the range of the governing parameters.

2.2 Discussion of the approximation methods.

3. Research Program

3.1 Testing series with extended range of parameters.

3.2 Further improvement of the nonlinear models for R/C member behaviour.

3.3 Checking of approximation methods used at present.

4. Experimental Facility, Computer Codes

Ref.3.1. R/C cantilever beams and simple frame structures are going to be tested on computer-aided experimental facility consisting of the components

a) Computer
b) Actuator
c) Shaking table
d) Measuring tools

Conversion of acceleration time-histories into control impulses for the actuator and registration and analysis of the results of measurement (accelerations, displacements, strains) by the computer.

Ref.3.2. For detailed interpretation of the experimental data a group of routines are developed.

5. Progress to Date

Ref.3.1 Completion of the tests

Ref.3.2 Development of an analytical model for the nonlinear inelastic behaviour of R/C members

Ref.3.3 Checking of approximation methods

6. Results

Ref.3.1 and Ref 3.2 Influence of bond between steel and concrete to deformation during cyclic loading.
Stiffness degradation and energy absorption in statically undetermined structures with capability of force distribution within the system.

15.10.82 - 31.12.82 150444 A

Ref.3.3 Approximation methods checked according to their suitness for design purposes

7. Next Steps

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8. Relation with Other Projects

Project 150 444

9. References

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10. Degree of Availability of the Reports

GRS - Forschungsbetreuung

Berichtszeitraum/Period 01.07.82-31.12.82	Klassifikation/Classification 3.1	Kennzeichen/Project Number 150459
Vorhaben/Project Title Schwingungsmessungen an Bauwerken Vibration Measurements at Buildings		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Universität Stuttgart Keplerstraße 7 7000 Stuttgart 1 Institut für Mechanik (Bauwesen)
Arbeitsbeginn/Initiated 01.06.80	Arbeitsende/Completed 31.12.82	Leiter des Vorhabens/Project Leader Prof. E. Luz
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

1. General Aim

Determination of natural frequencies, mode shapes and damping behaviour of optional buildings and structural parts by means of seismometer measurements at very low excitation level ("environment noise").

2. Particular Objectives

Optimization of experimental arrangement and working conditions to record and to analyze building vibrations.
Comparison between results of measurements and results obtained by calculations with different mechanical models, as well as with other measurement methods.

3. Research Program

3.1 Completion of the recording technic for different types of buildings and excitations.

3.2 Development of analyzing technic for spectral and covariant analysis.

3.3 Comparison between results of different measurement-methods and calculations of several different mechanical models.

4. Experimental facilities, Computer Codes

Experimental facilities:
12 seismometers system Willmore Mk III/A, 12 amplifiers system Tectronix AM 502, 14-channel tape recorder Sangamo Sabre VII, modal analyzer HP 5423A and a graphic plotter HP 9872B.
Computer codes to calculate natural frequencies and modes

of a bending-torsion-bar according to [1], [2], [3]:

HHS

HHS EX

HHS EXA

GHS

A plot program is available for all codes. The first three codes are published in [4], the last one will be described in [5].

5. Progress to Date

Ref. 3.2 Analyzation of the measurements were completed which had been taken at the HDR in Kahl in spring 1982 by using "environment noise" for excitation. Natural frequencies were determined in the range from 0 to 30 Hz, modes in the range from 0 to 16 Hz.

Ref. 3.3 To compare the results in 6. with the results of other methods different publications of the HDR-project can be used [6], [7].

6. Results

6.1 Results of seismometer-measurements at the HDR in Kahl.

The results are shown in the following table, for further information see [5]. Underlined figures means natural frequencies which are strongly marked, figures in parantheses indicate less marked frequencies or that ones which are not present in all spectra.

The damping values are calculated from the bandwidth at the half-power point from single spectra.

Table

Remarks	Natural Frequencies Hz	Damping %	Natural Frequencies in Hz PHDR 4-78 Tab. 10	PHDR 13-80 Tab. P. 215
X-direction	1,45 - 1,50	4,35	1,52	1,35 - 1,48
Z-direction	1,50 - 1,55	4,35	1,57	1,40 - 1,54
X-direction	2,55	2,82	2,63	2,44 - 2,53
Z-direction	2,55 - 2,65	2,82	2,81	2,56 - 2,66

Table (Cont.)

Remarks	Natural Frequencies Hz	Damping %	Natural Frequencies in Hz PHDR 4-78 Tab. 10	PHDR 13-80 Tab. P. 215
	3,30 - 3,35	2,15	3,35	
	(4,15)			
Torsion	4,55	0,95		
	(4,85)			
	5,05 - 5,25	0,99		4,98 - 5,00
Torsion?	5,75 - 5,95			5,70 - 5,90
	(6,35 - 6,70)			6,32 - 6,54
	7,20 - 7,25			7,32
	(8,20)		8,16	7,8
	(8,75)			8,50
	9,75	0,53		
	11,20 - 11,35	0,45		11,36 - 11,58
	12,45		12,25	
			12,80	
	13,30			13,00 - 13,22
	14,00	0,36	13,88	14,20
	(14,45)		14,60	
			15,25	15,07 - 15,26
	15,75	0,46		
	16,55			
	17,40			
	(19,65)			
Torsion?	20,00	0,29	20,00	
	(20,25)			
	(20,95)			
	23,45			
Torsion?	23,75	0,20		
	24,70 - 24,75	0,33	24,50	
Torsion?	26,05 - 26,15	0,17		
	(28,90)		29,00	
	29,75	0,34		

6.2 Summary and outlook

Measurements by seismometers when the structure is excited by environment noise only can be told as successful; nevertheless there are required further rationalizations in the analyzing technique to apply the method in testing practice.

7. Next Steps

Project is completed.

8. Relation with other Projects

150 444, RS 123 B, HDR EV 4000.

9. References

[1] Luz, E., Vorlesungen über Baudynamik, Teil 1 (Lectures on Structural Dynamics, Part 1), Institut für Mechanik (Bauwesen), Universität Stuttgart 1982.

[2] Luz, E. and Gurr, S., Berechnung von Hochhausschwingungen im Hinblick auf die Beanspruchung durch Erdbeben (Computation of Vibrations of Buildings in order to study the Earthquake-Resistance). Ing.-Arch. 51(1981), P. 75-88.

[3] Luz, E. and Gurr, S., Measurement and Calculations of Natural Frequencies and Coupled Bending-Torsion Modes of Highrise Buildings. Proc. 7th World Conference on Earthquake Engineering, Istanbul 1980, Vol. 7, P. 437-440.

[4] Luz, E., Gurr-Beyer, C., Stöcklin, W., Schwingungsmessungen an Bauwerken, Zusammenfassender Zwischenbericht zum 31.03.81 (Vibration Measurements at Buildings, Interim Report at 31.03.81), Institut für Mechanik (Bauwesen), Universität Stuttgart 1981.

[5] Luz, E., Gurr-Beyer, C., Stöcklin, W., Schwingungsmessungen an Bauwerken, Abschlußbericht (Vibration Measurements at Buildings, Final Report), Institut für Mechanik (Bauwesen), Universität Stuttgart, to be published in 1983.

[6] Steinhilber, H., Jehlicka, P., Malcher, L., HDR Sicherheitsprogramm, Technischer Fachbericht PHDR 4-78 (HDR Safety Program, Technical Report PHDR 4-78), Kernforschungszentrum Karlsruhe 1978.

[7] Jehlicka, P., Malcher, L., Steinhilber, H., Brendel, B., HDR Sicherheitsprogramm, Technischer Fachbericht PHDR 13-80 (HDR Safety Program, Technical Report PHDR 13-80), Kernforschungszentrum Karlsruhe 1980.

10. Degree of Availability

Ref. [1] is available at Institut für Mechanik (Bauwesen), Pfaffenwaldring 7, 7000 Stuttgart 80, FRG; Ref. [4] and [5] at GRS-Forschungsbetreuung, Glockengasse 2, 5000 Köln 1, FRG.

121-1-03		3 - 1
TITRE METHODOLOGIE POUR LE CALCUL DES SPECTRES DES SEISMES DE REFERENCE A PARTIR DE PARAMETRES PHYSIQUES		*Pays* FRANCE *Organisme directeur* CEA/DSN
TITRE en anglais METHODOLOGY FOR THE CALCULATION OF REFERENCE EARTHQUAKE SPECTRA BASED UPON PHYSICAL PARAMETERS		*Organisme éxécuteur* CEA/DSN/SAER FONTENAY AUX ROSES *Responsables* B.MOHAMMADIOUN DSN/SAER
Date de démarrage 01.01.1976 *Date d'achèvement* 31.12.1984	*Etat actuel* en cours *Dernière mise à jour* Mars 1983	*Scientifiques* X. GOULA DSN/SAER G. MOHAMMADIOUN M.BOUCHON UNI.GRENOBLE

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, de l'intensité du séisme de référence, des caractéristiques du sol, de la source, à l'aide d'une analyse des données obtenues dans le monde et sélectionnées selon les différents critères suivants :

  a) régional (CALIFORNIE, EUROPE, etc...), 1976 - 1984

  b) géologie similaire du site (roche dure, alluvions) 1982 - 1984.

- Etude probabiliste du risque sismique. Cette étude a pour but de définir une méthode de détermination de la probabilité annuelle de dépassement d'un spectre du mouvement du sol, permettant, en particulier, l'évaluation pour un site donné, de la probabilité de dépassement du spectre de dimensionnement 1980 - 1984.

3) INSTALLATIONS EXPERIMENTALES

Voir fiches :

121-1-04 : Surveillance de la sismicité des sites nucléaires.

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121-1-05 : Collecte de mesures sur les mouvements en zone épicentrale et d'information sur les dégâts correspondants.

4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

- 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 corrélation ont été obtenus pour 46 fréquences et 5 amortissements.

- Fondée sur ces études, une méthodologie de calcul des spectres adaptés aux sites a fait l'objet d'une Règle Fondamentale de Sûreté (I.2.c).
- Le séisme de Salon de Provence a fait l'objet d'une étude utilisant un modèle mathématique.

- Une étude de faisabilité d'une approche probabiliste du risque sismique, exprimé en intensité, a été réalisée pour le Sud-Est de la FRANCE, et pour le site de CREYS-MALVILLE.

- Une étude concernant la variation du mouvement sismique en champ proche a été effectuée à partir de données récentes.

- Principaux résultats

- 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, 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.

  La variation des spectres en fonction de l'intensité macrosismique a fait l'objet d'une étude détaillée. On constate que le mouvement est multiplié par des coefficients variables en fonction de la fréquence quand on augmente l'intensité d'un degré. Le spectre SMS est obtenu à partir du spectre de SMHV en multipliant ce dernier par des coefficients de majoration obtenus par cette étude.

  A l'heure actuelle, cette méthode permet de traiter les cas de sites dont l'intensité du SMHV est inférieure à VIII et à 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.

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Pour le séisme de distance focale inférieure à 10 km, de magnitude faible, on a adopté une forme de spectre forfaitaire que l'on cale en vitesse. Les enregistrements des séismes récents de grande magnitude obtenus sur des sites alluvionnaires, sur la composante horizontale, montrent que l'accélération maximale tend vers une limite de l'ordre de 0,5 g. L'accélération enregistrée sur la composante verticale, due aux ondes de compression, ne parait pas être soumise à une telle limite.

- L'approche probabiliste utilisée permet de calculer les probabilités de dépassement de différentes intensités, pour un site donné, à partir de la connaissance de la sismicité historique. Compte tenu des jugements déterministes, ainsi que des incertitudes des paramètres intervenant dans le calcul, les résultats sont obtenus avec une incertitude notable (pour un niveau de probabilité de 10^{-4}/an elle est de l'ordre d'un degré).

5) PROCHAINES ETAPES

Amélioration des coefficients de corrélation par l'introduction de nouvelles données (pour les intensités V, VI et VII) et surtout à courtes distances ($R \leq 10$ km).

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.

Utilisation de modèles théoriques pour le calcul du mouvement en champ proche, développement de l'étude du séisme de 1909.

Le développement d'une approche probabiliste du calcul du mouvement de référence sera poursuivi en intégrant les résultats des corrélations : spectre-distance focale-intensité.

6) RELATIONS AVEC D'AUTRES ETUDES

"Collecte de mesures sur les mouvements en zone épicentrale et d'informations sur les dégâts correspondants" - Fiche étude 121-1-05. Cette fiche apporte des informations essentielles à l'étude des corrélations entre les différents paramètres et au calcul des spectres de référence.

7) DOCUMENTS DE REFERENCE

- Détermination des spectres de référence adaptés aux sites
Note Technique SESRS n° 40 - Octobre 1978
B. MOHAMMADIOUN

- Détermination du mouvement de référence pour le site de CRUAS
Note Technique SESRS n° 50 - Novembre 1978
B. MOHAMMADIOUN

- Détermination des spectres de SMS adaptés aux sites
Rapport DSN n° 324 - Novembre 1979
C. DEVILLERS, B. MOHAMMADIOUN

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- Règle fondamentale de Sûreté I.2.c
  "Détermination des mouvements sismiques à prendre en compte pour la sûreté des installations".

- Variation des paramètres du mouvement sismique en champ proche : analyse de quelques exemples récents
  Note Technique SAER - 83 - 343
  B. MOHAMMADIOUN (mars 1983)

- Etude de faisabilité d'une approche probabiliste de l'évaluation du risque sismique. Application au Sud-Est de la FRANCE.
  Rapport DSN n° 387 - Octobre 1980
  X. GOULA

- Evaluation probabiliste du risque sismique sur le site de CREYS-MALVILLE
  Rapport SESRS n° 22 (confidentiel) - Septembre 1980
  X. GOULA, Agnès LEVRET

- Etude du séisme de 1909
  Etat d'avancement - Février 1983
  X. GOULA

8) DEGRE DISPONIBILITE

Documents internes, non disponibles sauf accord du CEA, à l'exception de la RFS I.2.c.

121-1-04				3 - 1
TITRE SURVEILLANCE DE LA SISMICITE DES SITES NUCLEAIRES				*Pays* FRANCE *Organisme directeur* CEA/DSN
TITRE en anglais INSTRUMENTAL MONITORING OF SEISMICITY SURROUNDING NUCLEAR SITES				*Organisme éxécuteur* CEA/DSN/SAER FONTENAY AUX ROSES *Responsables* H. FERRIEUX DSN/SAER
Date de démarrage	01.01.1976	*Etat actuel*	En cours	*Scientifiques* A. BRESSON G. MOHAMMADIOUN DSN/SAER
Date d'achèvement	31.12.1984	*Dernière mise à jour*	Mars 1983	

1) OBJECTIF GENERAL

L'objectif de cette étude est la recherche d'une meilleure connaissance de la sismicité aux alentours d'un site nucléaire en vue de l'évaluation du risque sismique.

2) OBJECTIFS PARTICULIERS

- Mise en évidence des zones sources au voisinage d'un site nucléaire (failles actives) 1976 - 1984.
- Détermination des caractéristiques spécifiques du site en ce qui concerne la transmission des ondes sismiques en fond de forage et en surface (TRICASTIN).
- Prévision des mouvements de référence à prendre en compte pour le calcul ou la vérification du dimensionnement 1975 - 1984.
- Enregistrement des mouvements sollicitant les installations du site lors d'un séisme fort.

3) INSTALLATIONS EXPERIMENTALES

- Observatoire de Cadarache : équipé de différents types d'appareils de mesure (accéléromètres, capteurs de vitesse disposant d'une dynamique étendue en amplitude et en fréquence).
- Réseau de surveillance du Tricastin, comprenant 3 stations de mesure, dont une comporte des capteurs au fond d'un forage de 80 mètres (MAZET), une autre un appareillage de surface (MAZET) et la troisième des accéléromètres (BOURG-SAINT-ANDEOL). Les stations de MAZET ont été équipées avec la nouvelle chaine de mesures mise au point par la laboratoire de mesures sismiques du BERSSIN.

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- Des accéléromètres type SMA1 sont installés au voisinage des installations nucléaires de la Hague et Marcoule.

4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

- Avancement à ce jour

. Surveillance de l'activité de la région de CADARACHE.

. Surveillance de l'activité sismique de la région de TRICASTIN.

- Résultats essentiels :

Station de CADARACHE :

Mise en évidence des activités des failles de JOUQUES et de BEAUMONT de PERTUIS près de CADARACHE et étude des séismes de la région de GARDANNE.

Enregistrement de petits séismes dans la région de MANOSQUE et MARSEILLE.
Enregistrement de séismes alpins et des séismes méditerranéens importants : FRIOUL, JURA SOUABE, KOTOR, EL ASNAM, MER IONIENNE.
La station de CADARACHE est un "Observatoire international" et ses données sont communiquées aux organismes intéressés, entre autres :

. Centre Sismologique Européo-Méditerranéen (CSEM) FRANCE.

. National Earthquake Information Service (U.S.A.).

. International Seismological Centre (ISC) NEWBURY (G.B.).

Station du TRICASTIN :

Enregistrements de séismes européens importants.

Détermination de fonctions de transfert locales et de lois d'atténuation.

Spectres de référence adaptés au site de TRICASTIN à l'aide des enregistrements effectués.

5) PROCHAINES ETAPES

- Poursuite de la surveillance des sites précités.

- Installation d'une station de surveillance sismique sur le site de GRENOBLE.

- Etude de faisabilité de l'implantation de stations supplémentaires dans la Basse Vallée du RHONE.

6) RELATIONS AVEC D'AUTRES ETUDES

Méthodologie pour le calcul des spectres des séismes de référence à partir des paramètres physiques - Fiche étude 121-1-03.

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7) <u>DOCUMENTS DE REFERENCE</u>

- Bulletin mensuel de l'Observatoire de CADARACHE
  G. MOHAMMADIOUN

- "Les études sismologiques effectuées au C.E.A. dans le domaine de la Sûreté des Sites Nucléaires"
  Rapport IAEA - VIENNE 1975
  A. BARBREAU, H. FERRIEUX, B. MOHAMMADIOUN.

- "Interprétation des campagnes de mesures sismiques effectuées au TRICASTIN"
  Rapport SESRS n° 1 - Juin 1978
  H. FERRIEUX, G. MOHAMMADIOUN.

- "Informations préliminaires concernant le séisme du 19 février 1975, ressenti dans la région de PIERRELATTE"
  Fiche DSN/SESRS 75/105 du 28 février 1975
  B. MOHAMMADIOUN.

- Station mobile à seuil pour l'enregistrement des mouvements sismiques
  Rapport DSN n° 488 - Décembre 1981
  E. FAVIER.

- Situation de la surveillance séismique du Sud-Est de la FRANCE...
  Note Technique SAER-82-329 - Décembre 1982
  G. MOHAMMADIOUN.

8) <u>DEGRE DE DISPONIBILITE</u>

Documents internes, non disponibles sauf accord du CEA, à l'exception du Bulletin de l'Observatoire de Cadarache et du rapport AIEA.

121-1-05			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
TITRE en anglais COMPILATION OF RECORDINGS OF NEAR FIELD MOTION AND OF INFORMATION ON THE CORRESPONDING DAMAGES			*Organisme éxécuteur* CEA/DSN/SAER FONTENAY AUX ROSES *Responsables* B. MOHAMMADIOUN DSN/SAER
Date de démarrage 01.01.1976	*Etat actuel* En cours		*Scientifiques* H.FERRIEUX -DSN/SAER X.GOULA -DSN/SAER A. LEVRET -DSN/SAER
Date d'achèvement 31.12.1984	*Dernière mise à jour* Mars 1983		

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, 1978 - 1984.

Rassemblement des caractéristiques correspondant à ces enregistrements (intensité au lieu d'enregistrement, conditions géologiques du site, instrument enregistreur, etc...), 1978 - 1984.

Rassemblement des répliques en zone épicentrale d'un séisme important à l'aide des stations d'intervention réalisées au DSN, 1973 - 1984.

Enregistrements de séismes dans les régions françaises où l'activité sismique est importante (1978 - 1984).

Classement et analyse de ces données en fonction des différents critères :

. régional,

. intensité au lieu d'enregistrement,

. condition du site (roche dure, alluvions) 1978 - 1984.

3) INSTALLATIONS EXPERIMENTALES

Stations sismologiques légères destinées à l'enregistrement des séismes (répliques) dans la zone épicentrale après un séisme important.

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Stations sismologiques temporaires dans des zônes à forte sismicité, (UBAYE, PYRENEES).

Réalisation des expérimentations sur un site à partir de sources sismiques artificielles afin d'étudier la loi de transmission des ondes sismiques de la région immédiate du site (TRICASTIN, NICE, etc...).

4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

- Avancement à ce jour :

Enregistrements de nombreuses répliques dans la région du FRIOUL avec la détermination de leur intensité : création d'un fichier informatique comportant accélération, vitesse, déplacement, spectres de réponse.

Enregistrements des séismes dans la vallée de l'UBAYE.

Enregistrements en plusieurs points des nombreuses répliques dans la région de TUBINGEN après le séisme du 3 Septembre 1978.

Enregistrements en six points de plus de 400 répliques à EL ASNAM après le séisme du 10.10.1980.

Acquisition des enregistrements du séisme du 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) en collaboration avec le Professeur AMBRASEYS de l'Imperial College de LONDRES.

Acquisition, dans le cadre de l'accord CEA/NRC, d'une bande réalisée à la demande de NRC, qui rassemble les caractéristiques des enregistrements du mouvement fort obtenus dans le monde.

Obtention de documents détaillés traitant des caractéristiques des sites d'enregistrement des mouvements forts réalisés aux ETATS-UNIS.

Mise au point à la CISI d'un programme d'accès aux fichiers informatiques des différentes données (sismothèque).

- Principaux résultats :

La sismothèque comporte actuellement de nombreux mouvements forts obtenus aux U.S.A.,Certains enregistrements Européens (Italie, Yougoslavie, Grèce, etc...) et les enregistrements réalisés par le DSN/BERSSIN au FRIOUL, JURA SOUABE, EL ASNAM.
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.

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Dans le cadre du contrat DSN-Imperial College de LONDRES, une première étude fait le point sur les différentes méthodes de traitement du signal.

5) PROCHAINES ETAPES

Etude des caractéristiques des séismes enregistrés en EUROPE (magnitude, intensité, distance focale, fonction-source).

Apport à la sismothèque de données nouvelles (complément d'enregistrements américains, données japonaises, européennes).

Poursuite des enregistrements des séismes en FRANCE et dans les pays limitrophes.

6) RELATIONS AVEC D'AUTRES ETUDES

Méthodologie pour le calcul des spectres des séismes de référence à partir de paramètres physiques - Fiche étude 121-1-03.
La présente étude fournit les données nécessaires à la bonne exécution de la fiche ci-dessus.

7) 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.

- Interprétation des campagnes de mesures sismiques effectuées au TRICASTIN
  Rapport SESRS n° 1 - Juin 1978
  H. FERRIEUX, G. MOHAMMADIOUN.

- Données du réseau "Strong Motion" américain actuellement disponibles
  Rapport SETSSR n° 78 - Juin 1978
  G. MOHAMMADIOUN.

- Analyse des données sur les mouvements forts actuellement disponibles au DSN/SESRS
  Rapport SESRS n° 15 - Mars 1980
  B. MOHAMMADIOUN, G. MOHAMMADIOUN.

- Etude des répliques du séisme du 10 Octobre 1980 à EL ASNAM
  Rapport DSN n° 453 - Juin 1981
  E. FAVIER et al.

- Les études géophysiques expérimentales dans l'évaluation du risque sismique
  H. FERRIEUX - Cours INSTN - Septembre 1982.

- Various Methods of Preprocessing the Strong Motion Records
  J.M.H. MENU - Septembre 1981.

- Station mobile à seuil pour l'enregistrement des mouvements sismiques
  Rapport DSN n° 488 - Décembre 1981
  E. FAVIER.

8) <u>DEGRE DE DISPONIBILITE</u>

Documents internes, non disponibles sauf accord du CEA, à l'exception de l'étude des répliques du séisme au Frioul.

142-1-01	3 - 1
TITRE ANALYSE PARASISMIQUE D'UNE CENTRALE NUCLEAIRE PWR - METHODE DE CALCUL .	*Pays* FRANCE
	Organisme directeur CEA/IPSN
TITRE en anglais SEISMIC ANALYSIS OF A P.W.R. POWER PLANT - CALCULATION METHOD .	*Organisme exécuteur* CEA/DEMT.SACLAY
	Responsables R.J.GIBERT/DEMT-SMTS
Date de démarrage 1.1.75 — *Etat actuel* en cours *Date d'achèvement* 1985 — *Dernière mise à jour* Mai 83	*Scientifiques* D.COSTES DAS/SAER F.GANTENBEIN - DEMT SMTS

1.) OBJECTIF GENERAL -

Le but est la mise au point d'un ensemble de méthodes permettant de calculer la réponse d'une tranche de centrale P.W.R. à une excitation sismique donnée, caractéristique du site de la Centrale.

2) OBJECTIFS PARTICULIERS

a) Analyse de l'interaction sol-fondation :

- Prise en compte des couplages entre bâtiments (79-82).
- Mise au point d'une méthode de détermination des ressorts et amortisseurs de sol (79-82).
- Effets des non-linéarités du sol (79-83).
- Amélioration de la représentation de la source et des ondes sismiques dans le sol au voisinage du radier (83-85).

b) Amélioration des méthodes d'analyse sismique (83-85) :

- Etude critique de la règle de combinaison quadratique des réponses modales.
- Mise au point de la méthode statistique d'analyse sismique.
- Interaction bâtiments-composants.

3) INSTALLATIONS EXPERIMENTALES -

Action en relation avec les essais sur maquette effectués à NICE.

4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS -

a) En ce qui concerne l'interaction sol-fondation, un ensemble de méthodes ont été développées et validées sur les résultats des essais de NICE.

- couplages entre radiers : mise au point du programme SOPHONIE.
- calculs des ressorts et amortisseurs de sol à l'aide du programme INCA adapté en conséquence.

- introduction d'un modèle de comportement non linéaire du sol dans INCA et analyse sur le cas d'une centrale type de l'influence de la non-linéarité du sol.

b) En ce qui concerne les méthodes générales d'analyse sismique une méthode de calcul direct des spectres de plancher et une méthode de calcul sismique des structures fixées en plusieurs points ont été mises au point.

5) PROCHAINES ETAPES -

- Poursuite de l'amélioration des méthodes de calcul de l'interaction sol-fondation (notamment de la représentation correcte des ondes sismiques dans le sol).
- Poursuite de l'amélioration des méthodes d'analyse sismiques.

6) RELATIONS AVEC D'AUTRES ETUDES -

142.1.09 : APPROCHE INTEGREE DU RISQUE SISMIQUE - MARGES DE SURETE DANS LES STRUCTURES DES INSTALLATIONS NUCLEAIRES.

7) DOCUMENTS DE REFERENCE -

- Note EMT 78/10 : Analyse sismique de structures fixées en plusieurs points.
- Communication 5ème SMIRT - BERLIN 1979 : Méthode directe de calcul des spectres de planchers. *
- Rapport EMT/SMTS/VIBR/80/14 : Méthode d'analyse sismique de structures de centrales nucléaires.
- Note Technique EMT/SMTS/VIBR/80/40 : Etudes sismiques des structures internes d'un réacteur P.W.R. - Vérification de la validation de la modélisation en poutre.
- Note Technique EMT/SMTS/VIBR/80/99 : Etude sismique des structures internes d'un réacteur P.W.R. - Modélisations fines en coques AQUAMODE et TRICO.
- Note Technique EMT/SMTS/VIBR/80/131 : Etude sismique des structures internes d'un réacteur P.W.R. - Recherche des premiers modes propres sur deux modélisations (TRISTANA et TRICO).
- Note Technique EMT/SMTS/VIBR/79/68 : Généralisation de la méthode des ressorts de sols et amortisseurs de sols pour les sols en couches et les fondations enterrées.
- Note Technique EMT/SMTS/VIBR/81/94 : Raideurs statiques et fonctions d'impédance d'une fondation rigide circulaire sur un demi-espace ou un sol en couches.
- Note Technique EMT/SMTS/VIBR/80/41 : Programme de calcul SOPHONIE - Impédances de fondations - Applications : radier rigide - radiers sur plots - radiers voisins.

- Thèse de Docteur-Ingénieur de T. ADYEL : Amélioration des méthodes de calcul des interactions sol-fondation pour la détermination de la réponse sismique des centrales nucléaires - Interprétation d'essais sur maquette .*

- Thèse de Docteur-Ingénieur de MMe FARVAQUE (à paraître) .*

8) DEGRE DE DISPONIBILITE

Documents internes non disponibles, sauf accord du CEA, à l'exception de ceux repérés par le signe *.

142.1.02	3 - 1
TITRE INTERACTION SOL-STRUCTURE MECANIQUE DU SOL AU VOISINAGE DES INSTALLATIONS ANALYSE PARASISMIQUE D'UNE CENTRALE NUCLEAIRE	*Pays* FRANCE *Organisme directeur* CEA/IPSN
TITRE en anglais SOIL-STRUCTURE INTERACTION SOIL MECHANICS IN THE VICINITY OF INSTALLATIONS SEISMIC ANALYSIS OF A NUCLEAR POWER PLANT	*Organisme éxécuteur* CEA/DMT SACLAY CEA/DSN/SAER FONTENAY AUX ROSES *Responsables* Me GANTENBEIN DEMT J. CULAMBOURG DSN/SAER
Date de démarrage 1978 — *Etat actuel* En cours *Date d'achèvement* 31.12.1985 — *Dernière mise à jour* Mars 1983	*Scientifiques* D.COSTES DSN/SAER B.MOHAMMADIOUN DSN/SAER

1) OBJECTIF GENERAL

Evaluer la sensibilité des résultats obtenus par les méthodes de calcul d'intéraction sol-structure couramment utilisées par rapport aux divers paramètres du sol et de la structure.

2) OBJECTIFS PARTICULIERS

2.1. - Traitement de bâtiments sur fondations séparées (1980-1983).

2.2. - Qualification par des essais in-situ des programmes de calcul relatifs à l'intéraction sol-radier plat (1981-1983).

2.3. - Mise au point de méthodes de calcul pour traiter les fondations sur pieux (1981-1983).

2.4. - Calcul d'impédances de sol liées à diverses configurations de sol (1983-1984).

2.5. - Etude de comportement non-linéaire du sol sous chargement dynamique (1983-1985).

3) INSTALLATIONS EXPERIMENTALES

ESSAIS IN SITU

Afin de valider les méthodes de calcul, l'étude de la réponse de maquettes d'un bâtiment réacteur (PWR 900) dotées de fondations de deux types, radier plan ou pieux, a été effectuée.
L'excitation représentative d'un signal sismique a été obtenue à l'aide d'une masse de 150 tonnes tombant d'une hauteur variable (de l'ordre de 20 mètres). Le site était le chantier de l'aéroport de NICE.

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4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

L'objectif 2.1 a été atteint pour des radiers plats.
Une interprétation des essais de NICE correspondant aux objectifs 2.2. et 2.3. a été effectuée en 1981 et 1982 par le DEMT.
Des calculs ont permis d'évaluer les impédances du sol et ont donné un bon accord avec les déplacements observés lors des essais.

On a mis en évidence l'influence de la non-linéarité du sol sur la variation du module d'élasticité en fonction de la déformation.

Une méthode de calcul adaptée au programme par élèments finis INCA a été mise au point au DEMT pour l'interprétation des maquettes sur pieux.

5) PROCHAINES ETAPES

- Développement du calcul d'impédances de sol pour diverses configurations de sol.

- Etude du comportement non linéaire du sol.

6) RELATIONS AVEC D'AUTRES ETUDES

- Méthodologie pour le calcul des spectres des séismes de référence à partir des paramètres physiques. Fiche étude 121-1-03.

- Surveillance de la sismicité des sites nucléaires. Fiche étude 121-1-04.

- Collecte de mesures sur les mouvements en zone épicentrale et d'information sur les dégâts correspondants. Fiche étude 121-1-05.

7) DOCUMENTS DE REFERENCE

1- Intéraction sol-fondation. Relations entre les mouvements de plusieurs zones à la surface d'un milieu élastique semi-défini. Note technique EMT/78/23.

2- Programme de calcul SOPHONIE. Impédance de fondations. Note technique EMT/SMTS/VIBR/80/41.

3- Dynamic behaviour of raft and pile foundations ; tests and computational models. Part 1 : General description and first interpretation. J. BETBEDER et AL. SMIRT 1981, PARIS.

4- Part. 2 : Ground motion induced by the compacting machine. H. FERRIEUX, G. MOHAMMADIOUN, SMIRT 1981, PARIS.

5- Part. 3 : 3D. dynamic analysis of group of piles and comparison with experiments. D. AUBRY, F. CHAPEL SMIRT 1981, PARIS.

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6 - Amélioration des méthodes de calcul des intéractions sol-fondation pour la détermination de la réponse sismique des centrales nucléaires. Interprétation d'essais sur maquettes.
T. ADYEL (Thèse de docteur ingénieur, 17 décembre 1982).

8) DEGRE DE DISPONIBILITE

Disponibles : /3/, /4/, /5/, /6/.
Autres documents internes, non disponibles sauf accord du CEA.

142-1-09			3 - 1
TITRE APPROCHE INTEGREE DU RISQUE SISMIQUE - MARGES DE SURETE DANS LES STRUCTURES DES INSTALLATIONS NUCLEAIRES			*Pays* FRANCE *Organisme directeur* CEA/DSN
TITRE en anglais COMPREHENSIVE APPROACH OF SEISMIC RISK - SAFETY MARGINS IN STRUCTURES OF NUCLEAR PLANTS			*Organisme éxécuteur* CEA/DAS/SAER FONTENAY-aux-ROSES *Responsables* D. COSTES DAS
Date de démarrage	3/1/80	*Etat actuel* en cours	*Scientifiques* R.J.GIBERT DEMT/SMTS
Date d'achèvement	1/1/85	*Dernière mise à jour* Mai 83	F.GANTENBEIN " J.GAUVAIN "

1) OBJECTIF GENERAL -

Pour les besoins de la sûreté, l'analyse de la tenue des structures vis-à-vis des séismes est effectuée eu égard aux critères des règles de l'art actuelles. Il convient de soumettre les méthodes, les critères et l'appréciation des marges de sécurité qui en découlent à une estimation critique. Celle-ci, probabiliste par sa nature, ne visant pas à la modification des critères réglementaires utilisés à la conception, est destinée à établir, suivant une procédure intégrée, l'enchaînement des évènements, une fois la sollicitation sismique donnée, jusqu'à l'éventuelle ruine des structures de génie civil et des équipements.

2) OBJECTIFS PARTICULIERS

- Point des documents d'orientation (1982-1983).
- Rassemblement des données expérimentales sur la tenue ultime des matériaux et structures (1982-1984).
- Analyse des données compte tenu de la réalisation et de l'histoire de la ruine : validation expérimentale (1983-1985).
- Etude de sensibilité sur les différents paramètres et validation (1983-1985).
- Application à l'étude de la boucle primaire d'un P.W.R. 1300 (1983-1985).

3) INSTALLATIONS EXPERIMENTALES -

Néant.

4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS -

Rassemblement de données expérimentales - Modélisation des structures Génie Civil d'une centrale P.W.R. 1300 - En cours : modélisation d'une boucle primaire.

5) PROCHAINES ETAPES -

Poursuite de la modélisation sismique du 1300 - Etude de sensibilité des différents paramètres.

6) RELATIONS AVEC D'AUTRES ETUDES -

Fiche 142-01-01 : Analyse parasismique d'une centrale nucléaire P.W.R. - Méthode de calcul.

7) DOCUMENTS DE REFERENCE -

- SMIRT 7 : Seismic analysis of a P.W.R. 900 reactor : Study of reactor building with soil-structure interaction - Evaluation of floor spectra (F. GANTENBEIN - M. AGUILAR).

8) DEGRE DE DISPONIBILITE

Document disponible

		CLASSIFICATION: 3.1
TITLE (ORIGINAL LANGUAGE): Studio sulla possibilità di previsione di terremoti con metodi idrogeochimici		COUNTRY: ITALY SPONSOR: ENEA
TITLE (ENGLISH LANGUAGE): Study on the possibility of predicting earthquakes by hydrogeochemical methods		ORGANISATION: ENEA 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) : Sviluppo di strumentazione e misure sismiche per la valutazione dei siti	COUNTRY : Italy SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : Seismic instrumentation development and seismic measurements for site evaluation	ORGANISATION : ENEA PROJECT LEADER : R. Cervellati
INITIATED : May 1974 | COMPLETED : STATUS : In progress | LAST UPDATING : March 1982	SCIENTISTS : P. Capocecera, G. Rienzo, F. Vitiello

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 stations and networks; 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 ENEA- 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.
2. Seismometric equipments have been set up, and are operated, in order to obtain a contribution to characterization of sites from the seismological point of view (determination of the earthquake mechanisms, hypocenters, etc.).

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Seismic instrumentation development and seismic measurements for site evaluation	3.1

3. Seismic networks have been set up. One is presently being installed in Valle del Sacco (Southern Latium). It will consist of 8 radio-linked seismometric stations. Another network (6 points, not radio linked) is operating around Brasimone (Bologna).

4. Equipments have been set up for direct transfer of seismic data into a computer.

Next steps:

- Cable telemetry.

Relation to other projects:

ENEA, ENEL programs (3.1).

Reference documents:

- Reports on the 1976 Friuli and on the 1980 Irpinia earthquakes.
- Studies on performance of accelerometers of the network; studies on performance of seismometers.

Degree of availability: Open

Contact person: R. Cervellati, ENEA, Istituto Casaccia, CP 2400, I-00100 Roma.

Additional information:

1. The research is performed in cooperation with ENEL and Istituto Nazionale di Geofisica. In particular a Joint Commission ENEA-ENEL has been established to study the seismicity of Italian territory for future nuclear power plants.

2. Personnel: 11 persons involved
   Budget: ~900 millions Lit (F.Y. 1982, besides cost of personnel).

	CLASSIFICATION : 3.1 11.4 14
TITLE (ORIGINAL LANGUAGE) : Comportamento e sicurezza dei componenti strutturali di impianti nucleari in zona sismica.	COUNTRY : ITALY SPONSOR : CNR - UNIV.OF PAVIA
TITLE (ENGLISH LANGUAGE) : Behaviour and Safety of Nuclear Power Plant Structural Components Subject to Seismic Actions	ORGANISATION : UNIVERSITY OF PAVIA PROJECT LEADER : Prof.F.CASCIATI

INITIATED :	COMPLETED :	SCIENTISTS :
1979	1983	Prof.L.Faravelli Prof.A.Gobetti
STATUS : IN PROGRESS	LAST UPDATING : SEPTEMBER 1982	

1. General Aim: To set up a procedure for seismic risk assessment

2. Particular Objectives: To define a probabilistic failure criterion
   To introduce model uncertainty

3. Experimental Facilities and Program : =

4. Project status: Computer codes capable of performing structural dynamic analyses and of simulating artificial ground motion are available; suitable probabilistic safety criteria have been introduced.

5. Nex steps: The Introduction of model uncertainty will be pursued. Particular care will be devoted to the uncertainty that affects the definition of the material constitutive law

6. Relation to Other Projects and Codes: Research in non linear analysis and seismic engineering are also developed independently of MMP applications.

7. Reference Documents: -Casciati F., Faravelli L., Gobetti A. - Inelastic Behaviour of Nuclear Power Plant Structural Components subject to Stochastic Ground Motions, CSNI Report N.44, 180, II, 70
   -F.Casciati - Probabilistc analysis of inelastic structures - Estratto dal Proc. 3rd Int.Seminar on Reliability of Nuclear Power Plants, Paris, August 1981, pp.35-41

8. Degree of Availability: The results are available without restrictions

9. Personnel involved: 3 researchers, 5 technicians (part time).

Prof. F. CASCIATI, Ist. Scienza e Tecnica Costruzioni, Università,
V. Luino 12, 27100 PAVIA

	CLASSIFICATION : 3.1.
TITLE (ORIGINAL LANGUAGE) : Ricerche strutturali e sismotettoniche ai fini della sicurezza degli impianti nucleari	COUNTRY : Italy SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : Structural and seismotectonic research for the safety of nuclear plants	ORGANISATION : ENEA PROJECT LEADER : G. Magri

INITIATED :	COMPLETED :	SCIENTISTS :
January 1975		V. Cagnetti, C. Carrara, F. Carlin, C. Cochi, G. Dai Pra, D. Molin
STATUS : In progress	LAST UPDATING : June 1982	

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

Geophysical apparatuses.

4) Project Status

Progress to date:
1) Paleogeographical data as well as Quaternary geological data have been collected.
2) The seismotectonic features of various Italian regions have been studied. (historical seismicity, instrument recorded seismicity, etc.).

5) Next Steps

Development of seismotectonics studies in Southern Latium, Central Apennine Region, etc.

6) Relation to other Projects: 3.1. (other ENEA, ENEL programs).

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Structural and seismotectonic research for the safety of nuclear plants	3.1.

7) Degree of Availability: Free; G. Magri, ENEA, CRE Casaccia, C.P. 2400, I-00100 Roma, Italy.

8) Additional Information
Personnel involved: 12 persons.
Budget (1982): 600 millions Lit., excluding personnel expenses.

	CLASSIFICATION: 3.1.
TITLE (ORIGINAL LANGUAGE) : Analisi di rischio sismico per possibili siti nucleari	COUNTRY : Italy
	SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : Seismic hazard analysis for candidate nuclear sites	ORGANISATION : ENEA
	PROJECT LEADER : M. Basili
INITIATED : 1976	COMPLETED :	SCIENTISTS : O. Iacurto
STATUS : In progress	LAST UPDATING : October 1982	

1. General aim

   Assessment of the maximum expected ground shaking on a given nuclear site via a probabilistic approach for the definition of the contributing factors of seismic risk mapping.

2. Particular objectives

   1. Statistical analysis of seismic occurrences and stochastic modeling of sequences.
   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.
   - Statistical analysis of seismic catalogues.
   - Mapping of seismogenetic structures.

   Essential results
   - Development of computer codes.
   - Completeness analysis of seismic catalogues.
   - Development of new stochastic functions in earthquake occurrence models.
   - Statistical analysis of a catalogue of historical seismic occurrences in the Central part of Italy.

TITLE (ENGLISH LANGUAGE): Seismic hazard analysis for candidate nuclear sites	CLASSIFICATION: 3.1.

- Statistical analysis of a catalogue of historical seismic occurrences in the Northeastern part of Italy.
- Statistical analysis of a catalogue of historical seismic occurrences in Mugello and Forli areas.

4. Next steps
   Seismic risk analysis of selected areas.
5. Relation to other projects
   3.1. Other ENEA projects.
6. Reference documents
   N. Pacilio, M. Basili, Seismic risk: (3) some stochastic functions with hazard implications - CNEN RT/AMB(80)2
7. Degree of availability: free
   Please contact: Laboratorio Ingegneria Siti, PAS-ISP, ENEA, CRE Casaccia, C.P. 2400, I-00100 Roma
8. Additional information
   Personnel involved: 4 persons
   Budget (1982): 200 millions Lit.

	CLASSIFICATION : 3.1.
TITLE (ORIGINAL LANGUAGE) : Studi di ingegneria dei siti	**COUNTRY :** Italy **SPONSOR :** ENEA
TITLE (ENGLISH LANGUAGE) : Studies of site engineering	**ORGANISATION :** ENEA **PROJECT LEADER :** M. Basili
INITIATED : 1974 / **COMPLETED :**	**SCIENTISTS :** A. Fontanive, A. Fels, V. Gorelli
STATUS : In progress / **LAST UPDATING :** October 1982	

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 strong motion accelerograms
- 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) Codes and experimental techniques have been developed.
3) Accelerograms from Friuli, Norcia and Irpinia earthquakes have been processed.

4) Next Steps

Besides development of above items: research on correlations between seismic parameters and sands liquefaction.

5) Relation to other Projects

3.1. ENEA, ENEL programs.

6) Reference Documents

1. Reports on the Friuli, Norcia, Irpinia earthquakes published by the ENEA-ENEL Commission on Seismic Problems Associated with the Installation of Nuclear Plants; e.g.: "Strong-motion Earthquake Accelerograms-Digitized and Plotted

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Studies of site engineering	3.1.

Data - Uncorrected Accelerograms-Part 1, 2, 3, 4, 5, 6", Roma, 1976-1980.

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 Siti, ENEA, CRE Casaccia, C.P. 2400, I-00100 Roma.

8) Additional Information

The research is performed in cooperation with ENEL and other Italian organizations; in particular a Joint Commission ENEA-ENEL has been established to study the seismicity of Italian territory for future nuclear power plants.

Personnel involved: 12 persons
Budget (1982): 500 millions Lit.

	CLASSIFICATION : 3.1
TITLE (ORIGINAL LANGUAGE) : Rete di rilevamento sismico	COUNTRY : ITALY SPONSOR : ENEL
TITLE (ENGLISH LANGUAGE) : Seismic monitoring network	ORGANISATION : ENEL PROJECT LEADER : DCO/Settore siti e ambiente
INITIATED : 1973 | COMPLETED :	SCIENTISTS :
STATUS : In progress | LAST UPDATING : September 1982	

1. General Aim

Characterization of earthquake activity for the location of candidate nuclear sites.

2. Particular Objectives

Collection of:
- time histories and response spectra of earthquakes in the different Italian regions;
- correlations 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 1g and with a threshold of 0.01g. The accelerographs are generally located inside electrical substations and installed on con-

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION :
Seismic monitoring network	3.1

crete columns directly anchored to the soil.

4. Project status

The seismic network has been completed. The main earthquakes recorded by the network were the earthquakes occurred in: Friuli on May 1976, Sicilia on April 1978, Valnerina on September 1979, Irpinia on November 1980.
Furthermore a computer program has been developed which enables to obtain the seismic spectra (acceleration, velocity and displacement and their envelops)
The results of data processing have been published by "ENEA-ENEL Commission on Problems Associated with the Installation of Nuclear Plants" .

5. Next steps

The network is working to collect strong-motion records of future earthquakes.

6. Relation to other projects

Joint Commission ENEA-ENEL to study the seismicity of Italian territory for future nuclear power plants.

7. Reference documents

Reports on the Italian eartquakes (1976-1982) published by "ENEA-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
f - Id., Part 5, Rome, July 1979
g - Id., Part 6, Rome, December 1980
h - Campania-Lucania earthquake on 23 november 1980 - Accelerometric recordings of the main quake and relating processing. Udine, May 1982.

(ENEL, Direzione delle Costruzioni - Settore siti e ambiente - Via G.B. Martini, 3 00198 - ROMA)

	CLASSIFICATION : 3:1
TITLE (ORIGINAL LANGUAGE) : Studier av svenska jordskalv	COUNTRY : Sweden SPONSOR : SKI
TITLE (ENGLISH LANGUAGE) : Research on Swedish Earthquakes	ORGANISATION : FOA PROJECT LEADER : Ragnar Slunga
INITIATED : 1979 / COMPLETED : 1983 STATUS : in progress / LAST UPDATING : June 1983	SCIENTISTS : Ragnar Slunga, Peter Norrman, Anne-Christine Glans

1. General aim: To study the seismic risks for the Swedish nuclear power plants.

2. Particular Objectives: Estimate the ground motions associated with the Swedish earthquakes.

3. Experimental Facilities and Programme: A digital seismological network of 21 stations covering southern Sweden is operated.

4. Project status:

   1. Progress to date: The network has been operating since the end of 1979 and some 150 erthquakes within the area recorded.

   2. Essential results: The ground motions associated with the small earthquakes so far observed have been studied and a two parametric (seismic moment and stress drop) scaling of the Swedish earthquakes has been developed.

5. Next steps: A still more detailed study of the physics of the earthquakes and relating these to the observable faults.

6. ---

7. Reference Documents: The FOA report "Research on Swedish Earthquakes" C 20477-T1 by Ragnar Slunga, Nov 1982.

8. Degree of availability: On request to Ragnar Slunga, FOA 290, Box 27322, S-102 54 STOCKHOLM, Sweden.

	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: MAHG ALDERSON
INITIATED: JANUARY 1978 / COMPLETED:	SCIENTISTS: DR P WINTER
STATUS: IN PROGRESS / LAST UPDATING: MAY 1980	

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 issued
   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:- Unclassified report SRD R 135

9. ADDITIONAL INFORMATION:-

	CLASSIFICATION: 3.1
TITLE (ORIGINAL LANGUAGE): SEISMIC ANALYSIS (GSD2.1) WITH APPLICATION TO PWRS	COUNTRY: UK
	SPONSOR: UKAEA
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/UKAEA
	PROJECT LEADER: RNH McMILLAN
INITIATED: DECEMBER 1978 / COMPLETED:	SCIENTISTS/ENGINEER: MARC ALDERSON
STATUS: / LAST UPDATING: SEPTEMBER 1980	

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 SARSPEC SAP IV NONSAP FLUSH

(b) Experimental No UK facilities identified.

OVERSEAS COLLABORATION

France: CEA/EDF/AEA/NPC/HSE information exchange

3.2. MISSILES

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1982 - 31.12.1982	3.2	150 0 121 (RS121)
Vorhaben/Project Title 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)		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Bundesanstalt für Materialprüfung (BAM), Berlin
Arbeitsbeginn/Initiated 1.10.1977	Arbeitsende/Completed 31.12.1983	Leiter des Vorhabens/Project Leader Brandes
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

1. General Aim

The safety analysis of nuclear power plants includes the "External Event" aircraft crash on the containment of the reactor building or on auxiliary buildings.

The design of containment structures against aircraft crash is based on several assumptions on the mechanical behaviour of reinforced concrete structures. The aim of the project is a contribution to the verification of some of these assumptions.

2. Particular Objectives

The investigation in the project is mainly directed to experimental work. The purpose is a description of the mechanical behaviour of reinforced concrete structural members under impact of "soft" missiles including different failure mechanisms and strain rate effects in order to obtain a realistic value of the energy absorption capacity.

3. Research Program

3.1 Tests

Tests have to be performed on different types of specimens (beams and slabs) by varying the most important parameters: Thickness of specimens; type, location and percentage of bending reinforcement; stirrups; deflection-time-curve resp. load-time-curve; impact time.

3.1.1 Specimen Parameter Variation

Reinforced concrete beams (span l = 3.20 m)

Thickness	$h = 12 \ldots 50$ cm
Reinforcement	$\rho = 0.4 \ldots 1.5$ %

Reinforced concrete slabs (square, supported at corners, span:256cm)
Thickness $h = 16 \ldots 22$ cm
Reinforcement $\rho_x = \rho_y = 0.4 \ldots 1.5$ %, no stirrups.
(Slabs with stirrups are investigated in project 150 460)

3.1.2 Impact load

The load or deflection is prescribed as a function of time (impact time $T_s \geq 30$ ms), deflection rate (v at loading point):$10^{-4} \ldots 5$m/s

3.2 Evaluation

The purpose of the evaluation is the identification of the values of mechanical quantities defined in different mechanical models, mainly material characteristics (yield moment, ultimate rotational capacity in plastic hinges).

4. Experimental Facilities

The tests are performed in a servohydraulic testing setup operating under closed loop control, which has been especially designed to perform impact tests, (max F = 1000 kN, max. v = 8,5 m/s).

With respect to simple mechanical models (e.g. beams) the time histories of the different mechanical quantities, measured in each test, represent a complete set of data describing the mechanical behaviour of the model.

5. Progress to date

Up to now 40 beams and 6 slabs have been tested. Three Technical Reports (drafts) have been completed. The relation of plastic strain of reinforcement bars during test and the reached ultimate rotational capacity of plastic hinges was analyzed (Fig. 1).

6. Essential results

The tests on beams failing in bending mode show a considerable increase of the energy absorption capacity with the deflection-velocity for BSt 420/500 RK bending reinforcement.

7. Next Steps

After finishing the analysis of the beam tests some additional tests shall be performed.

8. Relation to other Projects

RS 165/149, RS 337, RS 467, 150 408, 150 410, 150 437, 150 456, 150 460.

9. Reference Documents

See half-yearly report II/82 (German)

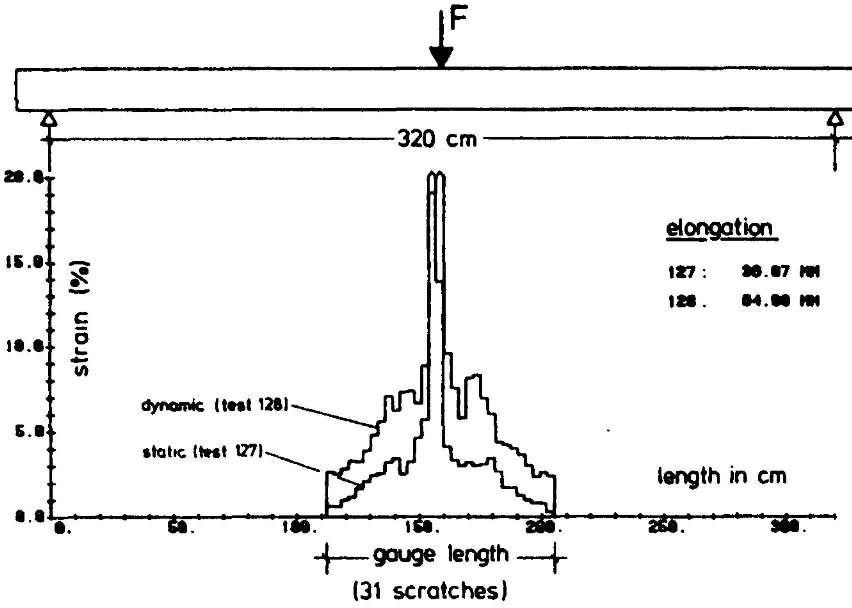


Fig. 1:

Influence of strain rate on permanent strains of the tensile reinforcement of a RC-beam (BSt 420/500 RK, d_s = 18 mm). Strain rate about 10^{-4} s^{-1} in static test no. 127 and about 2 s^{-1} in dynamic test no. 128).Total elongation means the plastic lengthering of the gauge length.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1982	3.2	150408 (and RS 467)

Vorhaben/Project Title	Land/Country
Theoretische Untersuchungen zur Ermittlung der kinetischen Grenztragfähigkeit von Stahlbetonplatten beim Aufprall stark deformierbarer Metallflugkörper Theoretical Investigations on the Kinetic Bearing Capacity of Reinforced Concrete Slabs under the Impact of Strongly Deformable Metal Missiles	FRG **Fördernde Institution/Sponsor** BMFT **Auftragnehmer/Contractor** Hochtief

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.8.1979	31.5.1983	Riech (coordination)
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1982	

1. General Aim

Within the scope of investigations on the protection of nuclear power plants against external events, the combined research projects No. 150408 and RS 467 are dedicated to the theoretical and experimental investigation of the essential problems connected with the loading case "aircraft crash".

2. Particular Objectives

Both above mentioned projects continue and improve the investigations of the preceding projects RS 165 and RS 149.

2.1 Partial Project No. 150408:

The preceding and the attendant theoretical studies within the scope of this project aim at contributing to the understanding of the following items:

- quantitative improvement of the calculation models to determine the impact history for the impact of deformable missiles onto rigid concrete structures
- determination of the failure mechanism of reinforced concrete slabs subjected to impact loading, laying special emphasis on the determination of the parameters characterizing a specified failure mechanism. (To this purpose, in addition to the tests of Project RS 467, the investigations must also evaluate a considerable number of small scale tests performed in other research projects).

2.2 Partial Project RS 467:

This project comprises large scale firing tests under use of deformable missiles and reinforced concrete slabs.

3. Research Program

3.1 Impact load/time characteristics during the impact of deformable missiles onto quasi-rigid reinforced concrete structures. (5 tests within RS 467).

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. (11 tests within RS 467).

3.2.1 Investigation of the influence of several design parameters on the local bearing capacity of the structural member: Variation of the concrete strength, shear reinforcement, thickness of the member, bending reinforcement.

3.2.2 Investigation of the 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 (ultimate bending bearing capacity) of reinforced concrete slabs.

4. Experimental Facilities, Computer Codes

4.1 Experimental Facilities (Partial Project RS 467)
Essential components are:
Gas-operated accelerator, target abutment, measuring and control equipment. All measurements are taken time-dependently during the total impact time.

4.2 Computer codes: (Partial Project No. 150408)
The following computer codes were compiled for the project RS 165 and will be further developed within the scope of the present project:

4.2.1 (see 3.1) Codes for the calculation of the impact load/time function for deformable missiles striking against rigid targets.

4.2.2 (see 3.2) Codes for the dynamic calculation of reinforced concrete slabs.

5. Progress to Date

5.1 Partial Project RS 467:

Six tests of test series II and one test of series I were realized.

Main objectives of the test series:

II/16 : Influence of the shape of the shear reinforcement (comparison with test II/14).

II/17 : Target type: the same as for test II/15; determination of the perforation velocity by reducing the impact velocity relatively to test II/15.

II/18 : Target type: the same as for test II/12; determination of the influence of the loading characteristic by using a modified missile.

II/19 : Repeat of test II/12 (changed steel grade for bending reinforcement (BSt 420/500 RK instead of BSt 420/500 RU); reduced concrete strength).

II/20 : Prevention of shear failure by increased quantity of shear reinforcement and modified shape of the stirrups (comparison with test II/15 and II/17).

II/21 : Influence of the target thickness on the bearing capacity of the reinforced concrete slab (comparison with test II/15 and with a series of small scale tests of UKAEA at Winfrith).

The tests are characterized by the following data:

test no. (date)	missile type / impact velocity m/s	target slab: thickness cm	target slab: bending reinforcement front % (e.w.)	target slab: bending reinforcement rear % (e.w.)	target slab: shear reinforcement cm² / cm²	test result d= rear heave cm
II/16 (5.5.1982)	type 11 247.1	70	0.22	0.45	50.2	rear scabbing max. d = 16
II/17 (26.5.82)	type 11 178	50	0.33	0.80	64.9	rear scabbing max. d = 12
II/18 (16.6.82)	type 17 237.4	70	0.29	0.57	52.3	rear scabbing max. d = 11
II/19 (13.12.82)	type 11 240.3	70	0.29	0.57	52.3	perforation
II/20 (3.11.82)	type 11 197.7	50	0.46	1.16	87.0	rear scabbing max. d = 15
II/21 (25.11.82)	type 11 237	90	0.12	0.28	26.5	rear cracks

-4-

For all tests deformable tubular missiles (mass = 1000 Kg) were used. The missile type 11 differs from type 17 by the wall thickness and the length.
The concrete strength of the targets reached values in the range of 33 to 39 MPa.

5.2 Partial Project 150408:

- The design of the test slabs for the Meppentests no. II/16 to II/21 and for 10 small scale tests at Winfrith was performed.
- Precalculations for the Meppen and Winfrith tests were set up.
- The evaluation and documentation of the Meppen hard missile test and a series of comparable small scale tests at Foulness and Winfrith has been completed.
- The evaluation of the Meppen tests II/11 to II/21 has been continued.

6. Results:

As the evaluation of the recent tests is not yet completed new results cannot be presented.

7. Next steps:

150408: Evaluation and documentation of the recent tests.

8. Relation with other projects:

RS165, RS149, RS121, RS337, 150410, 150416, 150437, 150460.

Berichtszeitraum/Period 1.1.82 - 31.12.82	Klassifikation/Classification 3.2	Kennzeichen/Project Number 150410
Vorhaben/Project Title Stahlbetonkonstruktion unter Flugzeugabsturzbelastung - Theoretische Nutzung der Meppener Versuche unter besonderer Berücksichtigung des Materialverhaltens Reinforced Concrete Construction Loaded by Crashing Aircrafts - Theoretical Utilization of the Experiments at Meppen Especially Considering the Behaviour of Materials		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Consulting Engineers Zerna, Schnellenbach und Partner GmbH
Arbeitsbeginn/Initiated 15.8.79	Arbeitsende/Completed 31.3.83	Leiter des Vorhabens/Project Leader Dr.Stangenberg,Dr.Nachtsheim
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

1. General Aim

Analytical and experimental response of reinforced concrete structures to extreme dynamic external loads due to impact of deformable missiles; beyond, the methods of safety assessment of nuclear power plants with respect to an aircraft impact will be improved. From the Meppen tests and from other experimental results, conclusions for practical applications are derived especially with regard to the dynamic material behaviour of reinforced concrete.

2. Particular Objectives

2.1 Developing analytical methods according to experimental results from Meppen tests; interpretation of the physical phenomena of the tests and preparation for numerical computations. The results from the UKAEA experiments on a reduced scale are included; the literature on the subject is taken into account.

2.2 Development of realistic failure mechanisms of reinforced concrete structures and evaluation of parameter dependences during impact loadings; Evaluation and updating of computer codes and theoretical models; Pre- and postcomputations of the experimental tests.

3. Research Program

3.1 Coordination with Hochtief with respect to the particular objectives of each test at Meppen, series II, and determination of parameters to be varied.

3.2 Participation in the discussions within the scope of BMFT-UKAEA cooperation about small scaled pretests.

3.3 Evaluation and updating of existing own computer codes with respect to special failure mechanisms during impact loads; Evaluation of parameter dependences. Parametrical pre-investigations and post-computations for all slab tests; comparison of numerical and experimental results. In case, development of new computer codes or parts of codes. Finally, adjusting of computational models and computer codes to the experimental results.

3.4 Improvement of methods of safety assessment and comparison with current practice of structural design.

4. Experimental Facilities, Computer Codes

Ref.to 3.3 The existing in-house computer code has the following capabilities: Physically non-linear dynamic step-by-step integration of the response of slabs to any load function; realistic material properties of reinforced concrete as multi-axial stress-strain relations up to the state of failure, and bond interaction between concrete and reinforcement.

5. Progress to Date

Ref.to 3.1 According to the test series II/16-18 and II/19-21, the test parameters have been fixed in detail, in co-operation with Hochtief.

In comparison with former Meppen tests, each one of the following parameter variations has been realized in at least one experiment: application of a different type of stirrups, closed on the rear face of slabs, and another type of projectile with a softer front part and a harder tail; reduction of the slab thickness from 70 to 50 cm as well as an increase to 90 cm.

Ref.to 3.2 The program for ten small scale tests to be performed by UKAEA has been determined in detail. Herein the slab thicknesses as well as the amount of reinforcement vary, whilst the nominal design level remains unchanged.

The conditions of some further experiments, carried out by BAM and by Karlsruhe University, have been fixed.

1.1.82 - 31.12.82 150410

Ref.to 3.3 The 3rd technical report has been completed, containing post-computations of slab tests 11 and 12.
To each of the tests 17 - 21, an extensive parametrical pre-investigation has been performed.

6. Results

The post-computations of slab tests carried out up to now show a good agreement between numerical and experimental results, especially with respect to the global deformation behaviour. Herein the actual material properties must be taken into account, particularly according to the yield strength of reinforcing steel.
With regard to the different relations between bending and shear bearing capacity, existing in each test, local concentrations of shear resp. punching distortions can arise (simulation of a punching cone).
From the tests in Meppen a great sensivity follows between the experimental results and the impact velocity, resp. referring to the deformation behaviour of projectiles. Thus the velocities to be associated with different limit states of slabs (scabbing resp. perforation), e.g., lie very close together. In the computational analyses, the same strong dependence exists with respect to the load histories from which the investigations have to start.

7. Next Steps

Ref.to 3.3 After availability of all results of the last Meppen tests, the analyses will be accomplished; the final report will be composed.

8. Relation to Other Projects

RS 165/149, 150 408, RS 467, 150 0121, 150 437, 150 460

Berichtszeitraum/Period 1.1.1982 - 31.12.1982	Klassifikation/Classification 3.2	Kennzeichen/Project Number 150 460
Vorhaben/Project Title Energieaufnahmevermögen von Stahlbetonbauteilen bei Stoßeinwirkung Energy Absorption Capacity of Reinforced Concrete Structural Members under Impact Force		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Bundesanstalt für Materialprüfung (BAM) Berlin
Arbeitsbeginn/Initiated 1.6.1980	Arbeitsende/Completed 31.12.1983	Leiter des Vorhabens/Project Leader Brandes
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31.12.1982	Bewilligte Mittel/Funds

1. General Aim

In the safety assessment of nuclear power plants with respect to aircraft crash there are necessary several assumptions, mainly covering the field of materials behaviour of reinforced concrete structures subjected to impact load. The aim of the project is the verification of some of these assumptions and to contribute to an accurate transfer of results obtained from small-scale test to the full size.

2. Particular Objectives

Three different fields are included in the investigation. All of them are treated to ensure the transfer of small-scale test results to reinforced concrete(RC)structures of nuclear power plants.

2.1 Mechanical Behaviour of RC-slabs under impact load

Locally loaded RC-slabs show different failure mechanisms: bending failure in yield lines or punching shear. The mechanisms both occur in static and dynamic tests. The appearance of the different failure modes and the behaviour of the slabs in these modes are influenced by:

- thickness of slabs,
- concrete strength,
- type, amount and location of shear and bending reinforcement,
- impact time and shape of load-time-curve.

These dependences shall be evaluated and quantified.

2.2 Strain rate effects of reinforcing steel

Yield stress, strength and total elongation in tension tests of reinforcing steel are influenced by the strain rate. This effect is essential in the dynamic behaviour of RC-structures and is a basis to interpret the test results in the different projects concerned with aircraft crash resistance and blast load resistance of RC-structures. The influence of strain rate shall be investigated in experiments.

2.3 Mechanical modelling and numerical analysis of RC-members under impact load

The behaviour of RC-structural members is synthesized from the properties of the components of RC (concrete, steel, bond) including strain rate effects.

3. Research Program

3.1 Tests shall be performed on about 20 slabs.
The square slabs (3 m x 3 m, supported at corners, 22 cm thick)are loaded centrally. The load is applied by a servohydraulic actuator, reaching a load increase within 2 to 5 milliseconds. Parameters varied in the program are:

concrete strength: $f'_c = 20 \ldots 60$ N/mm^2
bending reinforcement: $\rho = 0.3 \ldots 1.0$ %; $\rho' \approx 0.5 \cdot \rho$
shear reinforcement: $q_s = 0;\ 25;\ 45$ cm^2/m^2

3.2 Different types of reinforcement steel are included in the tests: BSt 420/500 RK, BSt 420/500 RU, BSt 1080/1320, diameter ≈ 20 mm.
Strain rates: $\dot{\varepsilon} = 0.00005;\ 0.2;\ 2;\ 8.5\ s^{-1}$

3.3 The investigation is performed using widely spread computer codes which enables the user to implement special procedures.

4. Experimental Facilities; Computer Codes

4.1 Tests are performed in a servohydraulic testing machine, max.force: 1000 kN, max. piston velocity: 4.5 m/s (8.5 m/s)

4.3 Computer Code ADINA is implemented on Cyber 175 computer (Wissenschaftliches Rechenzentrum Berlin - WRB)

5. Progress to Date

Tests on tensile specimens of reinforcing steel are completed.

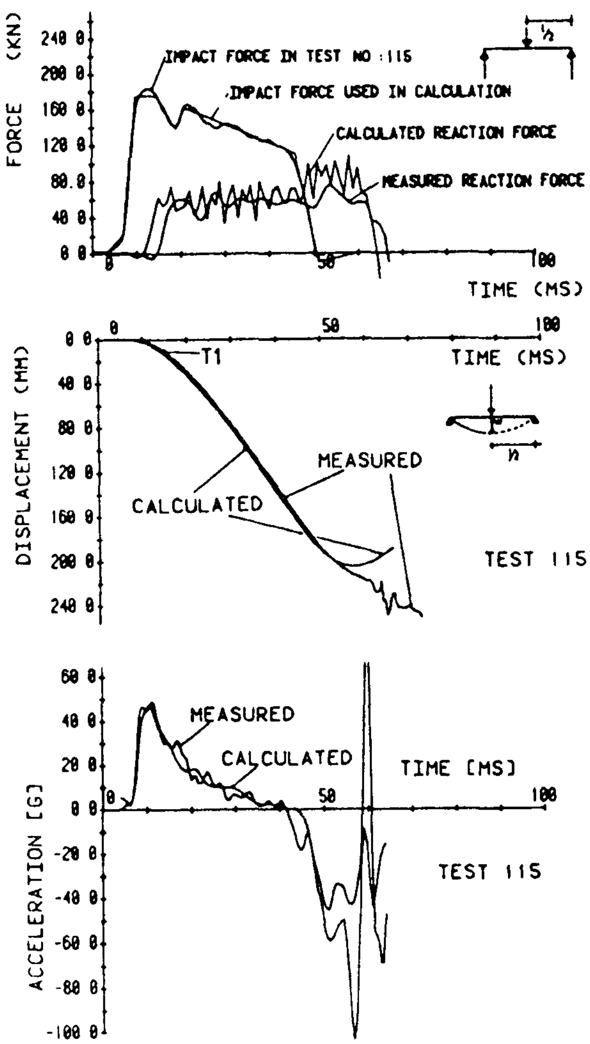
RC-slabs have been tested during the last year, varying different parameters (concrete strength, amount of shear reinforcement, displacement velocity). Results of Finite-Element-Analysis of RC-beams are presented in the figures.

6. Essential Results

Reinforcing steel is strain rate sensitive. This property is dependent upon the type of steel. The results of the FE-Analysis of RC-beams agree sufficiently with test results after including strain rate effects.

7. Relation to other Projects

RS 121, RS 165/149, RS 337, RS 467, 150 408, 150 410, 150 437.



8. Next steps

In 1983 shall be performed some more tests on slabs. The FE-Analysis of RC-beams shall be extended to a variety of beams.

9. Reference Documents

See half-yearly report II/1982

Figures 1 to 3.

Calculated mechanical behaviour of a RC-beam (test No.115) compared with the measured behaviour.

- Impact force-time curve and support force
- Comparison of calculated and measured displacement-time-curve.
- Acceleration-time-curves

Berichtszeitraum/Period 1.1.1982 - 31.12.1982	Klassifikation/Classification 3.2	Kennzeichen/Project Number 1500517
Vorhaben/Project Title Zugversuche mit Betonstählen unter hohen Beanspruchungsgeschwindigkeiten Tensile Tests with Reinforcing Steel Bars under Time-Dependent Load		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Universität Karlsruhe
Arbeitsbeginn/Initiated 1.1.1982	Arbeitsende/Completed 30.6.1983	Leiter des Vorhabens/Project Leader Henseleit
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31.12.1982	Bewilligte Mittel/Funds

1. General Aim

In the test projects /1, 2/ reinforced concrete beams were loaded up to failure under high loading velocities. The analysis of these test results showed a significant increase of the deformation capacity as well as of the maximum rupture stress when increasing the loading velocity. In this project it is therefore aimed to investigate whether the effect observed for the reinforced concrete beams can also be observed for dynamic tensile tests with other reinforcing steel qualities.

2. Particular Objectives

2.1 Investigation of the influence of the deformation velocity on the stress-strain diagrams. Reinforcing steel qualities mostly used for nuclear power plants are being used.

2.2 Quantification of the mechanical characteristics of several reinforcing steel qualities with various diameters in dependence of the strain velocity.

2.3 Investigation of the influence of a given strain history on the stress-strain diagram of reinforcing steel specimens.

3. Research Program

The following reinforcing steel qualities used for nuclear power plant construction, with various strain velocities ($\dot{\varepsilon}$ = 0,0005, 0,1, 1,0 > 2,0 m/m/sec) are being investigated:

1. BSt 420/500 RTS, Ø 25
2. BSt 420/500 RU, Ø 25
3. BSt 500/550 RTS, Ø 22
4. BSt 420/500 RTS, Ø 18

1.1.1982 - 31.12.1982 1500517

5. BSt 420/500 RTS, Ø 12

In addition, two representative strain histories are being examined on specimens of the Meppen test plates (RS 467).

4. Experimental Facilities

The experiments are being carried out using a computer-controlled servohydraulic test equipment. The unfinished specimens will be fixed in a hydraulic clamping device.

5. Progress to Date

5.1 Preparatory work for the tests

5.2 Providing the equipment and the steel specimens as well as mounting the test arrangement on the test field of the Institut für Massivbau und Baustofftechnologie

5.3 Carrying out preliminary tests

5.4 Begin with the main tests

6. Essential Results

Results can be expected only after finishing the evaluation of the tests carried out.

7. Next Steps

7.1 Completion of the remaining main tests

7.2 Evaluation of the measurement results

8. Relation to Other Projects

RS 165/149, 150 408, 150 410, RS 467, 150 460, 150 0121 (RS 121)

9. Referencer

/1/ Henseleit, O., Hehn, K.-H., Hoch, A.: Ultimate Bearing Capacity of Reinforced Concrete Beams under Time-Dependent Load, Test report of a research project of the Federal Ministry of Research and Technology (BMFT), RS 337, Karlsruhe 1980

/2/ Hoch, A.: Ultimate Bearing Capacity of Reinforced Concrete Beams under Impact, Test report of a research project of the Federal Ministry of Research and Technology (BMFT), 150 437, Karlsruhe 1982

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1982	3.2	RS 467
Vorhaben/Project Title		**Land/Country**
Experimentelle Untersuchungen an stoßartig belasteten Stahlbetonplatten Experimental Investigations at Reinforced Concrete Slabs loaded by Deformable Missiles		FRG
		Fördernde Institution/Sponsor
		Auftragnehmer/Contractor
		Bundesamt für Wehrtechnik und Beschaffung
Arbeitsbeginn/Initiated	**Arbeitsende/Completed**	**Leiter des Vorhabens/Project Leader**
1.8.1979	31.12.1982	Heine
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
	December 1982	

The report covering Project No. 150408 also contains Project RS 467.

	CLASSIFICATION: 3.2
TITLE (ORIGINAL LANGUAGE): THE EXPERIMENTAL AND THEORETICAL STUDY OF LOCAL EFFECTS IN THE IMPACT OF MISSILES ON STRUCTURES	COUNTRY: UK SPONSOR: SRD
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/AEEW UKAEA PROJECT LEADER: A R EDWARDS

INITIATED: MAY 1977	COMPLETED: 1983/4 APPROX.	SCIENTISTS: SRD J JOWETT, R L D YOUNG; AEEW P BARR, A NEILSON
STATUS: IN PROGRESS	LAST UPDATING: 29. 7. 80	

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 continues on the impact of missiles on concrete walls. This work was begun at AWRE Foulness. In particular scale modelling effects are being investigated. This is to be done in co-operation with other European experimental programmes. (France, Germany). The theoretical modelling of concrete behaviour under development 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 finit 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 capable of imparting an energy of about 1 MJ to a missile is being used. Maximum diameter is 150 mm. At a later stage of the programme a 300 mm diameter launch barrel will be added. This will increase the available energy to 3 MJ. High speed transducers with associated data processing are used together with high speed photography to monitor the impacts. Displac· ments and transmitted loads are routinely recorded. A Kistler load cell of capacity 10C · (200 te dynamic) enables the load delivered by a deformable missile's crushing to be measured.

A concrete construction and testing laboratory, capable of producing up to four concrete targets per week, with all associated quality control testing, is in use.

Scaling comparisons with large scale tests are available through a co-operation between GRS(FRG) and SRD. The larger tests are mounted at Meppen as part of a FRG programme.

3.3. EXPLOSIONS

	CLASSIFICATION : 3,3
TITLE (ORIGINAL LANGUAGE) :	COUNTRY : Denmark SPONSOR : CEC
TITLE (ENGLISH LANGUAGE) : Gas Explosion Characterization and wave propagation.	ORGANISATION : Risø Nat. Lab. PROJECT LEADER : S.I. Andersen
INITIATED : 1981.01.01 | COMPLETED : 1983.11.30 STATUS : In progress | LAST UPDATING :	SCIENTISTS : G. Larsen, J. Roed

1. General Aim:

Protection of Nuclear Power Plants against External Gas Cloud Explosions.

2. Particular Objectives:

The research project deals with finite-amplitude waves and shock waves arising from nonengulfing gas cloud explosions, and it particularly aims at an illumination of parameter variations for wave propagation through an inhomogeneous atmosphere and for wave/ boundary interactions influenced by the boundary material and boundary geometry.

The research project thus deals with:
a) influence of the boundary geometry (topography, vegetation, obstacles)
b) influence of the atmospheric stability and stratification,
c) influence of the soil boundary on wave propagation, following

a non-engulfing gas cloud explosion.

3. Experimental Facilities and Programme

Gas filled latex balloons up till 3 m in diameter is used as pressure source. A measuring system with up till 15 pressure measuring positions is established and incorporates tape recorder, transient recorder for digitizing and microcomputer for dataprocessing.

4. Project Status

Small scale experiments with 0.2-0.3 m diameter balloons are completed and the data analysed.

The test facilities for the 3 m diameter balloons has been established and the tests are being performed.

5. Next Steps

Evaluation and final report.

6. Relation to other Projects

The project is part of the CEC indirect action program on protection of NPP against gas cloud explosions.

Berichtszeitraum/Period 01.01.82 - 31.12.82	Klassifikation/Classification 3.3	Kennzeichen/Project Number 150 385
Vorhaben/Project Title Experimentelle Untersuchung zur Ermittlung des Druckfeldes infolge Wechselwirkung von Druckwellen mit Gebäudestrukturen (Teilforschungsprogramm Gasexplosionen). Experimental Investigations to Determinate the Pressure Field in Consequence of Interaction between Pressure Waves and Building Structures.		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Fraunhofer-Institut für Kurzzeitdynamik ERNST-MACH-INSTITUT
Arbeitsbeginn/Initiated 01.04.1979	Arbeitsende/Completed 28.02.1982	Leiter des Vorhabens/Project Leader Dipl.-Ing. G. Hoffmann
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating 31.12.1982	Bewilligte Mittel/Funds

1. General Aim

In the frame of the mainproject concerning gasexplosions the pressure-time-history of waves generated by an exploding gas cloud and interacting with buildings are to be studied by experimental simulation. The results of these investigations will enlarge the information of possible loads and stresses on nuclear power plants and thus support the calculation principles for the layout of nuclear installations.

2. Particular Objectives

If the pressure or shock waves originating at an exploding gas cloud interfere with a complex layout of buildings as it is found in nuclear power plants reflections and diffractions occur. The interaction of the different influences leads to a very intricate wave pattern. For the calculation of this extensive problem the support of experimental results is needed.

Depending on the type of the explosion the pressure or shock waves involve a large amount of different parameters as peak pressure, duration of positive and negative phase, impuls and load-characteristic. The particular aim of this project is to model the wave pattern generated by deflagration- and detonation waves interacting with complex building structures.

3. Research Program

3.1 Preliminary Work

Formation of a new working group, adaption of existing shock tubes; fixation of a representative layout of a nuclear power plant; preparing of the selected single buildings as models.

3.2 Simulation experiments with deflagrative profiles (peak pressure $\leq$0,3 bar). Tests with basic geometric structures (cuboid, cylinder, hemisphere).

3.3 Main tests.

3.3.1 Deflagrative load ($\leq$ 0.3 bar) on a model of a power plant.

3.3.2 Detonative load ($\geq$ 0.8 bar) on the same model.

3.4 Determination of the characteristic load parameters.

4. Experimental Facilities

Shock-tube: High pressure section: dia. 1 m

Test section : dia. 2.4 m

Registration and evaluation by a 16-channel Transient-Recorder-System and a Tektronix Desk Top Computer.

5. Progress to Date

Ref. 3.3 For the main tests with deflagrative load on the model of nuclear power plant (scale 1:200) the pressure waves were generated by a newly developed method. The driver section (1 m dia.) was reduced to a length of 0.15 m and separated by two plastic membranes. By destroying these membranes at the same time it was possible to generate a pressure pulse with over- and under-pressure phase of the same amplitude /1/. The model consisted of a scaled replica of a containment with different additional buildings. This set-up was loaded with pressure waves (deflagrative and detonative) from different directions. The peak amplitude for the deflagration wave that could be simulated without development of a shock was about 70 mbar. The peak blast wave overpressure was in the order of 0.7 bar, positive pulse durations in both cases being about 5 - 6 ms.

6. Results

Ref. 3.3 The evaluation of the experimentally determined pressure-time histories shows a magnification of the reflected pressure ratio to the free field value higher than for a normal reflection. Due to the focussing effects and superposition caused by the additional buildings this maximum pressure ratio reaches a value of about 4.3 in the case of blast wave-loading and a value of 2.5 in the deflagrative case.

Scaled net forces were calculated, which showed good agreement with values obtained for a real size building /2,3/. Evaluated pressure profiles on the containment are presented in a final report /4/ together with scaled forces. The results will also be presented at the 7. SMIRT, Chicago, 1983 /5/.

7. Next Steps

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8. Relation with other Projects

Part project: Gasexplosions (BMFT) : RS 318, 150411 to 150419

9. References

/1/ H.J. Thor
Belastung einer KKW-Anlage im Modellmaßstab mit simulierten Deflagrationswellen
Technischer Fachbericht, BMFT - 150 385 - 3. Juni 1981

/2/ H.J. Thor
Belastung von Reaktorgebäuden durch Druckwellen aus explodierenden Gaswolken
Transaction of 6. SMIRT, Paper J 10/10, Paris, August 1981

/3/ R. Zinn, F. Stangenberg
Response of PWR-Containment Structure to an External Blast Wave Using Different Geometric and Load Models
Transaction of 6. SMIRT, Paper J 10/12, Paris, August 1981

/4/ H.J. Thor
Modelluntersuchungen zur Belastung eines Kernkraftwerks durch Druckwellen aus Gasexplosionen
Abschlußbericht, BMFT - 150 385 - 4, Juni 1982

/5/ H.J. Thor
Experimental Simulation of Gas Cloud Explosion Effects on a Reactor Containment
Transactions of 7. SMIRT, Paper J 10/3, Chicago, August 1983

10. Availability of Reports

GRS mbH, Köln, Germany

Berichtszeitraum/Period 1.1.82 - 31.12.82	Klassifikation/Classification 3.3	Kennzeichen/Project Number 150 411
Vorhaben/Project Title Mogliche Initiierung detonationsähnlicher Explosionsformen in freien Gas-Luft-Gemischen und entstehende KKW-Beanspruchung Possible Initiation of Detonation-Like Explosion Modes in Free Gas-Air Mixtures and Resulting Nuclear Power Plant Load		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Battelle-Institut e.V.
Arbeitsbeginn/Initiated 15.8.79	Arbeitsende/Completed 30.9.82 (to be extended)	Leiter des Vorhabens/Project Leader W. Geiger
Stand der Arbeiten/Status	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

1. General Aim

Investigation of conceivable mechanisms for the initiation of detonation-like explosion modes in unconfined vapour clouds, assessment of the explosion modes with regard to the resulting loading and response of nuclear power plants ,taking into account the results of the other institutions participating in the subprogram Gas Explosions.

2. Particular Objectives

2.1 Examination of the possibility of initiation of detonation-like explosion modes by partial confinement

2.2 Determination of the principal conditions of turbulence generation and of spontaneous explosion of gas pockets trapped behind the flame front

2.3 Determination of the potential acceleration of the flame front due to localised spontaneous explosions behind the front

2.4 Assessment of the possibility of self-initiation in large dust-loaded vapour clouds due the preheating by radiation transfer from the flame

2.5 Assessment of the hazard presented by the induced ground wave in the case of a detonation-like explosion

2.6 Scientific coordination of the different groups participating in the subprogram Gas Explosions, compilation and integrated presentation of the results of the individual investigations.

3. Research Program

Corresponding to the six main objectives the project is divided into six subprojects:

3.1 Partial Confinement - Principal Mechanisms

Construction of the experimental setup; study of explosion buildup in the unconfined space due to jet initiation from the partial confinement; assessment with regard to real scale.

3.2 Spontaneous Explosion (McGill University, Montreal)

Basic theoretical and experimental investigations on turbulent flame propagation, quasi-detonations, detonation limits, SWACER initiation; assessment with regard to real scale.

3.3 Flame Acceleration by Localized Explosions

Planning and construction of the experimental setup; variation of explosion strength and ignition delay time in the experiments; assessment of results with regard to real scale.

3.4 Self-Initiation due to Radiation Transport (SAI, Calif., USA)

Numerical simulation of the preheating of dust-loaded clouds by radiation transfer and of the resulting flame acceleration.

3.5 Induced Ground Wave

Adaption of computer code DSIM to the problem given; specification of representative ground wave data and free-field air blast waves; performance of calculations.

3.6 Integration of the Battelle Investigations into the Overall Program

Scientific coordination of the Battelle investigations in conjunction with the other investigations within the subprogram; continuous observation and documentation of the results of the subprogram; establishment of a literature information center for the subprogram; compilation and assessment of the results obtained in the subprogram.

5. Progress to Date

Re 3.1 Additional jet initiation experiments with variation of size and shape of the orifice as well as of pressure buildup rate in the partially confined volume. Experiments on the influence of lateral walls (simulating a lane) in the free cloud.

Re 3.2 Experiments on the influence of gas concentration and degree of confinement on flame acceleration, investigations related to the question of maximum turbulent flame speed.

Re 3.3 Additional experiments with a setup allowing spherical instead of planar flame propagation, determination of flame acceleration by the blast wave generated behind the front, theor. analysis.

Re 3.5 Calculation of the ground wave induced by the detonation of a flat cloud for several sites with different types of idealized soil profiles.

Re 3.6 Scientific coordination of the different tasks of subprogram Gas Explosions, coordination meeting on the modelling of nonlinear structural loading due to gas cloud detonation, elaboration of revised load specifications for task 10.

6. Results

Re 3.1 Turbulence-inducing structures in the partially confined volume as well as lateral walls forming a lane in the free cloud enlarge the effect of considerable overpressures (of the order of 1 bar) over an extended distance in the free cloud.

Re 3.2 Flame acceleration and pressure buildup in repeated obstacle configurations are significantly reduced for off-stoichiometric mixtures and for configurations which are partly unconfined to above (as compared to tube geometry).

Re 3.3 When the blast wave generated by a local explosion behind the flame front interacts with the flame front, a Markstein-Taylor instability is induced. This leads to a very large increase of flame surface and hence to strong flame acceleration.

Re 3.5 Detonation of a flat cloud (radius 400 m, height 9 m) produces significantly larger accelerations and velocities at the cloud edge and beyond it than detonation of an ideal hemispherical cloud (radius 12 m).

7. Further work

Re 3.1 to 3.6 Work has essentially been completed. The compilation and assessment of the results from the various tasks of subprogram Gas Explosions is yet to be accomplished.

8. Relation to Other Projects

Subprogram Gas Explosions: RS 318, 150 385, 150 412, 150 413, 150 414, 150 415, 150 416, 150 418, 150 419; PNP Safety Research Program, subprogram "Process Gas Release", Gas explosion research program of the Commission of the European Communities.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.82 - 31.12.82	3.3	150 457
Vorhaben/Project Title: POSSIBLE INITIATION OF DETONATION-LIKE EXPLOSION MODES		Land/Country: FRG
		Fördernde Institution/Sponsor: BMFT
		Auftragnehmer/Contractor: Fraunhofer-Institut für Treib- und Explosivstoffe (ICT)
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.7.80	28.2.83	Dr. H. Pförtner
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing		-

1. General Aim

Investigations to estimate the effects of pressure waves from chemical reactions in which certain parameters let higher pressures be expected. Statements on possible hazards for a reactor plant involved in the respective explosion modes.

2. Particular Objectives

2.1 Possible initiation of detonation-like explosion modes with partial confinement by model simulation of the explosion accident of te Beek.

2.2 Propagation of an explosion ignited in a closed volume to the gas mixture in the surrounding space after failure of a vessel wall.

2.3 Determination of the blast wave parameters of non-stoichiometric gas mixtures in case of initiation in a detonative mode.

3. Research Program

According to the afore-mentioned individual objectives, the project has been subdivided into three particular tasks:

3.1 Partial Confinement

Scaled-down simulation of the explosion accident of te Beek, model scale 1 : 10 and 1 : 3. Ignition of stoichiometric mixtures of methane/air and ethylene/air in an outer space and in the partial confinement. Measuring of the local and temporal pressure course, and of the flame velocity inside and outside the partial confinement.

3.2 Isochoric Explosions

Determination of optimum failure of a vessel and of the propagation functions of bursting vessels at volumes of .5m³, 1 m³ and 2 m³. Measuring of flame velocities and pressure/time-characteristics of stoichiometric mixtures of methane/air and ethylene/air with central and peripheric arrangement of the vessel in a large hemispheric balloon ($\geq$ 50 m³).

3.3 Initiation of non-_stoichiometric mixtures in a detonative mode
Determination of the propagation functions (overpressure, duration of the positive pressure phase, positive impulse, time of arrival of blast wave) of detonating non- stoichiometric mixtures of ethylene/air. Determination of the minimum amount of high explosives required for initiation of five different concentrations. Simulation of a real cloud by initiation of various concentration layers in one balloon. Measuring of the local and temporal pressure course, and of the propagation velocity (detonation velocity). Correlation to the TNT-equivalent.

4. Experimental Facilities

Several acres of open-air terrain with a test site of 40 m diameter, quarry with test and measuring range, and another test site of 20 m diameter, short-time physical measuring equipments, high speed cameras, concentration measuring instruments.

5. Progress to Date

To 3.1: The provided tests were carried out according to the test plan, and the evaluation was terminated practically.
To 3.2: After the preliminary tests for optimum vessel failure the main tests were taken up and almost terminated.
To 3.3: The investigations were finished, and a report was prepared.

6. Results

To 3.1: In two partial confinements of different size stoichiometric methane/air and ethylene/air mixtures were ignited in a deflagrative mode inside and outside the partial confinement. The local and temporal pressure course and the flame velocity was measured. Whereas with ignition inside the partial confinement the velocity remains approximately constant, it increases slightly in the outer space as a consequence of a certain "jet-effect". With ignition in the outer space the velocity remains there practically constant, and decreases in the partial confinement as a consequence of the hindered expansion. The overpressures which are difficult to be analyzed because of multiple pressure peaks are obviously lower in the model scale 1 : 3 compared with the 1 : 10 model. The damages occured in te Beek are therefore due not only to the partial confinement but also to other mechanisms as for instant an additional excitation of turbulence.

To 3.2: As the investigations carried out are not jet fully evaluated no results can be given. As to be expected the overpressures, however, are higher than in a free, undisturbed "cloud".

To 3.3: In seven different ethylene/air mixtures the concentrations of which were between 4.15 % by vol. and 12.0 % by vol., the propagation characteristics were measured in the case of initiation by 40 grams of HE in distances between 2 m and 40 m in hemispherical balloons between 6 and 15 m³. Whereas the results in the detonation range (5 - 12 % by vol.) practically coincide with the values determined for the stoichiometric mixture, outside this range there are only shock waves supported by combustion with considerably lower overpressures. Outside the boundary of reaction products the results are consistent with the known propagation characteristics of HE so that a TNT equivalency can be defined. For the stoichiometric mixture this equivalency was determined to 5.8 for the pressure and about 3 for the impulse. The results from the multiple-balloons, in which there were several concentration layers between 4 % by vol. and 40 % by vol. ethylene in air, show when initiated within the detonation limits that only that portion reacts detonatively which is within the detonation range. When initiated outside the limits only detonation-like explosion modes with distinctly lower blast wave intensities occur.

7. Next Steps

Conclusive evaluation of the tests to 3.1 and 3.2 and preparation of the reports on these results.

8. Relation with Other Projects

"Release of Process Gas-Explosions in the Gas Factory and the Effects of Pressure Waves on the Containment" within the PNP Safety Programme, Gas Explosion Programme of the EC, RS 318, 150 383, 150 411, 150 412, 150 413, 150 414, 150 415, 150 416, 150 417, 150 418, 150 419.

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.82 - 31.12.82	3.3	150458
Vorhaben/Project Title Explosionsfähigkeit von Nebel/Dampf/Luft- oder Nebel/Gas/Luft-Gemischen Explosion possibility of mist/vapour/air- or mist/gas/air-mixtures		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Physikalisch-Technische Bundesanstalt Braunschweig Gruppe 3.4
Arbeitsbeginn/Initiated 1.9.80	Arbeitsende/Completed 31.12.82	Leiter des Vorhabens/Project Leader Prof. Dr. Steen
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

1. General aim

Safety aspects of explosive mist/gas/air- or mist/vapour/air-mixtures (hybrid mixtures), concerning the possibility and the effect of corresponding hybrid mixture cloud explosions.

2. Particular objectives

By explosion tests in a confinement the explosion process of hybrid mixtures shall be compared with that of corresponding gas/air-mixtures. Test data to be obtained are: maximum overpressure, maximum rate of pressure rise and velocity of flame spread.

3. Research programm

3.1 Review of the literature
3.2 Development and build-up of the test facilities
3.3 Preliminary tests: generation of artificial turbulence, preparation of the hybrid mixtures
3.4 Main tests: Pressure-time history of hybrid mixture explosions
3.5 Evalution of tests

4. Experimental facilities

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5. Progress to Date

To 3.3 The generated turbulence field has been investigated (rms-value of fluctuation-velocity, correlation length, frequency analysis). Drop-size distrubutions and mist concentrations have been measured as a function of the flow conditions at the atomizer-nozzles.

To 3.4 Explosion tests on some gas/air, mist/air- and mist/gas/air-mixtures have been conducted and gave data on maximum overpressure, rate of pressure rise and velocity of flame spread.

6. Results

To 3.3 The measured isotropy of the turbulence field was confirmed by a rather spherical flame-spread in turbulent propane/air explosions, which could be registrated by a high-speed camera. The local mist concentrations at higher liquid injection rates (mean drop size $\bar{d} \gtrsim 50\ \mu m$) show statistical fluctuations with time, which amount up to $\pm$ 50 % of the mean value $\bar{c} \leq 40\ g/m^3$ and which are presumably connected to larger turbulent eddies (characteristic length $l \gtrsim 0{,}1$ m).

A measure for violence of explosion, common in dust/air-explosions, is the number $K = V^{1/3}\ (dp/dt)_{max}$. For highly turbulent stoichiometric propane/air mixtures we got $K_G \lesssim 550$ bar ms^{-1} (maximum overpressure $p_{max} \approx 8$ bar). Mere mist/air-mixture explosions showed a maximum value of $K_N \lesssim 50$ bar ms^{-1} ($p_{max} \approx 6$ bar), surely implied by the rather lean fuel concentration.

Flame front velocities (v_F) have been deduced for some propane/air-explosions (4 % propane by volume) from high speed camera pictures as well as from time signals produced by the melting of fine wire probes. E.g. we find $v_F \simeq 4\ ms^{-1}$ for a mixture at rest and $v_F \simeq 15\ ms^{-1}$ for a moderate turbulent mixture. Simultaneous pressure reccords prove, that these data are related to the prepressure-period of the explosion ($\Delta p \lesssim 0{,}3$ bar) so that the confinement has nearly no influence on these results.

7. Next steps

Conduction of a test series on hybrid-mixtures and discussion of results in the final report

8. Relations with other Projects

Gasexplosions-Teilforschungsprogramm: RS 318, 150385, 150411, 150412, 150413, 150414, 150416, 150418, 150419, 150457

1.1.82 - 31.12.82 3 150458

9. References

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10. Degree of Availability of the Reports

GRS-Forschungsbetreuung

122-2-02				3 - 3
TITRE AGRESSIONS D'ORIGINE EXTERNE SUR LES INSTALLATIONS NUCLEAIRES : EXPLOSIONS CHIMIQUES NON CONFINEES DUES A UN ENVIRONNEMENT INDUSTRIEL OU AUX VOIES DE COMMUNICATION .				*Pays* FRANCE *Organisme directeur* CCE (DG XII) CEA/DSN/Fontenay/R EDF/SEPTEN
TITRE en anglais EXTERNAL IMPACTS ON NUCLEAR PLANTS : UNCONFINED CHEMICAL EXPLOSIONS DUE TO AN INDUSTRIAL ENVIRONNEMENT OR TO COMMUNICATION ROUTES .				*Organisme éxécuteur* CEA/DSN/SAER/Fontenay CEA/CESTA ENSMA *Responsables* JL.GARNIER CEA/DSN/SAER S.HENDRICKX EDF/SEPTEN
Date de démarrage	01.01.1976	*Etat actuel*	en cours	*Scientifiques* MM.BROSSARD-ENSMA LEYER -ENSMA PERROT -CEA/CESTA LANNOY -EDF/DER
Date d'achèvement	31.12.1983	*Dernière mise à jour*	mars 1983	

1) OBJECTIF GENERAL

Protection des installations nucléaires contre les 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 communications .

2) OBJECTIFS PARTICULIERS

- Recherche de lois d'échelle pour les caractéristiques de l'onde de pression aérienne engendrée par la détonation d'un mélange air-hydrocarbure . Conditions pour une initiation en détonation .

- Recherche de lois pour les caractéristiques de secousses telluriques induites par la détonation en surface d'un mélange air-hydrocarbure .

- 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 .

- 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 (déflagration ou pseudo-détonation) .

3) INSTALLATIONS EXPERIMENTATLES

Les essais sont actuellement réalisés au Centre d'Etudes Scientifiques et Techniques d'Aquitaine (CESTA), Terrain d'Expérimentation Extérieur (TEE) .

4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

4.1. - Programme AMEDE
Le programme AMEDE d'études de détonations de mélanges air-hydrocarbure

s'est achevé le 9 Février 1982 par un tir hémisphérique de 500 m^3 d'un mélange air-éthylène à 7,4 % .

L'exploitation des résultats obtenus a fait l'objet de rapports de synthèse . Cette synthèse conduit à un équivalent massique de 4 kg de TNT par kg d'hydrocarbure présent dans le mélange explosible .

4.2. - Programme TECMAH

Pour mémoire, aucun essai n'a été réalisé en 1982, priorité étant laissée aux essais suivants .

4.3. - Programme CHARLES

a) Déflagrations à grande échelle (Ballons de 12 m^3)

Rappelons que ces études effectuées en collaboration avec la CCE (financement par celle-ci à 50 %) comportent 4 phases :

1ère phase : déflagrations en milieu homogène . Pour mémoire 16 tirs effectués en 1981 . Phase achevée en 1981 .

2ème phase : déflagrations en milieu hétérogène .
Deux mélanges air-éthylène de concentration en hydrocarbure différente sont contenus dans deux ballons sphériques concentriques, en latex . Le ballon extérieur a un volume de 8,5 ou 11,5 m^3 environ, l'intérieur un volume de 0,5 ou 3 m^3 selon les tirs . Ces enveloppes sont détruites par un dispositif projetant des plombs de chasse préalablement à la mise à feu, effectuée au centre des deux ballons .

On se propose d'évaluer les conséquences sur le champ de pression engendré par l'explosion, de l'accélération de la flamme induite par la variation de concentration du mélange en hydrocarbure .

Après une période de mise au point des inflammateurs utilisés, deux tirs ont été effectués pour contrôler les conditions d'inflammation en milieu homogène (1 seul ballon) au voisinage des limites haute et basse d'inflammabilité (13 et 5 %) .

14 tirs ont été effectués de juillet 82 à Janvier 83 .

3ème phase : déflagrations rapides ou pseudo-détonations
Un premier essai (en milieu homogène dans cette phase) a été effectué le 27.9.82, sur un mélange air-éthylène à 8 % (plus riche que la stoechiométrie, égale à 6,54 %) . L'initiateur était constitué par un détonateur BRISKA renforcé de 5 g de plastic . Seul un régime de déflagration normale a été constaté . Devant le risque que fait courir au matériel d'expérimentation ce genre d'essais (si l'on obtient un régime de détonation franche), la poursuite de cette phase d'études a été repoussée après l'achèvement de la deuxième phase, c'est-à-dire en cours du premier semestre 1983 .

4ème phase : influence sur le champ de pression de certaines caractéristiques de l'explosion (forme de la charge, position de points d'inflammation, obstacles sur le trajet de la flamme)
Pour mémoire . Seul le programme de travail a été défini en 1982 et sera réalisé au 2ème semestre 1983, avec un dépassement prévisible en 1984 .

b) Interprétation des résultats expérimentaux

Les essais CHARLES ont permis de vérifier la validité d'un modèle acoustique précédemment mis au point par l'ENSMA à l'aide de volumes hémisphériques d'une dizaine de litres et ont montré que ce modèle était encore applicable avec précision au cas de charges d'un volume initial de 10 à 20 m^3,au travers desquelles la célérité de déflagration atteint au maximum 40 m/s . Il rend correctement compte des résultats expérimentaux observés lors de l'accélération du front de flamme, due au gradient de concentration entre les deux mélanges .

5) PROCHAINES ETAPES

5.1. - Programme TECMAH
Poursuite d'essais de déflagrations en tuyau (tube TECMAH - Tube d'Essais de Combustion de Mélanges Air-Hydrocarbures) en fonction des besoins liés à l'analyse de sûreté .

5.2. - Programme CHARLES
Poursuite d'essais de déflagrations à grande échelle (ballons de 12 m^3) sur le champ de tir du CESTA (programme financé conjointement par la CCE, EDF et le CEA, contrat n° SRF/005/F-(S) :

.achèvement de la phase n°3 (pseudo-détonation) en 1983,

.lancement de la phase n°4 (étude de l'influence de la forme du nuage, de l'emplacement du point d'allumage, des obstacles) en 1983 avec un débordement prévisible sur 1984 .

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 .

7) DOCUMENTS DE REFERENCES

1. Experimental study of the overpressures generated by the detonation of spherical air-hydrocarbon gaseous mixtures . ENS/ANS International Topical Meeting on Nuclear Power Reactor Safety . Bruxelles 16-19/10/1978 .

2. Flame propagation through unconfined and confined hemispherical stratified gaseous mixtures . Communications au 17th International Symposium on Combustion . Paper number 124 .

3. Effets de pression engendrés par la propagation des déflagrations en sepace libre . Laboratoire d'Energétique et de Détonique (Université de Poitiers) .

4. Procès-Verbal d'essais Air-Acétylène - CESTA/EX/ESP n°1354/76 .

5. Procès-Verbal d'essais Air-Ethylène - CESTA/EX/ESP n°0010/77 .

6. Additif - CESTA/EX/ESP n°0137/77 .

7. Procès-Verbal d'essais Air-Propane - CESTA/EX/ESP n°1214/77 .

8. Procès-Verbal d'essais Effets de Sol - CESTA/EX/ESP n°1512/77 .

9. Procès-Verbal d'essais Tirs de référence - CESTA/EX/ESP n°327/78 .
nouveau site .

10. Procès-Verbal d'essais hémisphériques - CESTA/EX/ESP n°923/78 .

11. Procès-Verbal d'essais Contrôle capteurs - CESTA/EX/ESP n°969/78 .
utilisés pour essais AMEDE .

12. Analyse des résultats expérimentaux obtenus au cours des essais air-acétylène et air-éthylène . Laboratoire d'Energétique et de Détonique . LA 193 - ENSMA - Université de Poitiers .

13. Etude systématique des facteurs d'accélération des déflagrations en espace libre . Laboratoire d'Energétique et de Détonique . LA 193 - ENSMA - Université de Poitiers .

14. Sur l'effet de sol au cours des essais AMEDE 2/1 à AMEDE 2/5 . Laboratoire d'Energétique et de Détonique . LA 193 ENSMA . Université de Poitiers .

15. Caractéristiques du champ de pression engendré par une flamme accélérée en espace libre par la présence d'obstacles simples et de confinements partiels . Laboratoire d'Energétique et de Détonique . LA 193 ENSMA - Université de Poitiers .

16. Effets de pression engendrés par l'explosion dans l'atmosphère de mélanges gazeux d'hydrocarbures et d'air . J.C. LEYER - Conférence à la Société Française des Thermiciens (Mai 1980) .

17. Propagation d'une onde de choc due à une explosion - Comparaison expérience - modèle BK-WAVE - EDF-DER - HP 219/79/53 . A.LANNOY (Août 79) .

18. Effet de surpression dû à une déflagration hémisphérique non confinée - EDF-DER - HP 219/79/58 - A.LANNOY .

19. Une comparaison entre un modèle de déflagration à vitesse variable et des essais de déflagration de mélanges air-hydrocarbure en milieu libre . EDF HP 219/81/25 A. LANNOY (Avril 1981) .

20. Essais de détonabilité de mélanges air-hydrocarbures - Essais TECMAH - Ordre d'essai - J.PERROT - CESTA/EX/ESP n°52 du 16.1.81 .

21. Essais d'explosions de mélanges air-hydrocarbures - Essais AMEDE - Traitements SIDEX relatifs à certains enregistrements sismiques : AMEDE 2/6 (tir n°2), AMEDE 2/8 (tir n°1) - CESTA/EX/ESP n°CI 432 du 27.3.81

22. Etude des comportements hydrocarbures-air réagissant librement en sphérique Essais CHARLES (phase 1) - Ordre d'essai - J. PERROT - CESTA/EX/ESP n°CI 494 du 7.4.81 .

23. Effets telluriques des explosions de mélanges hydrocarbures-air au-dessus de la surface du sol . (Point après les expérimentations du CESTA) - H. FERRIEUX - J. PERROT - DSN/SAER/81/29 de novembre 81 .

24⁑ Essais CHARLES (phase 1) . Qualification du système d'effacement d'enveloppe - Compte rendu d'essai - J. PERROT - CESTA/EX/ESP n°CI713 - 2.6.81.

25⁑ Essais CHARLES (phase 1) Déflagration de mélanges homogènes en milieu non confiné - Compte rendu d'essai - J. PERROT CESTA/EX/ESP n°CI1555, 26.11.81

26⁑ Caractérisation du champ de pression induit par l'explosion aérienne d'un mélange air-hydrocarbure. Déflagration lente, déflagration rapide, pseudo-détonation. Phase 1, Rapport d'interprétation des essais CHARLES (déflagration lente) J.C.LEYER, ENSMA - 22.12.81. Contrat CCE/CEA - SRF/005/F(S)

27. Etude des comportements hydrocarbures-air réagissant librement en sphérique
Essais CHARLES (phase 2) - Ordre d'essai . J. PERROT - CESTA/EX/ESP n°CI du 9 Avril 1982 .

28. Essais d'inflammation des mélanges C_2H_4-air - D.DESBORDES, J.C.LEYER, J.P.SAINT-CLOUD - ENSMA - Laboratoire d'Energétique et de Détonique - 21.4.82 - Contrat CCE/CEA-SRF/005/F(S)

29. Explosions aériennes de mélanges gazeux . Les essais AMEDE 2/8 et premiers résultats de synthèse . J.BROSSARD - Contrat n°79-046 EDF/SEPTEN - IRSTCO-ENSMA - Laboratoire d'Energétique et de Détonique - Juin 1982 - Note Technique EA n°79/15

30. Détonation dans l'air de mélanges gazeux air-hydrocarbure non confinés en géométrie sphérique - A.LANNOY - EDF/DER HP 219/82/32 .

31. Essais d'explosion de mélanges air-hydrocarbures . Essais AMEDE 2/7 et 2/8 Traitements SIDEX complémentaires sur capteurs de pression Kistler - J. PERROT _ CESTA/EX/ESP n° CI 51 du 12 Janvier 1983 .

8) DEGRE DE DISPONIBILITE

Les références marquées d'un astérisque sont diffusées par la Commission des Communautés Européennes .

Les références 1,2 et 16 sont disponibles; les autres références nécessitent un accord particulier du CEA et de l'EDF .

122-2-03	3 - 3
TITRE FORMATION ET DISPERSION ATMOSPHERIQUE DE NAPPES DERIVANTES DE GAZ OU D'AEROSOLS, EXPLOSIBLES OU TOXIQUES, SUITE A UN ACCIDENT SUR UNE INSTALLATION CHIMIQUE OU NUCLEAIRE.	*Pays* FRANCE *Organisme directeur* CEA /DSN
TITRE en anglais FORMATION AND ATMOSPHERIC DISPERSION OF DRIFTING CLOUDS OF EXPLOSIVE OR TOXIC GASES OR AEROSOLS AS A CONSEQUENCE OF AN ACCIDENT ON A CHEMICAL OR ON A NUCLEAR PLANT.	*Organisme éxécuteur* CEA/DSN/SAER FONTENAY-AUX-ROSES *Responsables* B.CRABOL/DSN/SAER
Date de démarrage 1980 — *Etat actuel* En cours *Date d'achèvement* 31/12/84 — *Dernière mise à jour* Mars 1983	*Scientifiques* A.BADR/DSN/SAER A.ROUX-LAGARDE/DSN/SAER

1) OBJECTIF GENERAL

Gaz ou aérosols toxiques : détermination en fonction du temps du champ de concentration dans l'air au niveau du sol et des dépôts sur le sol ou sur l'eau.

Gaz explosibles (hydrocarbures essentiellement) : détermination,en fonction du temps, des régions où le gaz 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 ET PLANNING

2.1 - Détermination du terme-source

- Cas des rejets accidentels à l'atmosphère d'installations nucléaires : aérosols sodés (feux de sodium), rejets d'UF_6 se transformant à l'air humide en UO_2F_2 et en acide fluorhydrique, ... (1984).

2.2 - Dispersion atmosphérique

Modélisation des phases initiales où le gaz ou l'aérosol ne sont pas des polluants minoritaires (1983).

- Cas des gaz lourds par rapport à l'air, 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 ... (1984).

Raccordement avec les modèles classiques de dispersion atmosphérique de polluant minoritaire (1983).

3) INSTALLATIONS EXPERIMENTALES

Les moyens expérimentaux comprennent :

- les moyens mis en oeuvre par le "Health and Safety Executive" (G.B.) pour la réalisation d'expériences de dispersion de gaz lourds sur le site de Thorney Island, auxquelles le DSN/SAER participe financièrement.
- Une veine hydraulique (30m de long, 3m de large, 1m de haut) servant de support aux études de simulation sur maquette.

4) ETAT DE L'ETUDE

Objectif particulier 2.2

- Implantation du code DENZ (UKAEA) sur le réseau CISI. Application au cas de l'épandage de chlore suite à la rupture d'un branchement sur un stockage.

- Interprétation des essais du HSE à Porton à l'aide du code DENZ et test sur les mêmes essais de différentes formulations de la modélisation de

l'entraînement de l'air (EIDSVIK, ZEMAN).
- Dépouillement des essais de Thorney Island réalisés en terrain plat (phase I).
- Participation à la préparation de la phase II des essais de Thorney Island : influence d'obstacles au sol sur le comportement du gaz lourd.
- Développement d'un code tridimensionnel pour les cas où le code DENZ n'est pas applicable (prise en compte de relief ou obstacles au sol).
- Etude expérimentale de la faisabilité de la simulation sur maquette en veine hydraulique de la dispersion de gaz lourds.
- Développement d'un modèle de jet de densité quelconque dans l'atmosphère avec prise en compte de la décroissance radioactive des produits de fission qui y sont contenus.

5) PROCHAINES ETAPES

Etude de la dispersion atmosphérique des gaz lourds ou aérosols en cas de rejet accidentel (par exemple : propane, chlore, ammoniac, aérosols sodés, ...).
- Analyse des documents concernés en provenance du Royaume Uni dans le cadre de l'accord d'échanges CEA-UKAEA signé le 18 septembre 1978 : bilan, synthèse.
- Expertise et comparaison des principaux codes de calcul utilisés pour la modélisation de gaz lourds.
- Interprétation des essais des phases I et II du programme HSE à Thorney Island (code DENZ et code tridimensionnel en cours de mise au point).
Amélioration de la modélisation, en tant que de besoin.
- Extension des possibilités d'application du code DENZ pour les besoins du DSN (en particulier au cas des rejets autres qu'instantanés).
- Etude expérimentale de la simulation sur maquette en veine hydraulique de la dispersion des gaz lourds : reproduction de certains essais réalisés à Thorney Island.

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 148.2.03

7) DOCUMENTS DE REFERENCE

/ 1 / Code DENZ - Manuel de l'utilisateur, Y.BERTHION, L.D. NUCHEZE.- Note DEMT/SYST/LECS/80/029.

/ 2 / Simulation sur maquette en veine hydraulique d'un rejet de gaz lourd en atmosphère neutre, B.CRABOL - Note SAER/81/283.

/ 3 / Contribution à l'étude des jets radioactifs et des jets lourds émis en présence d'un courant traversier. A. BADR - Thèse Docteur-Ingénieur - Décembre 1981.

/ 4 / Présentation de deux modèles d'évaluation de la dispersion des gaz lourds dans l'atmosphère - A.BADR, B.CRABOL - Note SAER/82/310.

/ 5 / Modélisation numérique de la dispersion des gaz lourds dans l'atmosphère (code DENZ) . Description et comparaison avec des rejets expérimentaux dans l'atmosphère à petite échelle - B.CRABOL, J.P.GRANIER,P.IFFENECKER, A.LAGARDE - Note Technique SAER/N° 83/345.

8) DEGRE DE DISPONIBILITE

Document disponible : /3/.
Autres documents internes, non disponibles sauf accord du CEA.

	CLASSIFICATION: 3.3
TITLE (ORIGINAL LANGUAGE): THEORETICAL STUDIES OF GAS CLOUD EXPLOSIONS MECHANICAL EFFECTS	COUNTRY: UK
	SPONSOR: SRD/HSE
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/UKAEA
	PROJECT LEADER: SF HALL
INITIATED: APRIL 1978 / COMPLETED:	SCIENTISTS:
STATUS: IN PROGRESS / LAST UPDATING: SEPTEMBER 1980	

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 ignition 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 be of value both in the prediction of hypothetical consequences and the analysis of previous incidents.

(iii) the effects of gas cloud shape and atmospheric conditions on explosion pressure waves propagation, since all present models are spherically-symmetric whereas real clouds will be asymmetric, expecially pancake shaped.

3. EXPERIMENTAL FACILITIES AND PROGRAMME

N/A

4. PROJECT STATUS

No results on (i) and (iii) to date. Preliminary results on (ii) obtained during EEC sponsored study and subsequently.

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, GASEX 1, GASEX 2.

7. REFERENCE DOCUMENTS

Nuclex 78 Paper summarises results to date on items (ii) and (iii). Also SRD R 104, SRD R 155, SRD R 126.

4. POWER TRANSIENTS

	CLASSIFICATION: 4
TITLE (ORIGINAL LANGUAGE): Comparative analysis between the behaviour of LWR uranium fuelled cores and LWR Pu-mixed fuelled cores in the accident of rod drop (BWR core) and of rod ejection (PWR core)	COUNTRY: Italy SPONSOR: CEE-ENEA-ENEL
TITLE (ENGLISH LANGUAGE):	ORGANISATION: ENEL PROJECT LEADER: F. DI PASQUANTONIO
INITIATED: January 1979 / COMPLETED: May 1980	SCIENTISTS: E. Brega
STATUS: Completed / LAST UPDATING: July 1980	

The purpose of this work was to compare the behaviour of LWR uranium fuelled and Pu mixed fuelled cores in the accident of rod drop (BWR core) and rod ejection (PWR core). The analysis has been carried out by means of the code SYNTH-C.

In the PWR case the rod ejection accident has been considered at hot core, zero power initial condition. For mesh spacing we took variable Δx and Δy between 6.737 cm and 0.654 cm and constant mesh spacing Δz = 16.6 cm for a total number of 85 x 85 x 21 = 151,725 mesh points. The calculations have been carried out with three energy groups (fast, epithermal and thermal). The different material compositions were 468 in the uranium core and 612 in the Pu mixed core, while the thermo-hydraulic channels were 112 in both cases.

For each material region the radial conduction has been discretized with 5 points in the fuel and one in the cladding.

The results point out the good behaviour of both uranium fuelled and Pu-mixed cores. In fact the peak temperature and enthalpy are well below the threshold operating limits and there are very large safety margins.

In the BWR case the rod drop accident has been considered at cold core, zero power condition (start up accident). For mesh spacing we took $\Delta x = \Delta y$ = 7.62 cm, Δz = = 7.73 cm for a total number of 57 x 57 x 61 = 198,189 mesh points. Also in this case all calculations has been carried out with three energy groups. The different material composition were 79 both for uranium a Pu mixed cores.

The thermal feedback has been calculated with a temperature model (like the one of the WIGL code).

The results show that also in these cases the peak of temperatures and enthalpy are well below the threshold operating limits.

(ENEL-CRTN, Bastioni Porta Volta, 10, I-20121 Milano)

45.897

	CLASSIFICATION : 4 / 6 / 1.1.2
TITLE (ORIGINAL LANGUAGE) : TRANSITORI TERMOIDRAULICI DI BWR PER MALFUNZIONAMENTI DEL SISTEMA DICONTROLLO	COUNTRY : ITALY SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : THERMOHYDRAULIC TRANSIENTS OF BWR'S FOR SYSTEM CONTROL FAULTS	ORGANISATION : CISE PROJECT LEADER : G. BARZONI
INITIATED : 1980 COMPLETED : 1982 STATUS : IN PROGRESS LAST UPDATING : APRIL 1982	SCIENTISTS : C. MEDICH C. SANDRI

1. General aim: to study the thermohydraulic behaviour of a BWR power channel during the so called "anticipated transients without scram".

2. Particular objectives: to measure and analize pressure drops, mixture densities, heat transfer coefficients, and critical powers in a simulated power channel during transients of power, inlet flowrate and pressure.

3. Experimental facilities:

   Test section geometry: tubular, ID = 12 mm. Heated section L=3.8 m; adiabatic section L = 4 m.

   Experimental loop : IETI-1 located in the "EMILIA" power station in Piacenza, suitably adapted .

4. Project status: experimental data available .

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
THERMOHYDRAULIC TRANSIENTS OF BWR'S FOR SYSTEM CONTROL FAULTS	4 / 6 / 1.1.2

5. Next steps: experimental data analysis by means of the RATT-1 code.

6. Relation to other projects and codes: none.

7. Reference documents
   - G. Barzoni, R. Granzini: "Proposta di programma sperimentale sulla termoidraulica dei transitori operazionali" CISE-N.T. 80.078
   - G. Barzoni, R. Martini: "Programma sperimentale sulla termoidraulica dei transitori operazionali (ATWS): specifiche per l'esecuzione della campagna sperimentale" Promemoria CISE 81/077
   - A. Azzolin, G. Barzoni: "Misure ed analisi del coefficiente di scambio termico in transitori operazionali dei reattori BWR" CISE-N.T. 81.188.

8. Degree of availability: to a limited extent.

9. Additional information:

   Budget about 90 millions Lit; personnel involved: 1,3 men year.

(CISE, P.O. Box 12081, I-20100 MILANO)

4.1. REACTIVITY INSERTIONS

	CLASSIFICATION : 4.1
TITLE (ORIGINAL LANGUAGE) : ANDYCAP: Tre dimensional dynamisk model af kogendevands reaktor kerne	COUNTRY : Denmark SPONSOR : Risø National Laboratory
TITLE (ENGLISH LANGUAGE) : ANDYCAP: 3-D-dynamical model of a BWR-core.	ORGANISATION : Risø National Laboratory PROJECT LEADER : A.M. Larsen
INITIATED : 1969 / COMPLETED : 1972 STATUS : in use / LAST UPDATING : 1977	SCIENTISTS : A.M. Larsen

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

DYN-4-78 (Internal Report)

8. Degree of availability

Not available.

	CLASSIFICATION: 4.1
TITLE (ORIGINAL LANGUAGE):	COUNTRY: Denmark
	SPONSOR: Risø National Laboratory
TITLE (ENGLISH LANGUAGE): NORHAV - Three-Dimensional Transient Calculation Program for the PWR Core (ANTI)	ORGANISATION: Risø National Laboratory
	PROJECT LEADER: Anne Margrethe Larsen
INITIATED: 1977 / COMPLETED:	SCIENTISTS: Anne Margrethe Larsen
STATUS: in progress / LAST UPDATING:	

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 program is running (in the debugging phase).

5. Next steps

Documentation. Improvements. Testing.

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   Risø-M-2209.

8. Degree of availability

Available on exchange basis when completed.

	CLASSIFICATION: 4.2
TITLE (ORIGINAL LANGUAGE): Udarbejdelse af en dynamikmodel for Barsebäck 2 kraftværket	COUNTRY: DENMARK SPONSOR: Risø National Laboratory
TITLE (ENGLISH LANGUAGE): Development of a BWR-power plant dynamic model for the Barsebäck 2 unit	ORGANISATION: Risø National Laboratory PROJECT LEADER: P. la Cour Christensen
INITIATED: 1977 — COMPLETED: 1980	SCIENTISTS: P. la Cour Christensen
STATUS: In use, being improved — LAST UPDATING:	

1. General aim

The goal of the project is to develop a general toal for calculation of transients in a BWR power plant of the ASEA-ATOM type. The initiating events may be any kind of disturbance which still maintain the normal reactor cooling mechanism.

2. Particular objectives

The model includes the reactor with recirculation pumps, the steam line, the turbine with one high and three low pressure sections and a reheater. The model of the feedwater system contains all five feedwater heaters and the pumps. The three main control systems are included, they are: Power control by recirculation. steam pressure control including the dump system and the feedwater control. The model of the reactor core and the steam line is one-dimensional, while other components are described by lumped parameters.

3. Experimental facilities

4. Project status

1. Progress to date: The model is finished and the program has been tested with calculation of several transients. Verification has been initiated.
2. Essential results.

5. Next steps

6. Relation with other projects

7. Reference documents

Risø-M report no. 2190.

8. Degree of availability

Not available.

	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 1980	

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

4.2. SECONDARY SYSTEM EFFECTS

	CLASSIFICATION : 4.2
TITLE (ORIGINAL LANGUAGE) : BWR-stations dynamik model	COUNTRY : Denmark
	SPONSOR : Risø National Laboratory
TITLE (ENGLISH LANGUAGE) : Development of a Dynamic Model of a BWR Nuclear Power Plant.	ORGANISATION : Risø National Laboratory
	PROJECT LEADER : E. Nonbøl
INITIATED : 1973 / COMPLETED : 1976	SCIENTISTS : E. Nonbøl
STATUS : in use / LAST UPDATING : currently	

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 and the multigroup kinetics equation is solved using the improved quasi static method combined with finite element technique. 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.
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

The thesis is fully available.

	CLASSIFICATION : 4.2,
TITLE (ORIGINAL LANGUAGE) : Udarbejdelse af en dynamikmodel for Ringhals 3 kraftværket	COUNTRY : DENMARK SPONSOR : Risø National Laboratory
TITLE (ENGLISH LANGUAGE) : Development of a PWR-power plant dynamic model for the Ringhals 3 unit.	ORGANISATION : Risø National Laboratory PROJECT LEADER : P. la Cour Christensen
INITIATED : 1980 / COMPLETED : 1982 STATUS : In use, being improved / LAST UPDATING :	SCIENTISTS : P. la Cour Christensen

1. General aim

The goal of the project is to develop a general tool for calculation of transients in a PWR power plant. The initiating events may be any kind of disturbance which still maintain the forced convection cooling of the reactor without two phase flow in the primary circuit.

2. Particular objectives

The model includes the reactor with one primary loop, one steam generator, one pump and pressurizer. The models of the core and the steam generator are one-dimensional, the primary tubes are described by delay time simulation, while the other parts are described by lumped parameters. The steam load model consist of the steam line, a turbine with a high and a low pressure section and a reheater. The feedwater system is modelled with all 6 feedwater heaters. Models for the main control loops are incorporated; they are: Reactor power control, primary pressure and volume control, steam generator water level control and steam load control with dump facility.

3. Experimental facilities

4. Project status

1. Progress to date: The model is finished, and the program has been tested with calculation of several transients.

2. Essential results

5. Next steps

The model will be improved as more detailed data for the steam and feed water circuits have been obtained.

6. Relation with other projects

7. reference documents

8. Degree of availability

	CLASSIFICATION : 1.1.2. 4.2.
TITLE (ORIGINAL LANGUAGE) : TERMOIDRAULICA DEI GENERATORI DI VAPORE IN CONDIZIONI INCIDENTALI	COUNTRY : ITALY
	SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : THERMOHYDRAULICS OF STEAM GENERATORS UNDER ACCIDENT CONDITIONS	ORGANISATION : CISE
	PROJECT LEADER : L. MAZZOCCHI
INITIATED : 1980 / COMPLETED : TO BE DEFINED	SCIENTISTS : G. CATTADORI G. MASINI C. MEDICH
STATUS : IN PROGRESS / LAST UPDATING : APRIL 1982	

1. General aim: to study the thermohydraulics of the secondary (and possibly of the primary) side of a steam generator during various accidental and operational transients both in the primary and secondary circuit.

2. Particular objectives: to measure pressure drops, mass holdup, void fraction, heat transfer coefficients and critical powers in the secondary fluid both in steady state and transient conditions; to verify existing prediction methods or to develop new ones.

3. Experimental facilities, programs: experimental facility is obtained by modifying and connecting two existing CISE loops: the IETI-4 loop (for the primary circuit) and the CIRCE loop (for the second. circuit). The experimental program includes:
   - steady state tests (both in adiabatic and diabatic conditions) with different values of secondary pressure, mass flow rate and inlet quality;

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<table>
<tr><td colspan="2"></td><td>CLASSIFICATION :
4.2</td></tr>
<tr><td colspan="2" rowspan="2">TITLE (ORIGINAL LANGUAGE) :
STUDIO DEL COMPORTAMENTO DINAMICO DI UN GENERATORE DI VAPORE A CIRCOLAZIONE NATURALE</td><td>COUNTRY :
ITALY</td></tr>
<tr><td>SPONSOR :
ENEA and
ANSALDO DBGV</td></tr>
<tr><td colspan="2" rowspan="2">TITLE (ENGLISH LANGUAGE) :
INVESTIGATION OF DYNAMIC BEHAVIOUR OF A NATURAL CIRCULATION STEAM GENERATOR</td><td>ORGANISATION :
ENEA and
ANSALDO DBGV</td></tr>
<tr><td>PROJECT LEADER :
M. SALA - ANSALDO DBGV
F. FABRIZI - ENEA</td></tr>
<tr><td>INITIATED :
1979</td><td>COMPLETED :</td><td rowspan="2">SCIENTISTS :
F. FABRIZI - ENEA
G. QUADRELLI-ANSALDO DBGV</td></tr>
<tr><td>STATUS :
IN PROGRESS</td><td>LAST UPDATING :
APRIL 1982</td></tr>
</table>

1) GENERAL AIM :

Investigation of a natural circulation steam generator performance under operational transients.

2) PARTICULAR OBJECTIVES :

A collection of experimental data for : i) optimization of physical models and ii) validation of computer codes describing the dynamic behaviour of a steam generator.

3) EXPERIMENTAL FACILITIES AND PROGRAMME :

CFA — A Freon-water loop operating at about 10 bar. The test section, a 200 KW model steam generator, consists of a 15 U-tube bundle having the same tube array, pitch and diameter as a 900 MW prototype. The tube bundle height has been scaled down by a factor 0.4 approximately.

GEST-GEN — A steam-water loop operating at about 70 bar. The test section, a 20 MW steam generator, consists of 98 full length U-tubes and represents a significant subassembly of a prototypical tube bundle. The working conditions are the same as in a 900 MW prototype.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
INVESTIGATION OF DYNAMIC BEHAVIOUR OF A NATURAL CIRCULATION STEAM GENERATOR	4.2

4) PROJECT STATUS :

1. Progress to date

CFA Loop : The FREGENE test section and the loop are under construction. The experimental test matrix has been completely defined.

GEST-GEN Loop : Loop and test section are under construction.

A dynamic computer code has been developed.

5) NEXT STEPS :

The CFA-FREGENE loop will be operative by the end of 1982.
The GEST-GEN loop will be operative by the end of 1983. The test matrix will be written in 1982.

7) REFERENCE DOCUMENTS :

a) G. Quadrelli, M. Sala "Modello "full lenght" di un generatore di vapore a circo lazione naturale senza preriscaldatore per il circuito GEST", Ansaldo DBGV int. rep. STI-022-CN-1980.
b) G. Quadrelli, B. Besozzi, M. Sala "Studio del regime transitorio e della instabilità oscillatoria di un GV
II - Dimensionamento di un modello di generatore acqua-freon che simula il GV Westinghouse senza preriscaldatore"
Ansaldo DBGV int. rep. STI-029-CN
c) V. Pietrelli "Progetto meccanico della Sezione di Prova FREGENE per esperienze sulla dinamica del GV" ENEA int. rep. (81)6/VBD-3/AA23-1

8) DEGREE OF AVAILABILITY :

Availability of technical reports is subjected to the authorization of the sponsoring organizations.

(Ing. R. De Santis - Direttore tecnico contratto ENEA - ANSALDO DBGV - c/o ANSALDO DBGV Viale Sarca, 336 - Milano)

9) ADDITIONAL INFORMATION :

Personnel involved: 2 researchers
Budget 1982: about 100 millions Lit (projects 4.2, 4.3)

4.3. INSTABILITY

	CLASSIFICATION : 4.3
TITLE (ORIGINAL LANGUAGE) : STUDIO DELLA SOGLIA DI INSTABILITA' DI UN GENERATORE DI VAPORE A CIRCOLAZIONE NATURALE	COUNTRY : ITALY SPONSOR : ENEA and ANSALDO DBGV
TITLE (ENGLISH LANGUAGE) : INVESTIGATION OF INSTABILITY THRESHOLD OF A NATURAL CIRCULATION STEAM GENERATOR	ORGANISATION : ENEA and ANSALDO DBGV PROJECT LEADER : M. SALA - ANSALDO DBGV F. FABRIZI - ENEA
INITIATED : 1979 · COMPLETED :	SCIENTISTS : F. FABRIZI - ENEA G. QUADRELLI - ANSALDO DBGV
STATUS : IN PROGRESS · LAST UPDATING : APRIL 1982	

1) GENERAL AIM :

Investigation of the thermohydrodynamic parameters characterizing the onset of instability in a natural circulation steam generator.

2) PARTICULAR OBJECTIVES :

A collection of experimental data obtained by a model steam generator which allows: a) the optimizazion of physical models, b) the validation of computer codes describing the dynamic behaviour and c) the stability ranges of a natural circulation steam generator.

3) EXPERIMENTAL FACILITIES AND PROGRAMME :

The experiments will be performed on the Freon-water loop CFA.
The test section, a 600 KW steam generator, consists of a 15 U-tube bundle having the same tube array, pitch and diameter as a 900 MW prototype. The tube bundle height has been scaled down by a factor 0.4 approximately.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
INVESTIGATION OF INSTABILITY THRESHOLD OF A NATURAL CIRCULATION STEAM GENERATOR	4.3

4) PROJECT STATUS :

1. Progress to date

The FREGENE test section and the loop are under construction. The experimental test matrix has been completely defined.

5) NEXT STEPS :

The experimental tests will start at the beginning of 1983.

7) REFERENCE DOCUMENTS :

a) G. Quadrelli - B. Besozzi - M. Sala "Studio del regime transitorio e della instabilità oscillatoria di un GV.
II - Dimensionamento di un modello di generatore acqua-freon che simula il GV Westinghouse senza preriscaldatore"
Ansaldo DBGV int. rep. STI-029-CN

b) V. Pietrelli "Progetto meccanico della Sezione di Prova FREGENE per esperienze sulla dinamica del GV"
ENEA int. Rep. (81)6/VBD-3/AA23-1

8) DEGREE OF AVAILABILITY :

Availability of technical reports is subjected to the authorization of the sponsoring organizations.

(Ing. R. De Santis - Direttore tecnico contratto ENEA - ANSALDO DBGV - c/o ANSALDO DBGV - Viale Sarca, 336 - MILANO).

9) ADDITIONAL INFORMATION :

Personnel involved: 2 researchers
Budget 1982: about 100 millions Lit (projects 4.2, 4.3)

	CLASSIFICATION : 4.3
TITLE (ORIGINAL LANGUAGE) : Studio sperimentale sulla instabilità di flussi bifasi in canali paralleli con differenti profili di flussi termici.	COUNTRY : ITALY SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : An Experimental Study on Two-Phase Flow Instability in Parallel Channels with Different Heat Flux Profile.	ORGANISATION : ENEA PROJECT LEADER : G. Palazzi

INITIATED : 1980	COMPLETED : September 1981	SCIENTISTS : L. Rinaldi
STATUS : Completed	LAST UPDATING :	

1. General Aim

The main purpose of this experimental work is to determine the parallel-channel instability threshold in the actual geometry of nuclear plant steam generator.

2. Particular Objectives

Investigation in:
a) influence of different axial heat flux profiles in two parallel channels for density wave instability phenomena.
b) Oscillation period behaviour in particular conditions of the system.

3. Experimental Facility

The experimental data have been obtained in the CUA-2 loop installed in the CSN Casaccia (Rome) . Maximum power 4 MW, maximum pressure 250 bar.

4. Project Status and Essential Results

a) completed
b) 116 tests have been performed to examine the flow oscillation nature in two round tubes, 9 m length and 2.12 internal diameter. The work conditions were pressure 3.5 and 5.0 MPa, specific mass flow-rate 300÷800 Kg/m^2s, subcooling until 500 KJ/Kg. Instability threshold was obtained with various combinations of the parameters and in two different ways, increasing power or decreasing flow-rate.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
An Experimental Study on Two-Phase Flow Instability in Parallel Channels with Different Heat Flux Prof.	4.3

The oscillations are typical of the density wave instability and their periods are nearly related to the transit time of the fluid through the heating zone.
The influence of the principal parameters of the system has been studied for the instability phenomenon and for the occurring of the second mode oscillations.
The experimental results of the threshold instability have been compared with the LOOP 1 code results.

5. Reference Documents

L. Rinaldi "Analisi qualitativa e metodi di scaling della instabilità termoidraulica dei deflussi bifase" CNEN, TERM-RIS, (80)5/VBD-1/G71A-4.

M. Cumo, G. Palazzi, L. Rinaldi "An Experimental Study on Two-Phase Instability in Parallel Channels with Different Heat Flux Profile", CNEN-RT/ING(81)1.

6. Degree of Availability

Free.
Contact Person: G. Palazzi - TERM-RISIL, ENEA, CSN Casaccia, CP 2400, I-00100 Roma

7. Additional Information

Staff: 3.5 man·year

Budget: ≃ 70 ML

5. BEHAVIOUR, TRANSPORT AND RELEASE OF RADIOACTIVE SUBSTANCES

148.1.10		5
TITRE Participation au programme MARVIKEN V		Pays FRANCE
		Organisme directeur CEA/DSN EDF
TITRE en anglais Participation to the MARVIKEN V project		Organisme éxécuteur STUDSVIK (Suède) CEA/DEMT/SYST Saclay
		Responsables J.FERMANDJIAN DSN/SAER
Date de démarrage 1/1/1983	Etat actuel En cours	Scientifiques G.LHIAUBET DEMT/SYST
Date d'achèvement 31/12/1985	Dernière mise à jour Janvier 1983	

1/ OBJECTIF GENERAL

-Acquisition de données expérimentales sur la rétention des produits de fission dans le circuit primaire des réacteurs à eau légère, pour les scénarios d'accidents dominant le risque.

-Démonstration à grande échelle du comportement des aérosols dans le circuit primaire.

2/ OBJECTIFS PARTICULIERS

Année 1983 - Essais "fissium" : simulation des produits de fission semi-volatils (CsI,CsOH,Te) émis durant la dégradation du coeur.
Année 1984-Essais "fissium+corium" : simulation des produits de fission semi-volatils précédents et des produits moins volatils (Fe,Cr,Zr,Mn,Cd,Ag) émis durant la fusion du coeur.
Année 1985 - Rédaction du rapport expérimental final.
- Validation expérimentale du code AEROSOLS.

3/ INSTALLATIONS EXPERIMENTALES

L'installation MARVIKEN (Suède) comporte en série trois capacités qui simulent respectivement une cuve (160 m^3) , un pressuriseur (50 m^3) et un réservoir de décharge du pressuriseur (50 m^3).

4/ ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

. Essais préliminaires effectués à ce jour.
- Calculs prévisionnels (séquence TMLB') effectués avec le code AEROSOLS/A2 : relâchement au niveau du réservoir de décharge du circuit PWR voisin de celui calculé dans le circuit MARVIKEN, mais dû à des phénomènes de rétention totalement différents.

5/ PROCHAINES ETAPES

. 3 essais "Fissium" en 1983
- → Simulation coeur dégradé (atmosphère sèche)
- → Simulation coeur dégradé (atmosphère condensante)
- → Simulation gros LOCA

. 3 essais "Fissium+Corium" en 1984
- → Simulation TMLB' (condensation dans le réservoir de décharge
- → Simulation gros LOCA
- → Simulation petit LOCA

. Interprétation des essais avec le code AEROSOLS/A2 modifié (prise en compte des dépôts dans les tuyauteries). Comparaison des codes TRAP-MELT (BCL/USA) et AEROSOLS/A2.

6/ RELATIONS AVEC D'AUTRES ETUDES

- Code de calcul JERICHO. Modélisation du comportement des produits de fission dans les accidents des réacteurs à eau - Fiche 148.1.09.
- Programme PITEAS - Etudes des transferts de radioactivité à l'intérieur de la centrale Filtration rustique Fiche 148.1.02.
- Thermohydraulique d'un coeur dégradé Fiche 145.2.05.
- Comportement d'un crayon combustible PWR dans le cas d'une perte de réfrigérant du type petite brèche. FLASH CSD (combustible sévèrement dégradé) Fiche 148.1.07.

7/ DOCUMENTS DE REFERENCE

1- J.COLLEN, D.MECHAM
A study report on the MARVIKEN full scale aerosol transport tests project MX5-4 April 1982.

2- G.LHIAUBET
Rétention comparée des aérosols dans le circuit d'un réacteur PWR et dans le circuit expérimental MARVIKEN
Note technique DEMT/SYST n°82/047 Août 82.

3- C.HERVOUET
Présentation du code de calcul TRAP-MELT
Note technique DSN/SAER n°82/326 Novembre 82.

8/ DEGRE DE DISPONIBILITE

Document n°1 distribué par le projet MARVIKEN.
Documents n°2 et 3 internes, non disponibles, sauf accord du CEA.

	CLASSIFICATION : 5.
TITLE (ORIGINAL LANGUAGE) : Översikt av arbetsområdet ANALYS AV SVÅRA HAVERIER i svenska kärnkraftverk	COUNTRY : Sweden
	SPONSOR : Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE) : Survey of the field "Analysis of Severe Accidents in Swedish Nuclear Power Plants"	ORGANISATION : STUDSVIK
	PROJECT LEADER : Kjell Johansson
INITIATED : 1982 Nov 9 / COMPLETED :	SCIENTISTS :
STATUS : Completed / LAST UPDATING : June 1983	

OBJECTIVE: To produce a programme of activities to survey the Swedish nuclear power plants concerning the consequencies if a severe reactor accident should occur and to study what measures should be done to mitigate the consequences of such an accident.

PROJECT STATUS: The project is completed and has resulted in the RAMA project (Reactor Accident Mitigation Analysis)

	CLASSIFICATION : 5.
TITLE (ORIGINAL LANGUAGE) : Underlag för beräkning av aerosoltransport före och efter planerade experiment i MARVIKEN	COUNTRY : SWEDEN SPONSOR : Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE) : Compilation of data as basis for calculation of aerosol transport before and after planned experiments at MARVIKEN	ORGANISATION : STUDSVIK PROJECT LEADER : Kjell Johansson
INITIATED : Dec 1982 / COMPLETED : March 1985 STATUS : In progress / LAST UPDATING : June 1983	SCIENTISTS :

GENERAL AIM: NRC (Nuclear Regulatory Commission) has commissioned BCL (Batelle Columbus Laboratories) to develop the TRAP MELT code and make pre- and post-test calculations of the aerosol transport at the Marviken tests. Data as basis for these calculations are to be supplied by Studsvik.

APPROACH: The Studsvik project comprises
- before each test compilation of predicted data of system specification, thermo-hydraulic conditions and specification of added amounts of fissium and corium before each test for pre-test calculations
- after each test compilation of real data of system specification, etc during the test for post-test calculations
- after each test compilation of results of the experiments, especially fraction of the added amount of fissium and corium found as deposited, penetrated and gas-borne material.

PROJECT STATUS: Data for pre- and post-test calculations for the two experiments hitherto performed at Marviken (of which one shake-down test) have been compiled and delivered to NRC.

	CLASSIFICATION : 5.
TITLE (ORIGINAL LANGUAGE) : Beräkningar av transport av jod och cesium vid de första eperimenten med fissium i Marviken	COUNTRY : SWEDEN SPONSOR : Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE) : Calculations of transport of iodine and cesium at the first experiments with fissium at Marviken	ORGANISATION : STUDSVIK PROJECT LEADER : Kjell Johansson
INITIATED : 1982 Dec 15 COMPLETED : April 1983 STATUS : Completed LAST UPDATING : June 1983	SCIENTISTS :

OBJECTIVE: Verifying the TRAP MELT 1 and HARM-S codes by composing results of calculations performed by the codes for the Marviken Aerosol Transport Tests with found results of the fraction of the injected material deposited outo the system surfaces in the tests.

PROJECT STATUS: The project is completed. Pre- and post-test calculations for shake-down test B and pre-test-calculations for Test 1 have been performed.

REFERENCE DOCUMENT: H. Häggblom
Calculations of aerosol retention for the first two Marviken-V experiments. (Restricted distribution)

8369

	CLASSIFICATION : 5.
TITLE (ORIGINAL LANGUAGE) : Projekt RAMA Anpassning av MAAP till tidiga svenska BWR	COUNTRY : SWEDEN SPONSOR : Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE) : Project RAMA Adjusting the MAAP code for application to early Swedish BWRs	ORGANISATION : STUDSVIK PROJECT LEADER : Kjell Johansson
INITIATED : 1983 April 1 — COMPLETED : Sept 1983 STATUS : In progress — LAST UPDATING : June 1983	SCIENTISTS :

OBJECTIVE: To prepare and sign a contract with Fauske & Associates INC (FAI) on adjusting the MAAP code to the nuclear power units Oskarshamn 2 and Ringhals 1. The modelling work is to be done by FAI with K. Johansson, STUDSVIK, as project leader.

PROJECT STATUS: The contract has been signed and the modelling work is in progress.

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CLASSIFICATION :
5.

TITLE (ORIGINAL LANGUAGE) :
Forskningsprojektet RAMA

COUNTRY :
SWEDEN

SPONSOR :

TITLE (ENGLISH LANGUAGE) :
THE RAMA RESEARCH PROJECT
RAMA = Reactor Accident Mitigation Analysis

ORGANISATION :

PROJECT LEADER :
Kjell Johansson

INITIATED :
1983, Febr 1

COMPLETED :
1985, January

SCIENTISTS :

STATUS :
In progress

LAST UPDATING :
1983, June

RAMA
Steering group
Kjell Johansson

THE RAMA RESEARCH PROJECT

RAMA = Reactor Accident Mitigation Analysis

Aims

The RAMA research project was initiated as a result of the policy declaration by the the Swedish authorities that measures if any should be be taken before 1989 at the Swedish nuclear power stations so as to provide additonal protection in the case of very severe accidents. The power station owners shall suggest effective measures in this respect for each individual plant, based on plant specific studies. RAMA is a joint project between the Swedish nuclear inspectorate and national institute of radiation protection and the nuclear power companies.The purpose of the RAMA project is to carry out R&D work of a generic nature so as to support the plant specific work, and to provide an improved basis for the authorities future scrutiny of proposed measures.

The technical areas which are of interest for the project are: the liberation, transport and retention of radioactive substances following core accidents, and the effects on the containment during accident sequences which are of interest for Swedish plants.

Purpose

The purpose of the project is to collect and digest results of relevant R&D work, so that these can be used for evaluations and assessments in the Swedish decision process.

Organisation

The project will be led by a steering group with representatives from the financing parties. SKI will also appoint a chairman for the steering group. The direct day-to-day management will be carried out by a project leader, who will be a member of the steering group and will report back to the group. In addition two working groups will be formed, one for questions concerning the liberation, transport and retention of radioactive substances and the other for analysis of the accidents impact on the containment. Each working group shall produce a detailed programme for its activities, and the programme shall be submitted to the steering group for approval. The aims of the group activities are:

working group, source terms:

> to develop methods and computer programmes for the calculation of liberation, transport and retention of radioactive substances in the primary systems and containment with associated systems, for application in nuclear power stations in Sweden. The work will be based on HAARM, RETAIN, NAUA and other computer programmes which are or can be made available to the project. Verification of methods and programmes, with utilization of available information and data from relevant experiments

working group, containment function:

> to develop methods and computer programmes for the calculation of core meltdown sequences and the impact on the containment during severe accidents, for application in nuclear

power stations in Sweden. The work will in the first phase proceed by utilizing the MAAP programme, developed within IDCOR. Useable versions for relevant Swedish reactor types will be developed, and test and verification calculations will be carried out.

Schedule

The work in the RAMA project will be carried out so that a final report can be issued in January of 1985. However, the development work in the two groups shall be carried out so that methods and computer programmes are available in time for the plant specific studies, which the power companies intend to carry out according to a separate plan, called MITRA = Mitigation of Reactor Accidents.

Costs

The total costs for the two years' work according to the budget proposal below amounts to SEK 10 millions.

Financing

It is proposed that the costs should be allocated as follows:

Nuclear Inspectorate	SKI	50%
Swedish State Power Board	SV	30%
Oskarshamn Power Group	OKG	15%
Swedish National Institute of Radiation Protection	SSI	5%
	Σ	= 100%

Budget

Project management	SEK 1 000 000
Working group for containment function	2 000 000
Working group for source terms	2 000 000
Membership IDCOR	2 000 000
For use by the steering group	3 000 000
Total	SEK 10 000 000

Steering group

Arne Hedgran	(chairman) KTH Royal Institute of Technology
Christian Gräslund	SKI Swedish Nuclear Power Inspectorate
Bo Liwång	"-
B Åke Persson	SSI Institute of Swedish Radiation Protection
Per-Eric Ahlström	SV Swedish State Power Board
Ralf Espefält	"-
Emil Bachofner	OKG AB
Johan Engström	"-
Ingmar Tirén	AA AB ASEA-ATOM
Kjell Johansson	(project leader) SE Studsvik Energiteknik AB
Anita Karlsson	(adm) "-

Working group:
Containment Function

KTH	Kurt Becker
SV	Ralf Espefält Wiktor Frid
OKG	Johan Engström
AA	Kjell Elisson
SKI	Per Bystedt
SE	Roland Blomquist

Working group:
Source Terms

CTH	Jan-Olof Liljenzin
SV	Gustaf Löwenhielm
OKG	Bertil Persson
SKI	Gunilla Bergström
SSI	Christer Wiktorsson
SE	Hans Häggblom

	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: SEPTEMBER 1980	

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 st dies 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

5.1. RELEASE FROM FUEL-ELEMENTS IN NORMAL OPERATIONS

Berichtszeitraum/Period 01.01.82 - 31.12.82	Klassifikation/Classification 5.1	Kennzeichen/Project Number 150 492
Vorhaben/Project Title Jod-Exhalation aus UO_2 unter stationären und transienten Bedingungen Iodine Release from UO_2 under Steady-State and Transient Conditions		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor KRAFTWERK UNION AG Reaktortechnik B 222, Erlangen
Arbeitsbeginn/Initiated 01.11.80	Arbeitsende/Completed 30.06.82	Leiter des Vorhabens/Project Leader G. Kaspar
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating 30.06.82	Bewilligte Mittel/Funds

1. General Aim

In nuclear fission, approximately ten times more cesium is produced than iodine. As the combination of cesium and iodine forms a more stable iodide than comparable zirconium iodides, analysis of the effect of iodine on zircaloy cladding should always be coupled with analysis of the concomitant quantity of cesium.

When the iodine and cesium release measurements are available, it will be possible to specify, from the iodine and cesium inventory resulting from burn-up, how the iodine which diminishes cladding tube ductility will affect fuel rod behaviour under LOCA conditions, taking transient discharge into account.

2. Particular Objectives

The fission iodine which is embedded in the lattice of the UO_2 or settles in pores and at grain boundaries after high burn-ups can only leave the UO_2 by diffusion to an open surface (inner or outer surface of a pellet or a surface of fracture). The same applies to the equally important cesium. The aim of the envisaged examinations is to investigate iodine and cesium escape from UO_2 in the following cases:

- release of iodine/cesium under steady-state conditions (isochronous, isothermal heat-up)
- release of iodine/cesium under transient temperature conditions (transient heat-up)

The results from these tests, togehter with the results of the examinations of LOCA bursting behaviour carried out on unirradiated cladding tubes at PNS/KFK and at KWU, the results obtained at PNS/KFK for cladding tubes doped with various quanitites of iodine and the in-pile tests which are also to be performed as a part onf PNS 4237 will then provide enough information to describe the behaviour of the cladding tubes of fuel rods in the event of a loss-of-coolant accident.

3. Research Program

Analysis of iodine and cesium release from UO_2 samples taken from actual spent LWR fuel. Investigation of the way in which release is influenced by time, temperature, burn-up and the radial position of the sample within the pellet.

4. Experimental Facilities

As part of RS 285, a test facility was set up for this project to analyse iodine/cesium release from spent fuel.

5. Progress to Date

Six series of tests have been carried out. In five cases the temperature of the sample was set at 1700 °C, in the other case it was set at 1300 °C. In all but one test the samples used had a low-level burn-up (10 GWd/tU), the remaining test was performed with a sample with a burn-up of 33 GWd/tU. In all the tests at 1700 °C, the collector was retained above the evaporator for a uniform 30 seconds per capture plate. In the one test at 1300 °C, the 22 capture plates were moved above the evaporator at a constant rate for a release time of approximately 2.3 minutes to enable comparisons to be made with tests previously performed on high burn-up samples. Evaluation was in all cases undertaken in respect of cesium only. The rest of the inventory was determined from the start by the different burn-up of each sample. The investigations which have been performed have been fully evaluated.

6. Results

The tests on low burn-up samples at 1700 °C clearly show that the average released volume of cesium is more than an order of magnitude lower than the values for tests on high burn-up samples. If the released volume is plotted as a function of time, it is clear that samples with low burn-up behave differently to samples with high burn-up. The time curve for all high burn-up tests evaluated to date demonstrated that the first collectors exhibited very high release values and that the values decreased constantly as a function of time. Evaluation of the low burn-up tests indicates that release over the period tested has a more uniform characteristic. The same also holds true for the low burn-up test at 1300 °C.

The experiments show that the fission elements iodine and cesium in the UO_2 lattice have the character of a noble gas. The kinetic behaviour of iodine/cesium release is marked by a release rate which is relatively higher immediately after the temperature transient but steadily drops off with the passage of time and reaches a "steady-state" value after about 1 minute. The initial high release originates both from those surfaces which already exist and from those which have been newly formed during the heat-up phase as a result of intergranular cleavage. The "steady-state" release rate is determined by diffusion of the iodine or cesium in the UO_2.

7. Next Steps

Work has been completed. The final report will be compiled.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

-

140-1-02	5 - 1
TITRE EVALUATION DU TAUX DE RELACHEMENT DES PRODUITS DE FISSION HORS D'UNE RUPTURE DE GAINE DANS UN REACTEUR A EAU SOUS PRESSION - FONCTIONNEMENT NORMAL.	*Pays* FRANCE *Organisme directeur* CEA/IPSN EDF
TITRE en anglais ASSESSMENT OF THE FISSION PRODUCT RELEASE RATE FROM A CLADDING DEFECT IN A PRESSURIZED WATER REACTOR UNDER STEADY STATE OPERATING CONDITIONS.	*Organisme éxécuteur* CEA/DMG GRENOBLE *Responsables* M. CHENEBAULT DMG
Date de démarrage 01.01.1976 — *Etat actuel* En cours *Date d'achèvement* 1983 — *Dernière mise à jour* Décembre 1982	*Scientifiques* M. SEVEON DSN M. STORA EDF

1) OBJECTIF GENERAL

Déterminer le taux de relâchement des produits de fission hors d'un crayon présentant une rupture de gaine et établir une relation entre la concentration d'espèces radioactives contaminant le circuit primaire, le nombre vrai de crayons défectueux et la gravité des défauts.

2) OBJECTIFS PARTICULIERS

1. Etudier le fonctionnement d'une rupture à différents niveaux de puissance stationnaire (expériences EDITH).

2. Etudier l'influence des transitoires sur le taux de dégagement en particulier des cyclages de puissance du type suivi de réseau ou téléréglage (expériences CYFON).

3. Evaluer et modéliser le terme source que constitue une rupture de gaine dans un réacteur, en vue de son utilisation comme module terme source dans le code PROFIP d'évaluation de la contamination du circuit primaire par les produits de fission.

3) INSTALLATIONS EXPERIMENTALES

Ces essais sont effectués dans des boucles de type thermosiphon (BOUFFON) installées dans le réacteur SILOE à GRENOBLE. En 1980, une nouvelle conception (BOUFFON-JET) a permis d'obtenir des conditions thermohydrauliques représentatives des conditions de fonctionnement d'un réacteur de puissance. En 1982, les boucles ont fonctionné avec un conditionnement chimique de l'eau plus représentatif (introduction d'hydrogène).

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4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

Avancement à ce jour

La partie expérimentale du programme sera achevée en 1983 par les essais suivants :

EDITH 5 : Essai à puissance stable avec rupture préfabriquée et calibrée sur un crayon instrumenté avec un capteur de pression. L'essai commencé fin 1982 est prolongé début 1983, après accord du DSN et d'EDF.

CRUSIFON 6 : Etude de l'évolution d'une rupture de gaine se produisant au droit de combustible sur un crayon à fort taux de combustion (35 000 MWJ/t).

MODELE 1 : Evolution de la thermique de combustible provoquée par l'entrée d'eau lors de la rupture et mesure des variations internes de pression après rupture. Cet essai programmé en 1982 a été reporté en 1983 suite aux difficultés technologiques rencontrées pour instrumenter le crayon.

Résultats essentiels

CRUSIFON 5 : (1) Rupture par interaction combustible-gaine à 24 000 MWJ/t représentative des défauts de gainage en réacteur. Les taux d'émission instantanés R/B* des gaz rares et des iodes sont faibles par rapport à ceux obtenus avec des crayons faiblement irradiés, ce qui provient soit du taux de combustion proprement dit, soit de la faible valeur du jeu combustible-gaine qui lui est associée.
Les examens métallographiques montrent que la structure de l'UO2 est différente de celle du crayon sain, notamment dans la zone périphérique.

Le DMG a effectué une synthèse des résultats du programme expérimental (2), en proposant des valeurs de taux d'émission des gaz et des halogènes hors de l'UO2 d'un crayon défectueux.
Les valeurs proposées sont cohérentes avec les taux d'émission observés sur les réacteurs de puissance.
Les changements de phase eau-vapeur dans le jeu sont déterminantes pour expliquer la sortie des produits de fission dans l'eau primaire à travers le défaut, une modélisation du fonctionnement durant les phases transitoires est en cours d'élaboration (code ANGELE).

5) PROCHAINES ETAPES

Cette étude doit s'achever en 1983. Le CTRC (Comité Technique Rejet Combustible) proposera un terme source "rejet des produits de fission hors de l'UO2" qualifié à partir de l'ensemble des résultats obtenus sur boucle(synthèse des essais BOUFFON effectuée par le DMG dans la référence (1)) et sur réacteurs (synthèse à effectuer par le CEA/DRE).

* $\frac{R}{B} = \frac{\text{Nombre d'atomes émis par seconde hors du combustible}}{\text{Nombre d'atomes formés par seconde dans le combustible}}$

6) RELATIONS AVEC D'AUTRES ETUDES

Activité des produits de fission dans les circuits des réacteurs à eau. Fiche étude 149-1-04.

7) DOCUMENTS DE REFERENCE

(1) Résultats de l'expérience CRUSIFON-5. Compte rendu DMG N° 46/82 du 14.6.82.

(2) Emission des produits de fission par les combustibles défectueux. Programme expérimental du DMG - Synthèse partielle - Compte rendu DMG N° DR 17/82 du 26/5/82.

Autres documents :

- Programme concentré CEA-DSN-EDF sur le comportement des éléments combustibles défectueux. Compte rendu DMG N° 40/82 du 25/5/82.
- Essais de perçage de gaine de zircaloy sur la maquette "MODEL 1". Note technique DMG/SER N° 9/82 du 29/3/82.

8) DEGRE DE DISPONIBILITE

Documents internes, non disponibles, sauf accord du CEA et d'EDF.

		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: 1974	COMPLETED:	SCIENTISTS: A.Foglio Para - Politecnico M.Mandelli-Bettoni " G.Sandrelli - ENEL
STATUS: In progress	LAST UPDATING: July 1980	

(*) 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.

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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

TITLE (ORIGINAL LANGUAGE) : Undersökning av defekta stavar från kraftreaktorer		CLASSIFICATION : 5.1 (18)
		COUNTRY : Sweden
		SPONSOR : Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE) : Examination of defect fuel rods from power reactors		ORGANISATION : Studsvik Energiteknik AB
		PROJECT LEADER : Tord Jonsson
INITIATED : 1981-01-01	COMPLETED : 1982-12-31	SCIENTISTS : R S Forsyth, T Jonsson
STATUS : Completed	LAST UPDATING : June 1983	

1. General Aim: Examination of fuel after long time irradiation in a defect condition.

2. Particular Objectives: Comparative measurements of retained fission products in intact and defect fuel and comparative measurements of leachability of fission products and actinides from intacts and defect fuel.

4. Project Status: Two fuel rods, one intact and one defect, with the same manufacturing and irradiation data have been examined in a comparative study. The defect rod has been irradiated in a defect condition during approximately one reactor cycle and has consequently some secondary defects. The defect rod has two penetrating defects at a distance of about 1.5 meters from each other. Comparison with the intact rod shows a large Cs loss from the defect rod, especially between the cladding defects, where the loss is measured to about 30 %. The leachability in deionized water is higher for Cs, U and Cm for fuel from the defect rod. The leaching results are more complex for Sr-90, Pu and Am. The fuel in the defect rod has undergone a change of structure with grain growth and formation of oriented fuel structure.

   The cladding of the defect rod is hydrided locally in some parts of the lower part of the rod and furthermore over a more extended region near the upper end of the rod.

7. Reference Documents: R S Forsyth, T Jonsson: Experimental study of defect power reactor fuel. STUDSVIK/NF(P)-82/72, 1982-12-14

8. Degree of Availability: The results are open.

		CLASSIFICATION : 5:1, 5:3, 5:4, 5:5 och 18
TITLE (ORIGINAL LANGUAGE) : Transuraner i reaktoravfall		COUNTRY : Sweden
		SPONSOR : S.S.I., National Institute of Radiation Protection
TITLE (ENGLISH LANGUAGE) : Transuranic Nuclides in Low-Level Wastes from Power Reactors in Sweden.		ORGANISATION : Dept. of Nuclear Chemistry Chalmers Univ. of Techn.
		PROJECT LEADER : J.O. Liljenzin
INITIATED : 1983-01-12	COMPLETED : 1984-09-30	SCIENTISTS : S. Johnson Tech. ass.: I-M. Ehrlund
STATUS : In progress	LAST UPDATING : 1983-06-01	

1. General Aim:

   To estimate the amount of transuranic nuclides originating from ion-exchange resins in the final storage for reactor waste.

2. Particular Objectives:

   To develop an analytical method for determination of the content of transuranic nuclides in ion-exchange resins.

   To correlate the content of transuranic nuclides or another measurable nuclide in the reactor cooling water with the content of transuranic nuclides in the ion-exchange resins.

   To achieve this, measurements on reactor cooling water and resins samples are made. Samples from both BWR and PWR reactors are analysed.

3. Experimental Facilities:

   Equipment for alfaspectrometry, etc.

4. Project Status:

   1. Progress to date: Testing of analytical method completed.

5. Next Steps:

   Testing of sampling methods of reactor cooling water and analysing kappa-sur ion-exchangers from BWR-reactors

8. Degree of Availability:

   Contact person: J.O. Liljenzin
   Department of Nuclear Chemistry,
   Chalmers University of Technology, S-412 96 GÖTEBORG
   SWEDEN

8369

		CLASSIFICATION : 5.1, 11.1
TITLE (ORIGINAL LANGUAGE) : Datorberäkningar på Haldens testfall för reaktorbränslebeteende		COUNTRY : Sweden
		SPONSOR : Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE) : Computer calculations on Halden test cases for reactor fuel behaviour		ORGANISATION : Studsvik Energiteknik AB
		PROJECT LEADER : K Malén
INITIATED : 1982-05-05	COMPLETED : 1982-12-31	SCIENTISTS : K Malén
STATUS : Completed	LAST UPDATING : June 1983	

1. General Aim: Fuel rod code benchmarking

2. Particular Objective: Comparison at codes used by STUDSVIK with Halden test cases

4. Project Status: Completed

7. Reference Documents: Malén K, Halden Fuel Modelling Workshop. Vääksy, Finland 15-19 juni 1982, STUDSVIK/NF(P)-82/71 (In Swedish)

8. Degree of Availability: Distribution restricted to Halden Project participants

	CLASSIFICATION : 5.1
TITLE (ORIGINAL LANGUAGE) : Modellering av fissionsgasfrigörelse från kraftreaktorbränsle vid hög utbränning (01/B1)	COUNTRY : Sweden
	SPONSOR : Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE) : Modelling of fission gas release from power reactor fuel at high burnup (01/B1)	ORGANISATION : Studsvik Energiteknik AB
	PROJECT LEADER : K Malén

INITIATED : 1981-08-24	COMPLETED : 1982-12-31	SCIENTISTS : K Malén
STATUS : Completed	LAST UPDATING : June 1983	

1. General Aim: Modelling of fission gas release in power reactor fuel at high burnup

2. Particular Objectives: Comparison with puncturing results from 50 rods from the reactors 01 and B1

3. Project status: Completed

7. Reference Documents: Malén, K, Modellering av fissionsgasfrigörelse i bränslestavar från 01/B1-50stavsprogrammet STUDSVIK/NF(P)-82/70 (In Swedish)

8. Degree of Availability: Distribution restricted to Sweden

5.2. RELEASE FROM OVERHEATED FUEL-ELEMENTS

Berichtszeitraum/Period 1.1.-31.12.1982	Klassifikation/Classification 5.2	Kennzeichen/Project Number 06.01.11/25A, 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 Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe Projekt Nukleare Sicherheit Inst. für Radiochemie
Arbeitsbeginn/Initiated 1973	Arbeitsende/Completed 1984	Leiter des Vorhabens/Project Leader Dr. H. Albrecht
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating	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 Release experiments with 200 - 250 g of corium containing slightly radioactive fission (UO_2 with a simulated burn-up in the range of 40 000 MWd/t).

3.2 Investigation of the release during melt/concrete interaction.

3.3 Characterization of the I- and Cs-species in the primary system as well as their interaction with aerosol particles and steel surfaces.

4. Experimental Facilities, Computer Codes

Melting plant SASCHA including a transport and collection system for the released products; facility for the production of slightly radioactive fissium; facilities for gamma spectrometry and analysis.

5. Progress to Date

5.1 Two tests were carried out with improved aerosol collection techniques (automatic filter changer) to determine the release rates of Ru, Ba, Sn, and In in steam at 2400 °C.

5.2 Two tests were performed with corium melts in ZrO_2 crucibles which intentionally failed at 2500 °C and caused the melt to react with a concrete crucible. While penetrating into the concrete, the melt was inductively heated for 6 to 8 min, and the resulting aerosol was collected in time intervals of 1 minute.

5.3 The release characteristic of I and Cs in a steam atmosphere was studied for varying compounds of these elements in the fuel (fissium).

6. Results

6.1 The results of the tests mentioned under 5.1 are summarized in the following table:

Element	Time at 2400 °C (min)	Integral Release (%)	Average Release Rate (%/min)
Ru	7.5	0.0003	0.00004
Ba	14	0.11	0.008
Sn	14	30	2.6
In	14	63	7.1

As a special feature of the release behavior of some elements, it was found that the release rates at constant temperature were not independent of time. E.g. for Ba and In, the release rates at 2400 °C related to the current inventory decreased by a factor of 3 within 12 minutes while they remained constant for Ru, Mo and Np.

6.2 During the melt/concrete interaction, the release rates were found to be extremely time dependent. This was due to the strong reaction in the first minutes and to the temperature decrease of the melt which could not entirely be avoided by the hf-heating. Some quantitative results can be seen from the following table:

Element	Integral Release (%)	Maximum Release Rate (%/min)	Average Release Rate (%/min)
Te	42	13.5	7.5
Sb	4.7	1.1	0.7
Ag	58	15.7	11.7
Mo	0.08	0.015	0.011
Ru	0.007	0.0014	0.0010

6.3 If the fissium contains I and Cs only in the form of CsI, the release characteristics of both elements were nearly identical. This indicates that CsI is stable in the presence of UO_2 and all other fission products up to at least 2200 ^{o}C. If Cs (or I) is not included in the fission product inventory of fissium, the main release of I (or Cs, respectively) occurs at temperatures which are about 300 ^{o}C lower than in the preceding case.
If I and Cs are both present as HIO_3 and Cs_2CO_3 and if the melt contains small amounts of absorber material (Ag-In-Cd), then the aerosol was found to contain most of the I in the form of AgI and only 0.1 % as a gaseous species.

7. Next Steps
Further investigation of the physical and chemical behavior of the released iodine, cesium, and tellurim species.

8. Relation with other Projects
PNS-4311, PNS-4314, PNS-4317

9. References
H. Albrecht, H. Wild:
Experimentelle Bestimmung von Quelltermen beim LWR-Kernschmelzen
Atomwirtschaft, Heft 1, 1983, p23

10. Degree Of Availability of the Reports
Unrestricted distribution

148.1.01			5 - 2
TITRE FLASH : EVALUATION DU TAUX D'EMISSION DES PRODUITS DE FISSION HORS D'UN CRAYON COMBUSTIBLE AU COURS D'UN ACCIDENT DE PERTE DE REFRIGERANT PRIMAIRE DANS UN REACTEUR A EAU SOUS PRESSION.			*Pays* FRANCE *Organisme directeur* CEA/IPSN EDF/SEPTEN
TITRE en anglais FLASH : ASSESSMENT OF THE FISSION PRODUCT RELEASE RATE OUT OF THE FUEL CLADDING DURING A LOSS OF COOLANT ACCIDENT IN A PRESSURIZED WATER REACTOR.			*Organisme éxécuteur* CEA/DMG GRENOBLE *Responsables* M. CHENEBAULT DMG
Date de démarrage 1977	*Etat actuel* En cours		*Scientifiques* M. DUCO DSN M. STORA EDF M. BRUCHET DMG
Date d'achèvement 1984	*Dernière mise à jour* Mars 1983		

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 de 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 entre la gaine et l'U02.

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.

PLANNING

- Prise en mains de l'installation expérimentale - Essai FLASH 1 - 1980.

- Etude du relâchement des produits de fission dans un scénario du type LOCA grosse brèche, mais en atmosphère non oxydante et avec renoyage différé à froid - Essai FLASH 2 - 1980.

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- Etude du relâchement des produits de fission dans un scénario du type LOCA grosse brèche, sous atmosphère de vapeur d'eau, mais avec renoyage différé à froid - Essai FLASH 3 - 1982.

- Etude du relâchement des produits de fission dans un scénario du type LOCA grosse brèche, sous atmosphère de vapeur d'eau et avec un renoyage rapide à chaud - Essai FLASH 4 - 1982.

- Etude du relâchement des produits de fission d'un crayon combustible très fortement irradié, dans un scénario du type LOCA grosse brèche, sous atmosphère de vapeur d'eau et avec un renoyage différé à froid - Essai FLASH 5 - 1984.

- Dépouillement des résultats et modélisation - 1982-1984.

3) INSTALLATIONS EXPERIMENTALES

Les essais sont réalisés dans un dispositif composé d'une partie basse GRIFFON, et d'une partie haute FLASH, installé dans le réacteur SILOE à GRENOBLE (réf. 1).

4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

Les essais FLASH 1 (réf. 2) et FLASH 2 ont été effectués respectivement en février et novembre 1980.

Lors de l'essai FLASH 2, une fissuration traversante de la gaine a été obtenue après un ballonnement court ; cette taille de la rupture n'a pas permis un lessivage interne sensible du crayon lors du renoyage à froid.

Les fractions de gaz de fission relâchées correspondent à celles libérées dans le crayon avant la séquence accidentelle,et l'analyse des structures entourant le crayon ne permet de mettre en évidence aucun dépôt de produits de fission solides ou volatils.
Les fractions relâchées de ces produits de fission dans l'eau de renoyage (effectué à froid) restent extrêmement faibles (réf. 3).

Des essais sur maquette hors pile ont par ailleurs été réalisés pour préparer les essais FLASH 3 et 4 ; les caractéristiques initiales des crayons expérimentaux ont été redéfinies de manière à obtenir une rupture importante de la gaine, en phase alpha (réf. 4 et 5).

Les essais FLASH 3 et FLASH 4 ont été exécutés respectivement en septembre et décembre 1982.

Lors de l'essai FLASH 3 (réf. 6), on a obtenu, conformément aux prévisions des codes Bicycle (EDF) et Cupidon (CEA) (réf. 7), une rupture franche de la gaine avec déformation de type " ballonnement court".
Cette rupture s'est produite à une température de gaine de 930°C.
Comme dans les deux précédentes expériences, FLASH 1 et FLASH 2, le relâchement des gaz de fission correspond au taux de sortie des gaz dans le crayon sain avant la séquence accidentelle.

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Les fractions de produits de fission relâchée dans l'eau de renoyage à froid sont très faibles, du même ordre de grandeur que dans les deux premières expériences. La tenue de l'empilement de combustible reste excellente et malgré les conditions assez sévères de fonctionnement, - température de la gaine entre 1000°C et 1270°C pendant une minute en atmosphère de vapeur d'eau - l'oxydation externe de la gaine reste très faible.

L'essai FLASH 4 a été l'objet d'une étude préliminaire de sûreté montrant que, même dans l'hypothèse de l'écroulement de la colonne combustible lors du renoyage, il n'y aurait pas de risque de fusion des fragments de combustible accumulés en fond de boitier pendant le maintien de la puissance résiduelle (réf. 8).

En ce qui concerne les gaz de fission libérés lors de l'essai, les résultats préliminaires d'analyse montrent qu'il n'y a pas d'augmentation du taux de relâchement par rapport à l'essai FLASH 3 : la fraction relâchée correspond à celle contenue dans le jeu U02-gaine avant la séquence expérimentale.

Une première estimation des principaux produits de fission entraînés par l'eau de renoyage (renoyage à chaud avec maintien pendant 10 mn de la puissance résiduelle) montre que par rapport à l'essai FLASH 3 (renoyage à froid) :

- la quantité d'iode est plus que doublée, mais pas de changement apparent pour les autres produits de fission volatils (Cs),

- la quantité de produits de fission solides augmente d'un facteur compris entre un et deux ordres de grandeur.

5) PROCHAINES ETAPES

La réalisation d'un essai complémentaire - FLASH 5 -, analogue à FLASH 3, mais portant sur du combustible fortement irradié (40 OOOMWJ/t) est en cours de discussion entre CEA/DSN et EDF.
L'intérêt d'un tel essai, qui ne serait pas réalisé avant 1984, serait de permettre de s'assurer que, lors d'un transitoire du type LOCA grosse brèche, il n'y a pas mise en jeu de mécanismes de relâchement supplémentaires par rapport à ceux identifiés lors des essais effectués avec du combustible faiblement irradié (essais FLASH 1 à 4).

Le dépouillement des essais FLASH 3 et 4 sera par ailleurs poursuivi en 1983, dans le but de modéliser les sorties de produits de fission en cas de transitoire LOCA sur des combustibles irradiés jusqu'à environ 3000 MWJ/t.

6) RELATIONS AVEC D'AUTRES ETUDES

Fiche étude 140-1-02. Evaluation du taux de relâchement des produits de fission lors d'une rupture de gaine dans un réacteur à eau sous pression. Fonctionnement normal.

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Fiche étude 170-1-04. Programme PHEBUS. Expérience de dépressurisation et de renoyage d'une grappe combustible PWR.

Fiche étude 148-1-07. Etude expérimentale et modélisation du taux de relâchement des produits de fission d'un crayon combustible d'un coeur de PWR, dans l'hypothèse d'un transitoire accidentel sévère hors dimensionnement.

7) DOCUMENTS DE REFERENCE

1. Compte rendu DMG N° 79-125.
   ExpériencesFLASH.

2. Compte rendu DMG N° 81-42. Mai 1981.
   FLASH 1 - Examens destructifs.

3. Compte rendu DMG N° 85-82. Janvier 1983.
   Expérience FLASH 2.
   "Déroulement de l'expérience et sortie des produits de fission volatils et solides".
   Relâchement des gaz de fission.
   Résultats des examens post-irradiatoires.

4. Compte rendu DMG N° 65-81. Octobre 1981.
   "Résultats des essais effectués au moyen de la maquette FLASH en vue de caractériser les conditions des expériences FLASH 3 et 4".

5. Compte rendu DMG N° 80-82. Octobre 1982.
   Fiche de fabrication du crayon combustible FLASH 3.

6. Compte rendu DMG N° 86-82. Novembre 1982.
   Expérience FLASH 3.
   Déroulement de l'expérience et résultats.

7. Note technique DTech/SECS/SEECRE N° 82-934.
   FLASH 3 - Comparaison expérience- calcul.

8. Compte rendu DMG N° 06-83. Février 1983.
   Expérience FLASH 4.
   "Comportement du combustible du crayon dans l'hypothèse d'une destruction éventuelle de la gaine lors du choc thermique lié au renoyage".

8) DEGRE DE DISPONIBILITE

Documents internes non disponibles, sauf accord du CEA et de l'EDF.

148.1.02	5 - 2
TITRE Programme PITEAS - Etudes des transferts de radioactivité à l'intérieur de la centrale - Filtration rustique	*Pays* FRANCE *Organisme directeur* CEA/DSN EDF
TITRE en anglais PITEAS project - Study of the transfers of radioactivity inside the plant - Rustic filtration	*Organisme exécuteur* CEA/DSN/SESTR CAD CEA/DCAEA/SEA GRE CEA/DPr/SPT FAR *Responsables* J.FERMANDJIAN DSN/SAER M.LUCAS DSN/SESTR
Date de démarrage 1/1/1981 *Fin d'achèvement* 31/12/1985 *Etat actuel* En cours *Dernière mise à jour* Décembre 82	*Scientifiques* M.LUCAS DSN/SESTR J.CERF DSN/SESTR D.BOULAUD DPr/SPT Mme A.M. PERROUD DCAEA/SEA D. MAGNAUD DEMT/SMTS

1/ OBJECTIF GENERAL

Détermination des conséquences radiologiques des rejets de produits de fission dans l'environnement à la suite d'un accident hors-dimensionnement.

2/ OBJECTIFS PARTICULIERS

- Comportement des produits de fission (aérosols et iodes) dans l'enceinte du réacteur

. Etudes expérimentales sur les aérosols visant à améliorer nos connaissances dans les domaines suivants : condensation de la vapeur d'eau sur les aérosols, coagulation par sédimentation différentielle et diffusiophorèse.

. Etudes expérimentales sur la chimie de l'iode permettant de chiffrer la fraction de l'iode qui serait sous forme volatile dans l'enceinte.

- Détermination des caractéristiques d'un filtre rustique d'efficacité 10 pour les aérosols (utilisation prévue pour écrêtage de la pression enceinte à 5 bars lors d'accidents sévères).

3/ INSTALLATIONS EXPERIMENTALES

. Installations de laboratoire
  - sur la filtration des aérosols (DSN/SESTR)
  - sur la chimie de l'iode (DSN/SESTR - DCAEA/SEA)

. Installation PITEAS (enceinte en acier de 3m^3) adaptable aux trois options du programme : filtration granulaire, comportement des aérosols et filtration piscine (DSN/SESTR).

. Enceinte en acier 6m^3 du DEMT : études des mécanismes de dépôt des aérosols sur les parois (dôme, murs et sol) d'une enceinte.

4/ ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

a/ Chimie de l'iode

. Mesure du taux de radiolyse de CsI des gouttelettes : environ 5 % par MRad, en milieu neutre, et 0,5 %, en milieu basique, de I^- est oxydé en iode moléculaire.

. Calcul du coefficient de partage de l'iode à partir de la mesure expérimentale à 140°C de la constante cinétique de dismutation de I_2 en I^- et IO_3^-.

. Interaction iode-acier 304 L : possibilité de séparer les iodes chimisorbé et physisorbé.

b/ Filtration rustique

Filtration granulaire

Mise au point d'une installation de filtration en laboratoire et réalisation de 40 essais de filtration avec différents sables, à 140°C et un mélange air (62 %) - H_2O (38 %).

Les résultats obtenus relatifs à l'efficacité sont voisins de ceux obtenus dans les laboratoires étrangers.

Filtration piscine

Piscine efficace pour le piégeage des aérosols de rayon 1µm.

c/ Physique des aérosols

. Etude préliminaire sur la génération d'aérosols solubles par pulvérisation acoustique.

. Etude préliminaire sur la possibilité de mesurer des aérosols dans des conditions de température et de pression élevées.

. Mise en évidence de 3 zones de dépôt sur les parois latérales d'une enceinte (essais DEMT/DPr).

d/ Boucle PITEAS

Mai 81 : Etude de conception
Juin 82 : Appels d'offre
Décembre 82 : Décision de réalisation

5/ PROCHAINES ETAPES

a/ Chimie de l'iode

- Radiolyse de solutions concentrées de CsI.
- Mesure de la constante de dismutation de I_2 en I^- et IO_3^- avec des solutions concentrées de CsI.
- Formation ICH_3 par radiolyse.
- Interaction iode-acier 304 L : cinétique de dépôt, profondeur de pénétration, rétentions réversible et irréversible.

b/ Filtration rustique

Filtre à sable (sable CATTENOM tamisé).

. Banc d'essai en verre : Mesures de l'efficacité de la perte de charge, du colmatage et de la durée du transitoire dans les conditions expérimentales suivantes :

Gaz	Vitesse	7-28 cm/s
	Température	135°C
	Composition	62% air - 38% H_2O
Aérosols	Nature	CO_3Cs_2
	AMMD	0,66-1,4 µm
Filtre	Diamètre	20 cm
	Hauteur	80 cm

. Banc d'essai en acier inoxydable : Banc de diamètre 40 cm et de hauteur 80cm, permettant :

- d'assurer la transition entre le banc d'essai en verre (diamètre = 20cm) et la boucle PITEAS (diamètre = 1 m),

- d'autoriser des pressions différentielles plus élevées de part et d'autre du lit de sable que celles permises avec le banc en verre (lors des essais de colmatage.

c/ Physique des aérosols

- Mise au point d'un dispositif de génération d'aérosols solubles fonctionnant à 5 bars et 140°C.
- Etude bibliographique sur la génération d'aérosols insolubles.

- Etablissement des formalismes de dépôt des aérosols sur le dôme et sur le plancher d'une enceinte (essais DEMT/DPr).

d/ Boucle PITEAS

- Calendrier de réalisation
  Janvier 83 : Appels d'offre "réactualisés".
  Juillet 83 : Réalisation des composants.
  Décembre 83 : Montage de l'ensemble sur le site.

- Configurations de la boucle PITEAS
  . Filtration granulaire (hauteur = 2m - diamètre = 1 m - injection des aérosols par le haut).
  . Filtration piscine.
  . Physique des aérosols (hauteur hors-tout = 3 m - diamètre = 1,2 m - cuve thermostatée.)

6/ RELATIONS AVEC D'AUTRES ETUDES

Code de calcul JERICHO.

Modélisation du comportement des produits de fission dans les accidents des réacteurs à eau - Fiche 148.1.09.

7/ DOCUMENTS DE REFERENCE

- M. LUCAS et al.
  Iodine behavior in PWR accidents leading to core damage*
  International Meeting on Thermal Nuclear Reactor Safety
  29 août - 2 septembre, 1982 CHICAGO.

- M. LUCAS, R. VENTRE
  Coefficient de partage de l'iode moléculaire entre l'eau et l'atmosphère gazeuse la surmontant
  1/ Détermination de la vitesse de dismutation de l'acide hypoideux à 140°C
  Note technique DSN/SESTR n°82/97 Février 1982.
  2/ Calcul des coefficients de partage à 140°C en fonction du pH de l'eau, de la concentration de l'iode et du temps écoulé
  Note technique DSN/SESTR n°82/101 Mars 1982.

- M. LUCAS, R.VENTRE
  Transformation de l'iodure de césium en iode moléculaire sous l'influence des radiations γ . Valeurs du coefficient de partage de l'iode en fonction du pH et du temps.
  Note technique DSN/SESTR n°82/116 Septembre 1982.

- M. LUCAS et al
  Etude du pouvoir filtrant de lits de sable et billes de verre vis-à-vis d'aérosols de CO_3Cs_2 : Comparaison avec les résultats obtenus dans la littérature
  Rapport DSN/SESTR en cours de diffusion

- J.CERF
  Programme PITEAS. Spécifications relatives à la cuve
  Note technique DSN/SESTR n°82/102 Mars 1982.

- J.CERF, J.C. DURAND
  Programme PITEAS. Projet de procédure d'exécution des essais. Configuration des circuits
  Note technique DSN/SESTR n°82/106 Mai 1982.

- D.BOULAUD
  Production d'aérosol dans les expériences du projet PITEAS
  Rapport DPr/SPT n°81/260 Septembre 81
  Mesure de la granulométrie de gouttelettes dans l'expérience PITEAS
  Rapport DPr/SPT n°81/261 Septembre 81
  Propositions pour la métrologie des aérosols dans l'expérience PITEAS
  Rapport DPr/SPT n°82/313 Juin 82.

- P. PEPIN
  Efficacité d'une piscine pour le piégeage des aérosols
  Note technique DEMT/SYST n°82/046 Juillet 82.

- C. HERVOUET
  Comportement des aérosols dans un réacteur PWR en cas d'accident*.
  Thèse Docteur-Ingénieur - Université Paris Val-de-Marne (PARIS XII) Créteil, Janvier 1983.

- R. ABID
  Etudes du transport et du dépôt des aérosols dans une couche limite.*
  Thèse Docteur-Ingénieur, Université PARIS VI, Décembre 1982.

- D.BOULAUD, J.C. GUICHARD
  Réalisation d'une installation d'étude en laboratoire des problèmes de filtration d'aérosols.
  Rapport DPr/SPT n°269 Février 1982.

8/ DEGRE DE DISPONIBILITE

Documents internes, non disponibles (sauf accord du CEA), à l'exception des documents repérés par *.

148.1.07		5 - 2
TITRE ETUDE EXPERIMENTALE ET MODELISATION DU TAUX DE RELACHEMENT DES PRODUITS DE FISSION D'UN CRAYON COMBUSTIBLE D'UN COEUR DE PWR, DANS L'HYPOTHESE D'UN TRANSITOIRE ACCIDENTEL SEVERE HORS DIMENSIONNEMENT		*Pays* FRANCE *Organisme directeur* CEA/IPSN
TITRE en anglais EXPERIMENTAL STUDY AND MODELING OF THE FISSION PRODUCT RELEASE RATE FROM A FUEL ROD OF A PWR CORE, ASSUMING THE OCCURRENCE OF A BEYOND-DESIGN, SEVERE ACCIDENTAL TRANSIENT		*Organisme éxécuteur* CEA/DMG GRENOBLE *Responsables* M. CHENEBAULT DMG
Date de démarrage 1979	*Etat actuel* En cours	*Scientifiques* M. DUCO DSN M. LEFORT DMG
Date d'achèvement 1986	*Dernière mise à jour* Mars 1983	

1) OBJECTIF GENERAL

Dans l'optique de l'évaluation réaliste du risque vis-à-vis du public, l'objectif général de la présente étude est la compréhension des mécanismes de la dégradation de crayons combustibles irradiés (500 à 40 000 MWJ/t) dans le domaine de température 1200°-2800°C, la quantification des taux des rejets concomitants de produits de fission tels I, Cs, Te et Ru, l'identification des formes physiques et chimiques de ces produits de fission en fonction des conditions d'ambiance accidentelle, et la modélisation en collaboration avec les autres parties concernées (DMECN, DSN) des résultats (phénoménologie de la dégradation du crayon, taux d'émission des divers produits de fission) sous des formes qualifiées, utilisables dans des codes d'accidents sévères des PWR.

2) OBJECTIFS PARTICULIERS

- Suivi et exploitation continus des résultats obtenus quant aux produits de fission dans le cadre d'accords avec des laboratoires étrangers.
- Mise au point de moyens expérimentaux hors pile pour l'étude de la dégradation sévère et de l'émission de produits de fission de crayons combustibles irradiés.
  L'accent sera mis sur la métrologie des I, Cs, Te et Ru et de leurs formes physiques et chimiques. Les conditions d'ambiance pourront être ajustées (présence d'hydrogène, d'un brouillard d'eau, d'aérosols de matériaux de contrôle et/ou de structures) - 1981-1984.
- Exécution d'essais analytiques hors pile sur des crayons irradiés (500 à 40 000 MWJ/t). Gamme de températures : 1200°C-2800°C (1982-1985).
- Modélisations de la dégradation d'un crayon combustible et du taux de sortie des produits de fission (1985-1986). Ces modélisations devraient pouvoir être utilisées pour les pré-calculs de sûreté de certains essais FLASH CSD et PHEBUS, phases III et IV.

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- Mise au point du dispositif d'essai en pile FLASH CSD (1979-1983).
- Exécution d'essais intégraux FLASH CSD, parallèles aux essais PHEBUS phases III et IV, représentatifs sur le plan des phénomènes de dégradation dans des régions sélectionnées d'un coeur de PWR pendant des créneaux de temps cruciaux de séquences accidentelles sévères types (1984-1986).
- Qualification des modélisations sur les résultats des essais intégraux FLASH CSD et PHEBUS phases III et IV (1986).

3) INSTALLATIONS EXPERIMENTALES

- Four HEVA en cellule blindée au DMG.
- Boucle d'essais en pile FLASH CSD, avec maquette hors pile (étude préliminaire de simulation de la phase dénoyée du transitoire accidentel).

4) ETAT D'AVANCEMENT

Le four HEVA permettant d'atteindre 1800°C en atmosphère de vapeur d'eau a été conçu et réalisé. Il utilise la zircone comme isolant thermique et le chauffage se fait par induction (suscepteur en graphite protégé de l'oxydation par un courant d'hélium).
Ce four permet de tester des tronçons de crayon PWR contenant sept pastilles, soit environ 10 cm de long. Lors des deux premiers essais effectués en atmosphère d'hélium, il a été observé une fusion partielle de la gaine ainsi qu'une réaction importante entre le zircaloy et le combustible. Les deux essais suivants, effectués en atmosphère de vapeur d'eau, ont conduit à une oxydation complète de la gaine. D'autre part, des interactions complexes ont été observées aux endroits où le contact entre le combustible et la gaine était préservé (ségrégations des composants de l'alliage de gaine).

Un premier dossier sur l'expérience FLASH CSD 01 a été élaboré.
Le scénario retenu est le suivant :
l'échantillon de combustible sera pré-irradié jusqu'à 2000 MWJ/t dans la boucle à thermosiphon coaxial ; lors de la phase accidentelle, le crayon sera rapidement porté à 1800°C (5°C/s) en présence de vapeur d'eau pure renoyé à chaud ; un prélèvement de la phase gazeuse sera effectué en continu, les pressions et les températures seront mesurées in situ.
FLASH CSD 01 est un essai de prise en mains de l'installation.

5) PROCHAINES ETAPES

Une vingtaine d'essais sont prévus en 1983-1984 dans le four HEVA en cellule ; ces essais sont en principe complémentaires de ceux effectués aux Etats-Unis (ORNL) et en RFA (KFK), essais dont nous disposons des résultats dans le cadre d'accords d'échanges.

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En ce qui concerne le programme FLASH CSD, un premier bilan sera fait après le 1er essai, en 1983, quant aux performances réelles et potentielles de l'installation.
Ensuite, sera définie une grille d'essais intégraux, représentatifs d'éléments de transitoires accidentels sévères dans un coeur de PWR ; ces essais constitueront le prolongement, dans le domaine du relâchement des produits de fission, des essais des phases III et IV du programme PHEBUS.

En fonction des difficultés rencontrées, des essais à caractère phénoménologique pourront également être proposés. L'ensemble de ces essais apportera une contribution déterminante à la qualification de modèles de taux de sortie des produits de fission, modèles établis jusqu'alors à partir de résultats d'essais hors pile.

6) RELATIONS AVEC D'AUTRES ETUDES

- Fiche 148-1-01. FLASH. Evaluation du taux d'émission des produits de fission hors d'un crayon combustible au cours d'un accident de perte de réfrigérant primaire dans un réacteur à eau sous pression.

- Fiche 170-2-01. Préparation du programme PHEBUS CSD (coeur sévèrement dégradé).

7) DOCUMENTS DE REFERENCE

1 - Compte rendu DMG n° 117/79 - Septembre 1979.
Proposition d'étude du dégagement des produits de fission d'un crayon combustible PWR dans le cas d'une perte de réfrigérant du type petite brèche.

2 - Note technique DMG/SER n° 81/31 - Septembre 1981.
Comportement des crayons combustibles en régime accidentel.

3 - Compte rendu DMG n° 81/72 - Novembre 1981.
Etudes de base sur le comportement du combustible PWR en situation de coeur sévèrement dégradé. Premiers essais en atmosphère de vapeur d'eau.

4 - Compte rendu DMG n° 75/81 - Novembre 1981.
Etude du comportement en pile d'un crayon combustible PWR au cours d'un accident de type "coeur sévèrement endommagé".

5 - Compte rendu DMG n° 82/95.
Etude du comportement en pile d'un crayon combustible de la filière à eau pressurisée au cours d'un accident de type "coeur sévèrement dégradé".
Expérience FLASH CSD 01 - Projet de dossier n° 1.

8) DEGRE DE DISPONIBILITE

Documents internes non disponibles, sauf accord du CEA.

148.1.09	5 - 2
TITRE	*Pays* FRANCE
CODE DE CALCUL JERICHO MODELISATION DU COMPORTEMENT DES PRODUITS DE FISSION DANS L'ENCEINTE DE CONFINEMENT LORS D'ACCIDENTS SEVERES SUR LES REACTEURS A EAU	*Organisme directeur* CEA/IPSN
TITRE en anglais JERICHO COMPUTER CODE MODELISATION OF THE BEHAVIOR OF THE FISSION PRODUCTS IN THE CONTAINMENT BUILDING DURING SEVERE PWR ACCIDENTS.	*Organisme éxécuteur* CEA/DSN/SAER FAR CEA/DEMT/SYST SACLAY *Responsables* J.FERMANDJIAN/DSN/SAER A.L'HOMME/DEMT/SYST
Date de démarrage 1.1.1981 — *Etat actuel* En cours *Date d'achèvement* 31.12.1985 — *Dernière mise à jour* Décembre 82	*Scientifiques* Y.BERTHION/DEMT/SYST J.M.EVRARD/DSN/SAER

1/ OBJECTIF GENERAL

Acquisition , mise au point et couplage des codes de calcul permettant le calcul réaliste des transferts de masse et de radioactivité dans l'enceinte de confinement correspondant aux séquences accidentelles dominantes.

2/ OBJECTIFS PARTICULIERS

. Mise en place des moyens de calcul des scénarios d'accidents hors-dimensionnement -(1981-1983)

. Calcul des scénarios d'accidents - Etudes de sensibilité -(1983-1985)

- Brèche 5 cm en aval du GV
- TMLB'
- RTGV + RTV
- ....

. Validation expérimentale des codes JERICHO et AEROSOLS : essais PITEAS effectués au DSN/SESTR à Cadarache -(1985)

3/ INSTALLATIONS EXPERIMENTALES

NEANT.

4/ ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

a/ Avancement à ce jour

- Acquisition et mise en place des codes de calcul suivants (données d'entrée du code JERICHO) :

. Thermohydraulique du circuit primaire	
Phase de dépressurisation	RELAP 4 Mod6 (INEL/USA)
Phase fusion du coeur	BOIL/MARCH (BCL/USA)
. Traversée de la cuve	HEAD/MARCH (BCL/USA)
. Interaction corium-béton	INTER/MARCH (BCL/USA)

- Code JERICHO

Opérationnel avec les modules suivants :

. Aspersion/recirculation
. Dégonflage enceinte
. Combustion hydrogène
. Puissance résiduelle

- Code AEROSOLS opérationnel
  - . Version A1 à hypothèse log-normale
  - . Version A2 à hypothèse log-normale (avec prise en compte de la condensation/évaporation sur les aérosols, possibilité de traiter un mélange de gaz et module performant d'intégration numérique).
  - . Version B1 (à spectre granulométrique discrétisé) en cours de test.

b/ Principaux résultats

- Calcul d'un scénario d'accident hors-dimensionnement (brèche 5 cm en aval du GV sans injection de secours et sans aspersion) : taux d'humidité relative dans l'enceinte inférieur à 1 lors de l'émission des produits de fission ; ainsi, l'influence de la vapeur d'eau sur les aérosols semble faible, à l'exception de la diffusiophorèse (dépôt des aérosols sur les parois de l'enceinte par condensation de la vapeur d'eau).

- Etudes de sensibilité avec JERICHO : mise en évidence de l'importance de la connaissance du coefficient d'échange entre le gaz et les parois de l'enceinte.

- Etudes de sensibilité avec AEROSOLS : mise en évidence de l'importance de la connaissance de l'efficacité de collision (coagulation par sédimentation différentielle).

5/ PROCHAINES ETAPES

- Mise au point des modules JERICHO : aérosols, iode, corium-béton.
- Ecriture d'une version JERICHO compartimentée.
- Intégration du formalisme de diffusiophorèse dans le code AEROSOLS/A2.
- Intégration des formalismes de diffusiophorèse et de condensation de vapeur d'eau sur les aérosols dans le code AEROSOLS/B1.
- Etudes de sensibilité et de scénarios (TMLB', RTGV + RTV, ...).
- Calculs prévisionnels des futurs essais PITEAS (enceinte en acier de 3m^3) avec le code AEROSOLS.

6/ RELATIONS AVEC D'AUTRES ETUDES

- Développement et évaluation des codes de calcul "enceinte" - Fiche 142.3.03.
- Production et comportement de l'hydrogène en situation accidentelle - Fiche 146.2.05.
- Accidents des réacteurs à neutrons rapides - Modélisation et codes de calcul du comportement des aérosols sodés - Fiche 156.1.01.
- Etude probabiliste du risque lié à une centrale PWR - Fiche 144.3.04

7/ DOCUMENTS DE REFERENCE

- C.DEVILLERS, J.FERMANDJIAN
  Transport des produits de fission dans les accidents de réacteurs à eau sous pression.- Séminaire sur la rétention d'iode dans les effluents gazeux de l'industrie nucléaire - 21-24 septembre 1981, MOL (Belgique).

- C.HERVOUET
  Comportement des aérosols dans un réacteur PWR en cas d'accident.- Thèse Docteur-Ingénieur - Université PARIS Val-de-Marne (PARIS XII), CRETEIL, Janvier 1983.

- A. L'HOMME et al
  JERICHO-Code de calcul réaliste du comportement thermodynamique et radiologique des enceintes de confinement PWR.- Note technique DEMT/SYST n°82/036 Juin 1982.

- Y. BERTHION
  Description physique du noyau thermodynamique du code JERICHO.- Note techni-

que DEMT/SYST n°82/056 - Septembre 1982.

- A. L'HOMME
  Code AEROSOLS/A1 - Présentation.- Note technique DEMT/SYST n°80/024 - Août 1980.

- A. L'HOMME et al
  Présentation du code AEROSOLS/A2
  Note technique DEMT/SYST n°82/061 - Octobre 1982.

- Y. BERTHION et al
  Modélisation des échanges aux parois d'une enceinte PWR par le code JERICHO
  Note technique DEMT/SYST n°82/020 - Mars 1982.

8/ DEGRE DE DISPONIBILITE

Documents internes, non disponibles (sauf accord du CEA), à l'exception des deux premiers.

149-1-04		5 - 2
TITRE ACTIVITE DES PRODUITS DE FISSION DANS LES CIRCUITS DES REACTEURS A EAU *TITRE en anglais* PRODUCT FISSION ACTIVITY IN PWRs CIRCUITS		*Pays* FRANCE *Organisme directeur* CEA/ IPSN *Organisme éxécuteur* CEA/DRE CADARACHE *Responsables* P. BESLU DRE
Date de démarrage 1/77	*Etat actuel* En cours	*Scientifiques* J.J SEVEON DSN M. LEUTHROT DRE
Date d'achèvement 1984	*Dernière mise à jour* Décembre 1982	

1) OBJECTIF GENERAL

Elaboration d'un code permettant de calculer l'activité de l'eau primaire en fonction du nombre et de la gravité des ruptures de gaine et des conditions de fonctionnement du réacteur.
Développement du code pour calculer le relâchement des produits de fission hors du combustible en situation accidentelle.

2) OBJECTIFS PARTICULIERS

- Calcul prévisionnel de l'activité primaire pour les PWR 900 MWe et 1300 MWe.
- Détermination du nombre et de la gravité des ruptures de gaine à partir des mesures d'activité primaire.
- Modélisation du comportement des produits de fission solides et des émetteurs alpha.
- Traitement des régimes transitoires en particulier le téléréglage et le suivi de charge.

3) INSTALLATIONS EXPERIMENTALES

Sans objet.

4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

1) Avancement à ce jour

Le code PROFIP calcule le relâchement des produits de fission dans le circuit primaire en 4 étapes.

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- Module 1 : calcul de l'activité des produits de fission dans l'UO2.

- Module 2 : transfert des produits de fission de l'UO2 vers le jeu combustible-gaine. Les différents processus de rejet pris en compte sont le recul et l'éjection qui sont indépendants de la température et la diffusion qui, elle, dépend de la température.

- Module 3 : transfert des produits de fission du jeu vers le fluide primaire à l'aide d'un modèle empirique. La taille de la rupture est prise en compte par un coefficient vg équivalent à un taux de fuite (s^{-1}); on considère, en outre, que les iodes et les césiums ont la possibilité de se piéger partiellement sur la paroi interne de la gaine.

- Module 4 : l'activité des produits de fission dans l'eau primaire est calculée en prenant en compte les différents termes d'épuration (résines, dégazage dans le RCV et le pressuriseur, décroissance, capture neutronique).

Les modules 1, 3 et 4 n'ont pas été changés, par contre, la modélisation du module 2 a été améliorée par un calcul plus réaliste de la thermique du combustible, en couplant à PROFIP le code CREOLE (CEA/DMG) qui prend en compte l'évolution du jeu combustible-gaine pendant l'irradiation et la composition chimique du gaz ou de la vapeur d'eau emplissant le jeu.

2) Principaux résultats

Des campagnes de mesure sur les réacteurs ont été effectuées pour caractériser l'état du combustible à partir des mesures d'activité primaire (1), (2), (3), (4).
Les mesures effectuées à TRICASTIN-1 (5) et à BUGEY-5 (6) montrent que les défauts de gainage n'ont pas évolué lors des essais de suivi de charge.
Des études paramétriques ont été effectuées pour déterminer la sensibilité et le domaine de validité des méthodes de détermination du nombre et de la gravité des ruptures de gaine à partir des mesures d'activité primaire (7).

5) PROCHAINES ETAPES

- Le CTRC (Comité Technique Rejet Combustible) proposera en 1983 un terme source, pour le module 2, "relâchement des produits de fission hors de l'UO2", qualifié à partir de l'ensemble des résultats expérimentaux obtenus sur boucles et sur réacteurs.

- Développement du code PROFIP pour traiter les régimes transitoires en particulier le fonctionnement en téléréglage et en suivi de charge.

- Modélisation du comportement des produits de fission solides et des émetteurs alpha.

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6) RELATIONS AVEC D'AUTRES ETUDES

Evaluation du taux de relâchement des produits de fission hors d'une rupture de gaine dans un réacteur à eau sous pression. Fonctionnement normal. Fiche 140-1-02.

7) DOCUMENTS DE REFERENCE

(1) TIHANGE - Cycle 6 - Contamination en produits de fission du circuit primaire - Rapport SEN/82/60.

(2) FESSENHEIM 1 - Cycle 3 - Contamination en produits de fission du circuit primaire. Note technique SEN/ECC/82-084. Mai 1982.

(3) Mesure de la contamination du circuit primaire de la centrale de BUGEY - tranche 2 lors de l'arrêt pour rechargement en juillet 1981 (2ème cycle). Rapport SEN/82-140. Février 1982.

(4) TRICASTIN 1 - Cycle 1 - Activité en produits de fission du circuit primaire - Relâchements des produits de fission volati s. Note technique SEN/ECC/82-076.

(5) TRICASTIN 1 - Cycle 2 - Activité en produits de fission du circuit primaire pendant les essais de suivi de charge. Note technique SEN/ECC/82-085. Juin 1982.

(6) BUGEY 5 - Cycle 3 - Activité en produits de fission du circuit primaire pendant les essais de suivi de charge. Note technique SEN/ECC/82-095. Septembre 1982.

(7) Sensibilité et domaine de validité des méthodes de détermination du nombre et de la gravité des ruptures de gaine à partir des mesures d'activité de l'eau primaire. Note technique SAER N° 82/290. Février 1982.

8) DEGRE DE DISPONIBILITE

Documents internes, non disponibles, sauf accord du CEA.

	CLASSIFICATION: 5.2 5.3 5.4
TITLE (ORIGINAL LANGUAGE): CRITICAL ANALYSIS OF ACCIDENT SCENARIOS AND CONSEQUENCES MODELLING APPLIED TO LWR POWER PLANTS, FOR ACCIDENT CATEGORIES BEYOND THE DESIGN BASIS ACCIDENT	COUNTRY: Italy SPONSOR: CEC
TITLE (ENGLISH LANGUAGE):	ORGANISATION: ENEA PROJECT LEADER: G.PETRANGELI

INITIATED:	COMPLETED: (expected)	SCIENTISTS:
Nov.1981	1982	C.BROFFERIO P.CAGNETTI V.FERRARA A.MANICLIA
STATUS: In progress	LAST UPDATING:	

1. GENERAL AIM

a) To identify, for a limited number of salient hypothetical accident scenarios, relevant items of uncertainty or inadeguate knowledge and their quantitative bearing on the predicted consequences.
b) To evidentiate, thereby, possible shortcomings in present assessment methodologies when applied to large accidents.
c) To point out major areas of inadeguate knowledge and to suggest appropriate research yet to be considered.

2. PARTICULAR OBJECTIVES

For any major hypothetical accident case under scrutinity, the study deals with two consecutive aspects:
a) The release, migration and abatement of activities within the containment boundary
b) The release of unabated activities to the atmosphere, their entrainment and deposition and the resulting consequences as function of time and distance to the source.

3. EXPERIMENTAL FACILITIES AND PROGRAMME

Theoretical study

4. PROJECT STATUS

1) Work in progress. Two accident scenarios are, up-to now, identified: the analysis of point a) and b) (Section n.2) are completed for one

TITLE (ENGLISH LANGUAGE): CRITICAL ANALYSIS OF ACCIDENT SCENARIOS AND CONSEQUENCES MODELLING APPLIED TO LWR POWER PLANTS, FOR ACCIDENT CATEGORIES BEYOND THE DESIGN BASIS	CLASSIFICATION: 5.2 5.3 5.4

of the two accidents identified; the major results are: it is necessary to investigate within some specific areas in which knowledge is inadeguate (building effect, plume rise, etc.)

5. NEXT STEP

To complete the study and produce a technical report.

6. RELATION TO OTHER PROGRAMS

To investigate on some specific aspects described in the Rasmussen report (Wash 1400) and German Risk Study.

7. REFERENCE DOCUMENTS

Benchmark (comparison of models) made by OECD.

8. DEGREE OF AVAILABILITY

To be requested directly to DG XII/D of CEC - Bruxelles.

9. Additional information

Personnel involved: 5 - 6 persons.

5.3. RETENTION

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	5.3	150 204
Vorhaben/Project Title Dosisabbau Dose Reduction		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor KRAFTWERK UNION AG Reaktortechnik R 352, Erlangen
Arbeitsbeginn/Initiated 01.04.76	Arbeitsende/Completed 31.03.83	Leiter des Vorhabens/Project Leader E. Schick
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31.12.82	Bewilligte Mittel/Funds

1. General Aim

Improvement of currently used units and development of advanced techniques for the management of gaseous and liquid radioactive waste, their optimization and the application of both lab. and operational plant experience to technically mature systems.

2. Particular Objectives

2.1 Gas treatment

In order to reduce radiation exposure, the quantities of long-lived nucledes 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.

2.2 Water-soluble radioactivity

The target is the use of an ion exchanger specifically for cesium, increased service lives and minimization of cesium buildup in the primary coolant, all of which will lead to a reduction of personnel exposure. In addition, waste water evaporator distillates from nuclear power plants will be decontaminated with I.T. to achieve optimum waste water purification.

2.3 Decontamination

Decontamination methods will be further developed for application in a nuclear power plant for the reduction of exposure to 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 a universally applicable system. Examination of oxygen and hydrogen measuring apparatus and selection of measurement procedure.

3.2 Water-soluble radioactivity

Examinations of decontamination of primary coolant and waste water evaporator distillates with filters and ion exchangers.

3.3 Decontamination

Decontamination of large-scale tanks in NPPs by special methods; preparation and treatment of decontamination solutions.

4. Experimental Facilities

Tests will be performed at model test facilities available either in Erlangen or in NPPs.

5. Progress to Date

No further adsorption tests have been carried out because no additional funds were available.

6. Results

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7. Next Steps

The project will be completed. The final report will be compiled.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	5.3	150 209
Vorhaben/Project Title Aktivierte Korrosionsprodukte in LWR-Kreisläufen Activated Corrosion Products in LWR Loops		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 45, Erlangen
Arbeitsbeginn/Initiated 01.01.79	Arbeitsende/Completed 30.06.82	Leiter des Vorhabens/Project Leader Dr. Neeb
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating 31.12.82	Bewilligte Mittel/Funds

1. General Aim

Development of realistic contamination models for LWR reactor coolant circuits with the aim of reducing the activity level of the circuits and loops, i.e. to reduce irradiation exposure of the control and maintenance personnel in nuclear power plants.

2. Particular Objectives

Improvement and completion of our knowledge of the sources, formation mechanisms and transportation and deposition behaviour of those radionuclides which are primarily responsible for contamination of circuits and systems of LWRs, i.e. for the local dose rates in the plant as well as for the activity inventory of radioactive wastes.

3. Research Program

3.1 Compilation and evaluation of operating data for PWRs and BWRs.

3.2 Data balancing in order to identify radionuclide sources.

3.3 Evaluations 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 operational parameters.

3.5.2 Exchange behaviour of deposits and protective films.

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 materials with a high cobalt content in a neutron field.

3. Research Program

3.1 Gas treatment

Laboratory test and evaluation of a noble gas separating test facility and development of a universally applicable system. Examination of oxygen and hydrogen measuring apparatus and selection of measurement procedure.

3.2 Water-soluble radioactivity

Examinations of decontamination of primary coolant and waste water evaporator distillates with filters and ion exchangers.

3.3 Decontamination

Decontamination of large-scale tanks in NPPs by special methods; preparation and treatment of decontamination solutions.

4. Experimental Facilities

Tests will be performed at model test facilities available either in Erlangen or in NPPs.

5. Progress to Date

No further adsorption tests have been carried out because no additional funds were available.

6. Results

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7. Next Steps

The project will be completed. The final report will be compiled.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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3.7 List of PWR and BWR contamination models.

3.8 Direct nuclide related activity measurement.

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 samples for analysis (system and fuel assembly deposits) and collection from a number of data points are closely tied in with reactor refuelling shutdowns.

5. Progress to Date

A mobile measurement unit intended for non-destructive direct measurement of nuclide-related activity concentration levels in piping deposits has been developed and used at GKN and KKI. The measuring unit consists of: a germanium radiation detector, a collimator, a lead shield, an analog electronics system with pile-up suppression and pulse inserting type dead-time compensation, and a front-end computer with a magnetic tape unit for mass storage. The spectra recorded have been evaluated on a central computer in the laboratory using newly developed software which allowed for exact dead-time and local dose rate compensation. The measuring unit has been calibrated against piping sections of typical diameters and wall thicknesses with line sources of selected nuclides attached on the inside; rotation of the pipes simulates a uniform distribution of activity on the inner surface of the pipe.

6. Results

The measuring system developed and the method of evaluation applied each proved their suitability in two cycles of measurements at GKN and KKI.

7. Next Steps

Work has been completed. The final report is being compiled.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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8. Relation with Other Projects

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9. References

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10. Degree of Availability

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	5.3	150 403
Vorhaben/Project Title Reduzierung der Strahlenbelastung Teilvorhaben 1: Verminderung des Aktivitäts-aufbaus Reduction of Radiation Exposure Part 1: Reduction of the Build-up Rate of Activity		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor KRAFTWERK UNION AG Reaktortechnik R 451, Erlangen
Arbeitsbeginn/Initiated 15.08.79	Arbeitsende/Completed 31.12.82	Leiter des Vorhabens/Project Leader Dr. E. Schuster
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating 31.12.82	Bewilligte Mittel/Funds

1. General Aim

System contamination shall be reduced by lowering the amount of activated corrosion products in the cooling systems loops of LWR plants thus lowering the exposure of personnel to radiation during maintenance and inspection procedures.

2. Particular Objectives

The project objective "The reduction of the build-up rate of activity" shall be performed within this project in two different ways:

2.1 As a supplement to project RS 209 knowledge on the formation and activation of corrosion products shall be extended with the aim of finding measures to reduce the amount of corrosion products in the coolant and to collect basic data for the dimensioning of high temperature filters.

2.2 Operating parameters shall be determined in parallel to 2.1 for an electromagnetic filter as a high-temperature filter for the removal of corrosion products from the coolant.

3. Research Program

3.1 Formation and transport of corrosion products

3.2 Investigation of deposition of corrosion products on the surface of the fuel rods of light water reactors

3.3 A study of the use of an electromagnetic filter for purification of reactor coolant in a nuclear power station

4. Experimental Facilities

Measurements on nuclear power plants.

5. Progress to Date

Re. 3.1 The operational evaluations which were already under way have been continued. The pH value at the operating temperature is one significant parameter in the formation and transport of corrosion products. As the pH value at the operating temperature cannot be measured, reference material was studied to produce a realstic model to calculate the pH value.

Re. 3.2 Further crud samples have been taken from the Isar nuclear power plant. On account of their high dose rate, these samples had to be opened and treated in a semi-hot cell.
In the Gösgen nuclear power plant (Switzerland), crud samples have for the first time been taken from PWR fuel rods with the aid of the brushing equipment. Atomic absorption spectroscopy has been utilized to determine the Fe, Co, Ni, Cr, Mn, Cu and Zn content of all the samples, and their activity content in respect of Cr-51, Mn-54, Co-58, Co-60, Fe-59 and Zn-65 has been measured by γ-spectrometry.

Re. 3.3 A study has been made of various high-temperature filtration devices and several possible locations within the primary system. In addition, two different methods of calculating the effectiveness of high-temperature filtration have been compared.

6. Results

Re. 3.1 Mesmer's model was preferred for pH calculations. The pH values as a function of boric acid and lithium concentration are indicated in Figs. 1 and 2.

Re. 3.2 To date, the chemical and radiochemical examinations of samples from the Isar NPP have yielded information on surface density, activity levels and the element-specific activity of ^{60}Co. In the case of Isar NPP, the X-ray microstructure investigations indicated that haematite with particle sizes between 0.5 and 3 µm is the only constituent present in a crystalline phase.
In the case of Gösgen, the investigation showed that αZr and ZrO_2 were present almost exclusively.
The content levels detected in the samples are as follows:

Elementcontents in %

NPPs	Fe	Co	Ni	Cr	Mn	Cu	Zn
Isar	75	0.1	0.3-0.6	0.2-0.6	0.1-0.3	0.1	0.1-2.9
Gösgen	0,8-8,5	0,5	1,0	2.1-0.1	0.1	0.2	0.5

The Fe contents of the samples from Isar NPP only fluctuate slightly about the stated value; where the differences in the individual samples are greater, the minimum and maximum percentages are given.

Re. 3.3 The high-temperature filter should have a flow rate equal to at least 0.5 % of the primary system flow rate and should be incorporated in the bypass to the reactor coolant pumps. The filters which are suited to filtering activated corrosion products under PWR primary system conditions are the electro-magnetic filter, the high-gradient magnetic separator and filter cartridges with a bonded metal fiber matrix. The radiation level on primary system components should be reduced by a factor of 2 at least. If there should be a requirement that the filter must be of the reverse pulse type, only the electro-magnetic filter is suitable for use in a PWR plant at the present state of the art.

7. Next Steps

Work has been completed. A final report will be compiled.

8. Relation with Other Projects

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9. References

-

10. Degree of Availability

-

pH values of borates according to Mesmer t=300 C

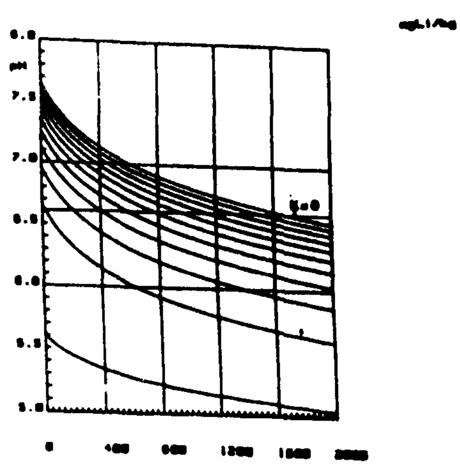

Fig. 1 Concentration in mgBor/kg

pH value curve at 25 °C as a function of borate concentration and pH value at 300 °C

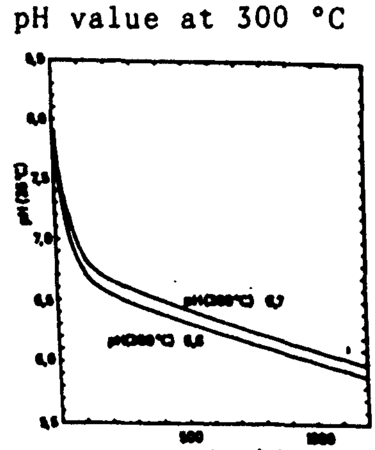

Fig. 2

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	5.3	150 404
Vorhaben/Project Title Reduzierung der Strahlenbelastung Teilvorhaben 3: Dosiseinsparende Arbeitsmethode für Reparaturen an Reaktorkomponenten Reduction of Radiation Exposure Part 3: Dose Reducing Work Procedures for Repairs to Reactor Components		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor KRAFTWERK UNION AG Reaktortechnik R 82, Erlangen
Arbeitsbeginn/Initiated 15.08.79	Arbeitsende/Completed 31.12.82	Leiter des Vorhabens/Project Leader R. Weber
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating 31.12.82	Bewilligte Mittel/Funds

1. General Aim

The increasing use of nuclear energy is also accoumpanied by an increase in expenditure for inspection and repair time and thus an increase in the radiation dose for such work procedures.

2. Particular Objectives

In order to counteract the tendency to increase the number of persons exposed to radiation and in order to reduce the radiation dose for certain work procedures, work methods and equipment shall be developed which either accelerate the work procedures or enable personnel to remain at a greater distance from the components undergoing repair.

3. Research Program

3.1 Application study on ultrasonic examination methods

3.2 Study of the replacement or repair of primary circuit components

3.3 Repair methods for RPV sealing surfaces

3.4 Plasma-cutting in large depths of water.

4. Experimental Facilities

Some of the experimental facilites for RS 2703 will be used.

5. Progress to Date

Re. 3.1 The method of disassembly has been investigated in respect of both components. There are two possibilities:

- removal of the whole component, tilting, transfer from the containment to a store on site

- breaking down the components, packing the sections into containers, transfer of the containers through the equipment hatch to the store (ultimate storage facility).

Re. 3.3 In addition, the pros and cons of dry breakdown and wet breakdown (under water) have been examined.
The viable cutting methods have been investigated.

6. Results

Work on disassembly of the two components leads to the following conclusions:

Re. 3.1 Reactor pressure vessel:

Although the removal and outward transfer of the whole component is an attractive option in view of the time saved thereby, radiation protection considerations weigh against it, as very extensive shielding measures must be taken to keep the exposure of personnel and the environment to radiation as low as possible. Appropriate shielding on the vessel is heavy and causes overloading of the reactor building crane, shields along the route from the containment to the store are bulky and must be discounted on grounds of economic viability. Neither of these courses seems very sensible and neither has been pursued further. Breaking down the component on site, packing the sections into containers and transferring the containers through the equipment hatch is a feasible solution. In adopting this option, it is expedient to exploit the good shielding effect of water by breaking down the component under water. The reactor cavity is flooded and the reactor pressure vessel broken down under remote control.

Steam generator:

The surface activity level of the steam generator is considerably lower than that of the reactor pressure vessel. Removal and outward transfer of the whole component is thus possible. This reduces erection time and hence power plant downtime.

Re. 3.3 The viable cutting methods for breaking down the reactor pressure vessel have been investigated. The following cutting methods are suitable:

- arc cutting
- flame cutting
- milling

7. Next Steps

Work has been completed. A final report will be compiled.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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Berichtszeitraum/Period 01.01.82 - 31.12.82	Klassifikation/Classification 5.3	Kennzeichen/Project Number 06.01.11/21A (PNS 4311)
Vorhaben/Project Title Untersuchungen zur Wechselwirkung von Spaltprodukten und Aerosolen in LWR-Containments Investigations on the interactions of fission products and aerosols in LWR-Containment		Land/Country Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe GmbH (KfK) Projekt Nukleare Sicherheit (PNS)-LAF I
Arbeitsbeginn/Initiated 01.01.77	Arbeitsende/Completed 31.12.1985	Leiter des Vorhabens/Project Leader Dr. W. Schöck
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 22.01.1983	Bewilligte Mittel/Funds

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 is being developed on the basis of an experimentally verified numerical model, and large scale aerosol behaviour experiments are conducted.

3. Research Program

3.1 Development of the NAUA model to describe the aerosol behaviour.

3.2 Experimental investigations of aerosol behaviour in steam atmospheres

3.3 Verification of the model and extrapolation to real containment systems

3.4 Large scale experiments to demonstrate the model

4. Experimental Facilities, Computer Codes

The NAUA facility consists of a 3 m^3 thermostated vessel with an operating temperature range from 20 to 150 °C in a saturated steam atmosphere. Peripheral instrumentation includes aerosol sources, steam generation and all necessary particles measurement devices. The large scale experiments will be conducted in the model containment at Battelle, Frankfurt which is a model of Biblis B in a 1 : 4 scale.

The computer code NAUA (current version Mod4) calculates the aerosol removal processes sedimentation, thermophoresis and diffusion, the coagulation and the steam condensation onto the particles by use of the significant thermodynamic functions

- 2 -

for realistic PWR core melt down sequences.

5. Progress to date

The code version NAUA Mod4 was released to the USNRC and to the EPRI. A workshop was performed at EPRI, Palo Alto on March, 29-30, 1982 to introduce the code to the potential users in the USA.

The calculations of aerosol transport and removal during a low pressure core melt accident with release category FK2 have been completed. Past test calculations of NSPP aerosol experiments have been performed.

The planning of the large scale experiment DEMONA (Demonstration Experiment for Modeling of Nuclear Aerosols) was continued. The experiments will be conducted in the period 1983-1985 in the Model Containment at Battelle Frankfurt. Participants will be KfK/LAF I, Battelle Frankfurt, EIR Würenlingen and KWU Erlangen. Besides KfK, BMFT and EIR will contribute to the funding of the project. The preparation of the experiment has been started.

The pre-tests of the aerosol instrumentation for the BETA experiment were completed.

6. Results

The aerosol removal calculations for the release category FK2 (following the German Risk Study) have been completed. In the calculations not only the containment but additionally the annulus and the auxiliary building have been modeled. Reductions factors of approximately 5 in the containment, 3 in the annulus, and 10 in the auxiliary building were obtained for the released aerosols. This gives a total reduction by a factor of 140 calculated with NAUA as compared to a factor of 4 in the German Risk Study.

Past test calculations have been performed for experiments carried out in the NSPP vessel at ORNL. In the NSPP 400 series of experiments the behaviour of UO_2 aerosols is measured in saturated steam NAUA-Mod 4 underestimated the removal rates considerably, which shows that an additional deposition mechanisms must have been active during the experiment. Such a mechanism could be diffusiophoretic deposition due to large flows of condensing steam to the vessel walls during the experiments. Including a preliminary model for diffusiophoresis in NAUA an excellent agreement between measured and calculated aerosol concentrations could be obtained. The modeling of diffusiophoresis will thus be pursued with high priority.

The planning of the DEMONA project has been intensified and experimental preparations have been initiated. The time schedule contains a number of seven tests in the period from late 1983 to the end of 1984. Using a basic set of thermodynmic and aerosol conditions as in a late overpressure core melt accident (release category FK6), variations of aerosol and steam source rates and also of the geometric structure of the Model Containment are planned. Non-radioactive mixed aerosols will be used, composed of up to four different metal oxides, which are generated by vaporizing and oxidizing metal powders in a plasma flame. Time dependent measurements will be made of all relevant thermodynamic data and aerosol parameters as generation rates, airborne and depleted masses, size distributions of both solid and liquid components.

Scaping calculations with NAUA-Mod4 have been performed to identify the required aerosol generation rates and to quantify the influence of the containment leakage on the accuracy of aerosol measurements. A test rig was constructed in which optimal operating conditions for the aerosol generator and feed lines will be experimentally tested. Further experimental work was conducted to test an optical mass monitor for airborne aerosol concentrations.
The preparatory acitivites of EIR, BF and KWU are reported elsewhere.

The tests of aerosol measurement methods for the BETA project have been completed. The apparatus will be constructed in 1983.

7. Next Steps

The aerosol generator tests will be conducted in early 1983. After construction of the aerosol measuring devices operational testing and calibration will be done in the NAUA vessel.
A thermodynamic test run of the Model Containment is scheduled in 1983, after which the installations of the instrumentation will take place. The facility should be ready for operation by the end of 1983.

8. Relation to other Projects

PNS 4315, PNS 4331; Cooperation with BF, EIR, KWU in the DEMONA project

9. References

W. Schöck, H. Bunz: Best estimate calculations of fission product release to the environment for some PWR core melt accident sequences, Int. Meeting on Thermal Nuclear Reactor Safety, Chicago, 1983
H. Bunz, W. Schöck: Transport und Rückhaltung von Aerosolen im Containment bei Kernschmelzunfällen, Fachtagung Reaktorsicherheit der KTG, Karlsruhe, Juni 1982

10. Availability of Reports

Unrestricted distribution

<table>
<tr><td>Berichtszeitraum/Period
01.01.82 - 31.12.82</td><td>Klassifikation/Classification
5.3</td><td>Kennzeichen/Project Number
06.01.14/13A (PNS 4414)</td></tr>
<tr><td colspan="2" rowspan="3">Vorhaben/Project Title
Bestimmung der Iodkomponenten in der Abluft von Siedewasserreaktoren

Determination of the Iodine Species in the Exhaust Air of Boiling Water Reactors</td><td>Land/Country
-</td></tr>
<tr><td>Fördernde Institution/Sponsor
BMFT / BMI</td></tr>
<tr><td>Auftragnehmer/Zuwendgsempf./Contractor
Kernforschungszentrum Karlsruhe

Projekt Nukleare Sicherheit (PNS) - LAF II</td></tr>
<tr><td>Arbeitsbeginn/Initiated
01.01.1980</td><td>Arbeitsende/Completed
31.12.1983</td><td>Leiter des Vorhabens/Project Leader
H. Deuber</td></tr>
<tr><td>Stand der Arbeiten/Status
in progress</td><td>Berichtsdatum/Last Updating
December 1982</td><td>Bewilligte Mittel/Funds</td></tr>
</table>

1. General Aim

Improvement of the assessment of the environmental impact of radioiodine released with the exhaust air of boiling water reactors; improvement of the ventilation concept of boiling water reactors (BWRs).

2. Particular Objectives

Determination of the radioiodine species elemental iodine (I_2), particulate iodine, and organic iodine (CH_3I) in the exhaust air of BWRs. (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

Operation of radioiodine species samplers in the exhaust air of BWRs.

4. Experimental Facilities

Rigs for the operation of radioiodine species samplers in the exhaust air of BWRs.

5. Progress to Date

Continuous measurements in a BWR (BWR 4) with radioiodine species samplers in the stack exhaust and in the various exhausts within the plant contributing to the stack exhaust.

6. Results

The main results of the ^{131}I measurements in the stack exhaust and in the various exhausts within the plant over a period of 12 months are as follows:

a) In the stack exhaust the fraction of the radiologically decisive elemental ^{131}I was 45 % on an average. The remainder consisted nearly completely of organic ^{131}I.

b) The elemental ^{131}I released with the stack exhaust was largely contributed by the unfiltered reactor building exhaust (throughput ca. 110 000 m^3/h). An other important source of the elemental ^{131}I released to the environment was the unfiltered turbine building exhaust (throughput ca. 120 000 m^3/h).

c) The potentially high release of elemental ^{131}I with the purge air was strongly reduced by iodine filtration (DF $> 10^3$).

7. Next Steps

Performance and termination of corresponding measurements in an other boiling water reactor.

8. Relation with Other Projects

-

9. References

H. Deuber: Die physikalisch-chemischen ^{131}I-Komponenten in der Abluft eines Siedewasserreaktors (SWR 4), KfK 3424 (1982).

10. Degree of Availability of the Reports

KfK literature department.

Berichtszeitraum/Period 01.01.83 - 31.12.83	Klassifikation/Classification 5.3	Kennzeichen/Project Number 06.01.14/14A (PNS 4415)
Vorhaben/Project Title Development and Improvement of Exhaust Air Filters for Accident Conditions Entwicklung und Verbesserung von Abluftfiltern für Störfallbedingungen.		Land/Country Fördernde Institution/Sponsor Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated 01.01.1978	Arbeitsende/Completed 31.12.1983	Leiter des Vorhabens/Project Leader DI Dillmann
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1981	Bewilligte Mittel/Funds

1. General Aim

Improvement of ventilation systems for operation under accident conditions in order to increase environmental protection around LWR nuclear power stations.

2. Particular Objectives

2.1 Specification of performance requirements for HEPA filters for operation under accident conditions.

2.2 Upgrading and, if applicable, new development of particulate air filter elements in conformity with 2.1.

3. Research Programme

3.1 Definition and analysis of accidents eventually endangering ventilation systems and estimation of the load parameters to be expected.

3.2 Establishment of a performance specification for HEPA filter elements to be used under accident conditions.

4. Experimental Facilities

Laboratory test facilities, technical test bench (TAIFUN) for measurement of the removal efficiency of full-scale particulate air filter elements under high temperatures and high humidities. A Testrig ist build for measuring the dust holding capacity of filter elements.

5. Work Performed

Ad 3.1 Since at present the input data for the NAUA code to be applied to core melt accidents are discussed, the computations have been discontinued for the time being . However, it is evident that the challenges on an exhaust air filter system will become lower than the values used in previous computations. Since the conditions resulting from first computations are considered to be perfectly controllable by a post-accident exhaust air filter system, reserves are still available in the development potential.

Ad 3.2 The investigations into metal fiber filters were continued. On 2 µm fiber filters measurements were performed of the removal efficiency under normal conditions, at elevated temperature and elevated pressure with varying face velocities. A first high loading test was performed.

To investigate the dust holding capacity a measurement technology had been ordered which will be upgraded on the basis of the first tests and allow to make measurements of the removal efficiencies on different prefilters, specific to particle sizes. This is to allow a faster evaluation (without SEM pictures) and optimization of the prefilters.

6. Results Obtained

Ad 3.1 -

AD 3.2 When 2 µm fibers were used, decontamination factors > 10^5 were achieved with face velocities up to 40 cm/s. The tests at elevated temperature were carried out up to 200 ^oC. With increasing temperature an increase in the removal efficiency was found, namely by a factor 5 upon transition from room temperature to 200 ^oC. The influence of the humidity of the air was studied up to 95 % RH. No significant influence of the humidity of the air was found. The influence of pressure on the removal behavior was studied within the range of 1-5 bar. For the face velocity values between 30 and 40 cm/s were chosen. With increasing pressure the removal efficiency decreases while the other conditions remain the same. First tests performed on highly loadable all-metal prefilters were started. Extremely high values of loading of approx. 5 kg/m^2 filter surface were attained for a particle size between 1 and 10 µm ($\eta \geq 97$ %).

7. Plans for Future Work

Ad 3.1 Work will be continued with new input data to the extent they are derived from other working programs.

Ad 3.2 The test of the metal fiber filters will be carried on under variation of the combination of pressure, temperature and relative humidity. The investigations in the efficiency of prefilters, specific to particle sizes and dust holding capacity, will be continued.

8. Relation with Other Projects

SR 0148, PNS 06.01.14/15A

9. Literature

17th DOE Nuclear Air Cleaning Conference

H.-G. Dillmann, H. Pasler;

Experimental Investigations of Aerosol Filtration with Deep Bed Fiber Filters .

10. Availability of Reports

–

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	5.3	06.01.14/15A (PNS 4416)
Vorhaben/Project Title Investigation into the behaviour of HEPA filters at high temperature, air humidity and elevated differential pressure. Studium des Verhaltens von Schwebstoffiltern unter hoher Temperatur, Luftfeuchte und erhöhtem Differenzdruck.		Land/Country Fördernde Institution/Sponsor Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.1981	31.12.1985	Dr. V. Rüdinger
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

Improvement of filter systems in view of accident situations.

2. Particular Objectives

2.1 Investigations into the response of HEPA filters under the challenges of high temperature, high humidity and high differential pressure.

2.2 Upgrading and new development of particulate air filters.

3. Research Programme

3.1 Testing the filtering efficiency at high temperatures including the development of a new suitable measurement procedure.

3.2 Improvement of humidity measurement and investigation into the flow-resistance characteristics of HEPA filters at high humidity.

3.3 Testing the structural response of HEPA filters at static pressure differentials under room air conditions.

3.4 Planning and build-up an installation for testing HEPA filters under the combined challange of elevated differential pressure, high rel. humidity and/or high temperature.

4. Experimental Facilities

Laboratory test facilities, technical test bench (TAIFUN) for measurement of the removal efficiency of full-scale particulate air filter elements at high temperatures and high humidities. Expansion of the testing facilities so as to allow examination of the mechanical stability at high differential pressure.

5. Work Performed

Ad 3.1 With respect to the determination of the aerosol mass concentration an instrument which provides flameless atomic adsorption spectroscopy was obtained and tested. A method for the solubilization of the aerosol substance deposited on the nuclepore filters was worked out.

Ad 3.2 Investigations into the response of HEPA filters to high humidity, in the range of 60 % RH and also under exposure to fog conditions, both at 50 ^{o}C was condinued. One new HEPA filter and two filters, respectively, preloaded with ambient dust up to 700 and 800 Pa pressure drop at rated flow, were tested during time periods between 150 and 250 h.

Ad. 3.3 A third, extensive, program to further investigate the response of HEPA filters to high differential pressure was carried out at the Los Alamos National Laboratory (LANL) test facility. It included about 70 structural tests and an additional 25 tests to determine filter flow resistance characteristics.

Ad. 3.4 The design of the planned facility BORA was carefully revised in order to extend flexibility of operation and to reduce equipment costs.

6. Results Obtained

Ad 3.1 Concentrations of the chosen thermally stable aerosol substances down to 0.1 μg/ml can be measured reliably enough to allow the determination of decontamination factors greater than 10^3.

Ad 3.2 Under 100 % rel. humidity the filter resistance at rated flow increased up to 1000 Pa with the new filter and up to between 1400 and 3500 Pa with the preloaded filters. Operation under fog conditions further increased the pressure drop and resulted in structural failure at 3.3 kPa differential pressure with the new metal frame high temperature filter and at 6.3 and 4.7 kPa with the preloaded filters with wooden frames and an elastomeric sealant.

Ad 3.3 The results obtained for the structural limits and the flow resistance characteristics of the HEPA filters tested, are consistent with earlier results. The structural limits of two types of modified conventionally pleated filters with wooden frames were found to exceed 27 kPa.

Ad 3.4 Significant modifications in the design of the test facility BORA resulted in an extention of the range of operating conditions as well as in a considerable reduction in the equipment costs. The Contracts for the construction of the test facility were negotiated and signed.

7. Plans for Future Work

Ad 3.1 The new method for the determination of filter efficiency will be tested under ambient conditions and checked against standard procedures.

Ad 3.2 The investigations into the flow resistance characteristics of HEPA filters at high humidities will be persued.

Ad 3.3 The results of the 3rd series of filter tests at the LANL-facility will be evaluated and documented. One more series will be carried out at the LANL-facility to conclude the investigations of currently standard commercial filters under air conditions of ambient temperature and humidity.

Ad 3.4 The construction of the facility BORA will be concluded by the end of 1983. The system for the humidification will be designed and built. The R+D-instrumentation and the data acquisition system will assembled, and tested.

8. Relation with other Projects

PNS 06.01.14/14A, SR 0148/1, SR 290

9. Literature

-

10. Availability of Reports

-

		CLASSIFICATION : S.3
TITLE (ORIGINAL LANGUAGE) : Trattamento dei gas nobili radioattivi		COUNTRY : ITALY
		SPONSOR : CNR-ENEA-MPI
TITLE (ENGLISH LANGUAGE) : Radioactive noble gases treatment		ORGANISATION : University of Pisa
		PROJECT LEADER : Giorgio CURZIO
INITIATED : July 1970	COMPLETED :	SCIENTISTS : F. CASTELLANI, A. GENTILI, L.PIEVE, S. VAGLINI
STATUS : in progress	LAST UPDATING : May 1982	

1. General aim

Theoretical and experimental research on the general problems involved in the production, release, treatment, storage and measurements of radioactive 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 and full scale

2.1.2. Acceptance and periodical tests of delay beds in laboratory and industrial scale.

2.2 Decontamination of the reactor off-gas by means of cryogenic techniques and other methods.

2.3 Treatment of the reprocessing plant off-gas.

2.3.1 Cryogenic distillation.

2.3.2 Absorption in organic solvent.

2.3.3 Experimental determination of the concentration of nobles gases and other gaseous radwaste in the off-gas of a reprocessing plant.

2.4 Storage of Kr^{85}.

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TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Radioactive noble gases treatment	5.3

2.4.1 Storage by bottling (eventually in containers filled with adsorbent media).

2.4.2 Storage by trapping in solid matrixes

2.5 Cost-benefit analysis

2.5.1 Cost-benefit analysis of nuclear reactors off-gas treatment systems.

2.5.2 Cost-benefit analysis of reprocessing plants off-gas treatment systems.

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.

2.7 Radon adsorption on charcoal beds.

2.7.1 Adsorption charachteristics in laboratory scale

2.7.2 Design of industrial scale adsorption devices.

3. Experimental facilities
   a) Charcoal bed testing facility in dynamic conditions
   b) Nuclear detection devices
   c) Granular charcoal testing facilities in dynamic and stating adsorption conditions.
   d) Facilities for off-gas characterization.

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
   Item 2.6 is in program and item 2.7 is in development and will be completed at the end of the 1983.

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. F. CASTELLANI, G. CURZIO, A.GENTILI:"Un problema di conduzione del calore in un mezzo cilindrico omogeneo con sorgente in movimento" La Termotecnica, Vol XXXI, 3 (1977)

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Radioactive noble gases treatment	5.3

2. F. CASTELLANI, G. CURZIO. A. GENTILI:"Facilities for Conditioning Noble Gases produced by Nuclear Power Reactors in West Europe" Kerntechnik 19(1), (1977)

3. F.CASTELLANI, G. CURZIO, A. GENTILI :"Analisi comparativa tra i sistemi di trattamento dei gas nobili radioattivi prodotti negli impianti Nucleari"
Atti Istituto di Impianti Nucleari , Università di Pisa, RP 280(77)

4. F. CASTELLANI, G. CURZIO, A. GENTILI :"Stoccaggio di gas nobili radioattivi in bombole o contenenti carbone attivo: valutazione dei costi".
Atti Istituto di Impianti Nucleari, Università di Pisa, RP 283(77)

5. F. CASTELLANI, G. CURZIO, A. GENTILI :"Metodologia di collaudo di un sistema di trattamento per il rilascio ritardato di gas nobili radioattivi prodotti negli impianti nucleari".
Atti Istituto di Impianti Nucleari, Univosità di Pisa, RP284(77)

6. F. CASTELLANI, G. CURZIO, A. GENTILI :"Collaudo del sistema di ritenzione dei gas nobili radioattivi della Centrale ENEL -IV di Caorso".
Atti Istituto di Impianti Nucleari, Università di Pisa, RP285(77)

7. F. CASTELLANI, G.CURZIO, A. GENTILI:" Metodi di controllo dell'efficienza di un sistema a fuoriuscita ritardata dei gas nobili radioattivi prodotti per fissione"
Atti Istituto di Impianti Nucleari, Università di Pisa, RP295(77)

8. F. CASTELLANI, G. CURZIO, A. GENTILI :"Caratterizzazione di materiali adsorbenti di gas nobili radioattivi"
Atti Istituto di Impianti Nucleari, Università di Pisa, RP 310(78)
RP 315 (78), RP 334(78).

9. F CASTELLANI, G. CURZIO, A. GENTILI: "Le piègeage du Kr^{85} rejectè par lés usines de retraitement de combustible".
Seminar on radioactive Effluents from Nuclear Fuel Reprocessing Plants Karksruhe 22- 25 nov. 1977.

10. F. CASTELLANI, G. CURZIO, A. GENTILI :"Caorso off-gas system: acceptance test of the noble gases delay charcoal beds".
5th Technical Meeting, Nuclex 1878, Basel 3-7 Ottobre 1978.

11. F. CASTELLANI. G. CURZIO : Concentrazione di Kr^{85} nelle bombole commerciali di Krypton"
Atti Istituto di Impianti Nucleari, Università di Pisa, RP 362(79)

12. G. CURZIO, P.P.MIQUEL, R.D.PENZHORN: "Krypton ^{85} management"
First European Community Conference on Radioactive Managementand

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Radioactive noble gases treatment	5.3

disposal.
Luxembourg, 20-23 May 1980

13. F. CASTELLANI, G. CURZIO, A. GENTILI, A. LEUDET, P.P.MIQUEL: "Etude comparative des strategies de gestion du Krypton 85." Rapporto finale in corso di stampa come Rapporto.

14. F. CASTELLANI, G.CURZIO, A. GENTILI, L. PIEVE:"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 irraggiato in reattore tipo CANDU".
EUR 7934. In corso di pubblicazione.

15. F. CASTELLANI, G. CURZIO, A. GENTILI :"Effetti dell'irraggiamento sulle caratteristiche di adsorbimento del Kr su carbone attivo"
Atti Istituto di Impianti Nucleari RP 479(81)

16. F. CASTELLANI, G. CURZIO, A. GENTILI :" Effetti dell'impiego prolungato sulle caratteristiche di adsorbimento del Kr su carbone attivo"
Atti Istituto di Impianti Nucleari RL 496(81)

17. G. CURZIO, G.ROSSI :"Messa a punto di un dispositivo per la determinazione del carico di rottura di materiali granulari"
Atti Istituto di Impianti Nucleari RL 497(81)

18. F. CASTELLANI, G. CURZIO, L.PIEVE :"Determinazione delle caratteristiche di adsorbimento del Rn su carboni attivi:realizzazione del sistema di misura".
Atti Istituto di Impianti Nucleari RL 499(81)

19. F. CASTELLANI, G. CURZIO, A. GENTILI :" Analisi di un sistema di ritardo a bassa temperatura dei gas nobili radioattivi prodotti per fissione"
Atti Istituto di Impianti Nucleari RL 500(81)

8. Degree of availability
Availability of reports related to the work in progress is subjected to the authorization of the sponsoring organization.

9. Additional information
Budget foreseen: about 50 millions Lit, without considering salaries.
Technicians: one, part-time

<table>
<tr><td colspan="2"></td><td>CLASSIFICATION:
5.3</td></tr>
<tr><td colspan="2">TITLE (ORIGINAL LANGUAGE):
IL CONTROLLO DEI SISTEMI FILTRANTI INSTALLATI NEGLI IMPIANTI NUCLEARI PER LA RIMOZIONE DELLE PARTICELLE E DELLO IODIO</td><td>COUNTRY:
ITALY

SPONSOR: ENEA
(formerly CNEN)</td></tr>
<tr><td colspan="2">TITLE (ENGLISH LANGUAGE):
Testing of the filter systems used in nuclear plants for particle and iodine removal</td><td>ORGANISATION:
University of Pisa

PROJECT LEADER:
Salvatore LANZA</td></tr>
<tr><td>INITIATED:
end of 1967</td><td>COMPLETED:</td><td rowspan="2">SCIENTISTS:
Iliano CIUCCI
Antonio CUCCURU (°)
Giovanni LOMBARDI (°°)</td></tr>
<tr><td>STATUS:
in progress</td><td>LAST UPDATING:
May 1982</td></tr>
</table>

(°) A. CUCCURU of CAMEN
(°°) G. LOMBARDI of ENEA

1) General aim
Testing of the efficiency of HEPA and iodine filters both in Laboratory and in situ, with reference to standard or accident conditions.

2) Particular objectives
   a) Comparison of the iodine and freon methods, used for testing charcoal filters in situ.
   b) Testing of the efficiency of materials used in nuclear plants for iodine removal in strictly controlled conditions of temperature, velocity and relative humidity of gas stream.
   c) Efficiency determination of two HEPA filters in cascade.
   d) Testing of HEPA filters under heavy environmental conditions (high temperature and relative humidity, overflow), with different particle size distribution.

3) Experimental facilities and program
Facilities:
- a rig operated below the atmospheric pressure and related instrumentation for testing HEPA filters by NaCl, DOP, uranine and condensation nuclei methods;
- a rig and the related instrumentation for testing commercial charcoal filters by Freon and iodine methods;
- an apparatus to perform iodine efficiency tests on granular beds.

Program:
   a) to perform some sets of tests on charcoal filters to try to relate freon and methyl iodide efficiency;
   b) to improve the performance of the apparatus for iodine tests on granular beds so that H.T. and P. tests can be easily run;
   c) 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
   d) the apparatus related to the tests mentioned in point 2.d.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Testing of the filter systems used in nuclear plants for particle and iodine removal.	5.3

4) Project Status

With reference to the point 2

a) A new rig and the related instrumentation has been assembled and preliminary tests were performed.

b) The apparatus has been qualified to perform iodine tests at atmosphere pressure, temperature ranging from room to 150°C, velocity up to 50 cm/sec, water vapour content up to 0.1 g H_2O/g dry air.

c) This item is in standby.

d) Project of the apparatus is under preparation.

5) Next Steps

- To run tests of item 3a
- To improve the apparatus of point 4b
- To design the apparatus of item 2d
- To improve uranine testing method for the objective 2c

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 of 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 Seminary in High Efficiency Aerosol Filtration sponsored by CCE at Aix-en-Provence (F), 22-25 November 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'AIFSPR, Pisa 28-29 ott.1976

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Testing of the filter systems used in nuclear plants for particle and iodine removal.	5.3

5. LANZA S., MAZZINI M., PISANI U.
"L'essai en situ et en laboratoire des filtres a iode en Italie"
IAEA SM-245/45, paper presented at the International Symposium on management of gaseous wastes from nuclear facilities. Vienna 18-22 Febr. 1980

6. CIUCCI I, CUCCURU A., LANZA S.
"Experimental comparison of halogenated hydrocarbons and CH_3I traced by I-131 as test agents for impregnated charcoal beds".
Paper presented at " Seminar on iodine removal from gaseous effluents in the nuclear industry" CEC, Mol 21-24 September 1981.

8. Degree of availability

References are free. Please, contact: S. LANZA -Dipartimento di Costruzioni Meccaniche e Nucleari. Via Diotisalvi, 2-56100 PISA, ITALY

9. Additional information

For one year the budget will be about Lire 100 Millions without any consideration of salaries. Two laboratory operators are involved in addition to scientists.

TITLE (ORIGINAL LANGUAGE) : VALUTAZIONE DEL COMPORTAMENTO IN CONDIZIONI INCIDENTALI DEI SISTEMI FILTRANTI PER LA RIMOZIONE DELLE PARTICELLE, INSTALLATI NEGLI IMPIANTI NUCLEARI.		CLASSIFICATION : 5.3
		COUNTRY : ITALY
		SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : EVALUATION OF HEPA FILTER SYSTEMS, USED IN NUCLEAR PLANTS, UNDER HEAVY ENVIRONMENTAL CONDITIONS.		ORGANISATION : ENEA
		PROJECT LEADER : G. Beone, G.Caropreso
INITIATED : 1981	COMPLETED :	SCIENTISTS : E. Neri
STATUS : In progress	LAST UPDATING : MARCH 1982	

1. General aim

Testing of the performance (efficiency and pressure drop) of HEPA filters under accident conditions.

2. Particular objectives

a) Filter testing at high temperatures (up to 500°C).
b) Filter performance in presence of high relative humidity (up to 100% at 96°C).
c) Clogging test of filters, using smokes generated by the combustion of rubber, wood, plastic materials.
d) Shock overpressure through the filter banks (up to pressure difference of 2000 mm w.g.).
e) Definition of new methods of test efficiency suitable for non normal work-conditions.
f) Evaluation of the ageing of filters by exposure of filters at certain conditions for scheduled times.

3. Project status
1. Progress to date

A test rig for generation of accident conditions and for evaluation of filter efficiency and pressure - drop was completely designed.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
EVALUATION OF HEPA FILTER SYSTEMS, USED IN NUCLEAR PLANTS, UNDER HEAVY ENVIRONMENTAL CONDITIONS.	5.3

4. Next steps

   The rig will be assembled within 1983.

5. Degree of availability

   Open.

   Contact person: G. Caropreso, ENEA, CRE Casaccia, C.P.2400, I-00100 Roma.

6. Additional information

   Personnel involved: 5 persons.

   Budget (1982): 1600 millions Lit.

	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 1980	

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: 5.2 5.3 5.4
TITLE (ORIGINAL LANGUAGE): CRITICAL ANALYSIS OF ACCIDENT SCENARIOS AND CONSEQUENCES MODELLING APPLIED TO LWR POWER PLANTS, FOR ACCIDENT CATEGORIES BEYOND THE DESIGN BASIS ACCIDENT	COUNTRY: Italy SPONSOR: CEC
TITLE (ENGLISH LANGUAGE):	ORGANISATION: ENEA PROJECT LEADER: G.PETRANGELI

INITIATED: Nov.1981	COMPLETED: (expected) 1982	SCIENTISTS: C.BROFFERIO P.CAGNETTI V.FERRARA A.MANICIA
STATUS: In progress	LAST UPDATING:	

	CLASSIFICATION : 5.3
TITLE (ORIGINAL LANGUAGE) : Kompletterande jodprov med sand	COUNTRY : SWEDEN
	SPONSOR : Swedish Nuclear Power Inspectorate
TITLE (ENGLISH LANGUAGE) : Complementary tests on the retention of iodine in sand beds	ORGANISATION : Studsvik Energiteknik AB
	PROJECT LEADER : Rolf Hesböl

INITIATED : 1982-07-21	COMPLETED : 1983-09-31	SCIENTISTS : Rolf Hesböl
STATUS : In progress	LAST UPDATING : June 1983	

In the event of a core melt down accident containment damage is prevented by the rupture of a pressure disc. By this mean the overpressure is released and the gases are vented through a large stone condenser. The retention of elemental iodine in the stone condenser has been examined experimentally.

In previous studies with gas or steam flows it was learned that elemental iodine was retained by chemisorption (1.2).

In the present project the removal of elemental iodine from water running through a testbed has been examined. The iodine species in water upstream and downstream from the test bed have been measured by a spectrophotometer.

It has been demonstrated that elemental iodine is reduced to iodide, quantitatively if the bed is properly dimensioned.

The technical part of the project is finished and a final report will be issued Sept 30, 1983.

References

1. HESBÖL, R
The retention of elemental iodine in sandstone beds.
Experimental results.
Studsvik Technical Report NW-81/138

2. HESBÖL, R
The retention of elemental iodine in gravel beds.
Experimental results of steam runs.
Studsvik Technical Report NW-82/211

Contact person

Bo Liwång, Swedish Power Inspectorate
Sehlstedtsgatan 11
S-102 52 STOCKHOLM
SWEDEN

	CLASSIFICATION : 5:1, 5:3, 5:4, 5:5 och 18
TITLE (ORIGINAL LANGUAGE) : Transuraner i reaktoravfall	COUNTRY : Sweden SPONSOR : S.S.I., National Institute of Radiation Protection
TITLE (ENGLISH LANGUAGE) : Transuranic Nuclides in Low-Level Wastes from Power Reactors in Sweden.	ORGANISATION : Dept. of Nuclear Chemistry Chalmers Univ. of Techn. PROJECT LEADER : J.O. Liljenzin
INITIATED : 1983-01-12 / COMPLETED : 1984-09-30 STATUS : In progress / LAST UPDATING : 1983-06-01	SCIENTISTS : S. Johnson Tech. ass.: I-M. Ehrlund

	CLASSIFICATION : 5.3
TITLE (ORIGINAL LANGUAGE) : Jodretention i stenbädd	COUNTRY : Sweden SPONSOR : SKI
TITLE (ENGLISH LANGUAGE) : Iodine retention in a gravel bed	ORGANISATION : Studsvik Energiteknik AB PROJECT LEADER : Hans Häggblom
INITIATED : July, 1982 — COMPLETED : October, 1983	SCIENTISTS : Hans Häggblom
STATUS : In progress — LAST UPDATING : June 1983	

1. General Aim: Development of a computational model for iodine adsorption in granular beds. Comparison with experiments. Calculations for the FILTRA filtered venting system.

2. Particular Objectives:
Prediction of the iodine adsorption in the FILTRA gravel bed for different meltdown accident sequencies.

3. Experimental Facilities and Programme:
Experiments have been made by R Hesböl in another part of the FILTRA project.

4. Project Status:
   1. Progress to Date
      The work has been essentially finished.

   2. Essential Results
      The computer program SAD has been developed and gives satisfactory agreement with experiments. Calculations on the FILTRA bed shows that in most cases all of the iodine will be adsorbed in the bed. In one case the penetration factor was 10^{-3}.

5. Next Steps: No future work is planned.

6. Relation to Other Projects and Codes:
Experimental work: R Hesböl, STUDSVIK Techn Rep NW-81/138 (1981)

7. Reference Documents:

H Häggblom,
Surface Adsorption of Chemically Active Gases After LWR Core Meltdown Accidents,with Applications to a Bed for Filtered Venting
STUDSVIK/nR-82/181 (1982)

H Häggblom,
Iodine Retention in FILTRA for Some BWR Core Meltdown Accident Scenarios
STUDSVIK Techn Rep NR-83/223 (1983)

8. Degree of Availability:

The SAD code is the property of the Swedish Nuclear Inspectorate.

	CLASSIFICATION: 5.3
TITLE (ORIGINAL LANGUAGE): PWR Coolant Chemistry Studies C.1	COUNTRY: United Kingdom
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA Winfrith
	PROJECT LEADER: Dr D J Ferrett

INITIATED: April 1979	COMPLETED:	SCIENTISTS:
STATUS: In progress	LAST UPDATING:	

Background: This work is designed to quantify the influence of coolant chemistry in determining annual man-rem dose to PWR operating and maintenance personnel.

Objectives:

(1) To identify the major contributors to dose in a PWR.

(2) To measure experimentally the release rates and subsequent fate of dose significant nuclides from reactor components operating in a range of PWR chemistries.

(3) To quantify the effect of circuit purification techniques, in particular magnetic filtration, on the deposition of active nuclides on plant pipe-work surfaces.

Experimental facilities and programme:

On-going programme of magnetic filter studies on the primary circuit of SGHWR. Refreshed PWR autoclave work on release of Co^{60} from irradiated Inconel. Dynamic loop studies on wear/erosion of hard facing alloys.

	CLASSIFICATION: 5.3
TITLE (ORIGINAL LANGUAGE): DECONTAMINATION (C 2.1)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA WINFRITH
	PROJECT LEADER: DR DJ FERRET (Winfrith)
INITIATED: MARCH 1978 | COMPLETED:	SCIENTISTS:
STATUS: IN PROGRESS | LAST UPDATING: SEPTEMBER 1980	

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.

PROGRAMME

Laboratory experimental studies at Winfrith.

FACILITIES

Laboratory rigs at AEE Winfrith.

DIDO Water Loop at AERE Harwell.

	CLASSIFICATION: 5.3
TITLE (ORIGINAL LANGUAGE): GAS PHASE TRAPPING STUDIES	COUNTRY:
	SPONSOR: UK
TITLE (ENGLISH LANGUAGE):	ORGANISATION: WNL WINDSCALE
	PROJECT LEADER: JJ HILLARY

INITIATED: 1972	COMPLETED:	SCIENTISTS:
STATUS:	LAST UPDATING: AUGUST 1980	

DESCRIPTION

1. GENERAL AIM

Improvement and standardisation of aerosol trapping, with particular reference to normal emissions for reactors.

2. PARTICULAR OBJECTIVES

To define and thus control the qualities of charcoal which affect ageing.

3. EXPERIMENTAL FACILITIES AND PROGRAMME

Apparatus 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

Some results have been published recently in IAEA-SM-245/17 (Feb 1980). The studies are continuing.

	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: SEPTEMBER 1980	

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 exponential programme in the DIDO Water Loop commenced in mid-1980 when the loop modifications were tested, and base-line data on the chemical and radiochemical characteristics of circulating corrosion products are being 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.

FACILITIES

DIDO Water Loop at Harwell.

5.4. ENVIRONMENTAL EFFECTS

	CLASSIFICATION : 5.4
TITLE (ORIGINAL LANGUAGE) : Konsekvenser af frigørelser af fissionsprodukter	COUNTRY : DENMARK SPONSOR : Risø National Laboratory
TITLE (ENGLISH LANGUAGE) : Consequences of Releases of Fission Products to the Atmosphere	ORGANISATION : Risø National Laboratory PROJECT LEADER : O. Walmod-Larsen
INITIATED : 1972 / COMPLETED :	SCIENTISTS : S. Thykier-Nielsen, Torben Mikkelsen
STATUS : Progressing / LAST UPDATING : September 1983	

1. General aim

Calculation of consequences of realease 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 deposited on the ground.

b. Consequences 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 meteorolocical parameters.

e. Isodose curves: Shape, area and number of people receiving doses within specific limits.

f. Number of consequences (i.e. number of people having e.g. early illness, cancer) for a given release.

Both normal and accidental releases are considered. Furthermore the parameters in the models are studied: Duration of release, atmospheric stability, plume rise, ground roughness etc.

3. Experimental facility and programme.

None

4. Project status

A computer model, PLUCON4, based on the gaussion plume model and fulfilling the objectives a. = f. mentioned in section 2, has been developed. PLUCON4 can be used for calculation of doses and consequences in near-zone, i.e. the area within 50 km from the release point.

A limited comparison between PLUCON4 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.

PLUCON4 has participated in the CSNI International Problem on Consequence Modelling.

Plucon4 has been used for a calculation of doses from hypothetical accidents at a nuclear power plant [2,4] .

5. Next steps

Development of a model, PUFFCON, for calculation of consequences of accidental releases in situations where the meteorological conditions varies with time. This model is based on a puff dispersion model, It is a three dimentional computer model which simulates the release of pollutant puffs and predicts their concentration as they diffuse while being advected downwind by a time-dependent wind.

Calculation of external gamma doses from airborne as well as deposited radioactivity will be included in the model. Thus the final model will be able to calculate collective doses, number of consequences etc.

6. References

1. Comparison of Nordic Dose Models, S. Thykier-Nielsen, Risø-M-1972.

2. Calculation of the Individual and Population Doses on Danish Territory Resulting from Hypothetical Core-melt Accidents at the Barsebäck Reactor, P. Hedemann Jensen, E. Lundtang Petersen, S. Thykier-Nielsen and F. Heikel Vinther, Risø Report No. 356.

3. The Risø Model for Calculating the Consequences of the Release of Radioactive Material to the Atmosphere, S. Thykier-Nielsen, Risø-M-2214 (September 1980).

4. Some Consequences of Land Contamination on Danish Territory from Radioactivity following a Hypothetical Core-melt Accident at the Barsebäck Nuclear Power Plant, H.L. Gjörup et. al., Risø-M-2212 (November 1981).

5. Description of the Risø Puff Diffusion Model, Torben Mikkelsen, Søren E. Larsen and S. Thykier-Nielsen, RNL (May 1983).

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

Available on an exchange basis. The computer programmes are written in Burroughs ALGOL for a B 7800 computer.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	5.4	06.02.01/08A (PNS 4513)
Vorhaben/Project Title		Land/Country
Störfallablaufanalyse für die große Wiederaufarbeitungsanlage (Extraktion) Incident Analysis for the Large Reprocessing Plant (Extraction)		Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe GmbH (KfK) Projekt Nukleare Sicherheit (PNS) IDT
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.01.82		Dr. K. Nagel
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
continuing	December 1982	-

1. General Aim

Disturbances in the dissolver off-gas cleaning system for a large reprocessing plant will be investigated by a time-dependent simulation model as well as an analytical model.

2. Particular Objectives

2.1 To describe the physical behaviour of installed barriers.

2.2 To evaluate a time-dependent model for calculating probability distributons of released gaseous radioactive fission products.

3. Research Program

3.1 We will build fault trees of possible failures in seperate units.

3.2 Investigations of a disturbance in one of the seperate units and their consequences on the compound system.

3.3 Dependences of barrier behaviour on temperature, pressure, chemical composition of the off-gas.

3.4 Development of an analytical stochastic model on the basis of renewal and Semi-Markov processes giving the probability distributions of the radioactive products released during the first incident as well as at any time t.

4. Experimental Facilities, Computer Codes

RIDO (TU Berlin), SIPAS (KfK, IDT).

5. Progress to Date

5.1 Incident analyses of a fiber packed mist eliminator (Brink filter), a HEPA-filter, and a iodine-sorption-filter have been performed using RIDO and SIPAS respectively.

5.2 Based on new experimental data the physical behaviour of barriers could be modelled in more detail.

6. Results

6.1 Incresing amounts of released aerosols seem only possible by random cracks in the HEPA-filter.

6.2 A large amount of released iodine is given if both heater of PASSAT failed within the dissolvation period of fuel (8 hours).

6.3 First numerical results for iodine released in PASSAT using the SIPAS Code for different maintenance and repair policies.

7. Next Steps

7.1 We will describe the physical behaviour of following units: ADAMO and KRETA.

7.2 We will use new experimental results to get better physical dependences of existing models.

7.3 Numerical evaluation of the analytical model and comparison of the results given by the simulation models RIDO and SIPAS.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.82 - 31.12.82	5.4	06.03.01/07A (PNS 4812)
Vorhaben/Project Title		Land/Country
Investigation of the physical and chemical environmental behavior of radionuclides characterized by a particular biological effectiveness. Pu, Am, Cm, and Np.		Fördernde Institution/Sponsor BMFT
		Auftragnehmer/Zuwendgsempf./Contractor Kernforschungzentrum Karlsruhe Projekt Nukleare Sicherheit
Arbeitsbeginn/Initiated 01.07.1978	Arbeitsende/Completed 31.12.1982	Leiter des Vorhabens/Project Leader Dr.H.Schüttelkopf
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

   Determination of the long-term exposure of the environmental population after an accidental release of Pu, Am, Cm, and Np.

2. Particular Objectives

   The behavior of Pu, Am, and Cm in the environment of the Karlsruhe reprocessing plant.

3. Research Program

   3.1 Development of analytical methods for the determination of Pu, Am, Cm, and Np.

   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.

   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 Program

5. Progress to date

   5.1 The analytical procedures for Pu, Am, and Cm were proved and adjusted on all plant materials which will be produced in a greenhouse.

   5.2 The literature was investigated for analytical methods to determine Np in environmental materials. The development of an analytical procedure was started, which will allow to determine Np parallel to Pu, Am, and Cm.

   5.3 The releases of Pu with gaseous and liquid effluents of the Karlsruhe reprocessing plant were measured monthly.

- 2 -

5.4 Experiments were continued to increase the mobility of actinides in soil columns.

5.5 Preliminary work was done for the measuring of transfer factors in a greenhouse.

5.6 Investigations on the field of conventional soil analysis were continued.

5.7 The ICP-instruments which are available on the market were compared. That one which is the best suitable for simultaneous determination of 40 elements in soil and biological materials was ordered.

6. Results

6.1 Samples from the vincinity of the Karlsruhe Nuclear Research Center of 20 different agricultural plants, which will be grown in the greenhouse, were analysed for Pu, Am, and Cm. The analytical procedure was improved and successfully applied. Besides of leafy vegetables, which showed a very low Pu contamination, no activity concentrations higher than the detection limit were found.

6.2 With Np-239 as a tracer the analytical procedure used for Pu was modified in such a way that it is possible to determine Np in environmental samples in high yields. The applicability to Np-237 must be investigated, before the analytical methods for Pu and Np can be connected for a simultaneous determination.

6.3 Pu-241 is measured monthly in the gaseous and liquid effluences of the Karlsruhe reprocessing plant in addition to Pu-238 and Pu-239+240. Further more, Pu-241 was determined in samples from the terrestrial and aquatic environment of the Karlsruhe Nuclear Research Center.

6.4 In total, 20 soil columns of 30 cm diameter and 80 cm of length, which were taken undisturbed at 9 locations of different agricultural important soil types, were installed in an greenhouse and grass was sown. The columns become irrigated until a grassplot has build up and constant conditions concerning soil physics have been established, before the investigation of migration behavior will be started up.

6.5 For growing plants on contaminated soils the growing procedures in common use in agriculture and horticulture are not suitable. Therefore plant cultivation was started on not contaminated soils to work out special cultivation technics which can be used when plants are to be grown on contaminated soils. Different methods for covering the soil surface to prevent resuspension were proved and suitable irrigation procedures were determined. Moreover, climatic parameters for all plants to be cultivated in the greenhouse were proved and established.

7. Next steps

The monthly measuring ot the releases of Pu from the Karlsruhe reprocessing plant and the determination of the actinide concentrations in the environment of this plant will be continued. In addition to experiments to increase the mobility of actinides in soil the measuring of transfer factors will be started up. The work on conventional soil analysis will be continued and extended.

8. Relations with other projects

9. References

10. Degree of Availability of the Reports

Unrestricted distribution.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.82 - 31.12.82	5.4	06.03.02/18A (PNS 4826)
Vorhaben/Project Title		Land/Country
Modelling of the Long Range Transport of Pollutants Modellierung des weiträumigen Schadstofftransports		Fördernde Institution/Sponsor
		Auftragnehmer/Zuwendgsempf./Contractor Nuclear Research Center Karlsruhe (KfK) Project Nuclear Safety (PNS)
Arbeitsbeginn/Initiated 1.1.81	Arbeitsende/Completed 31.12.1984	Leiter des Vorhabens/Project Leader Dr. G. Halbritter
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

Work performed

Within the framework of this project, the Trajectory Puff modell MESOS, which was developed by H. ApSimon and A.J.H. Goddard at Imperial College, London, was implemented and tested on the KfK computer facility. The model was modified in order to take account also of the daughter nuclides generated during the atmospheric transport of radioactive pollutants (max. 3), their atmospheric transport as well as their radioactive decay and deposition on soil.

Calculations were performed for the site at Jülich and a pollutant emittor (source) of a height of 10 m for the long-range dispersion of Caesium-137 (aerosol), Iodine-131 (reactive gas) and a nobel gas. Mean values for atmospheric and ground-level contamination as well as statistical indices for short-term exposure from accidental releases were calculated. These calculations form the basis for the comparison of the MESOS model and the diffusion model as applied in the accident-consequence-model of the German Risk Study on Nuclear Power Plants (UFOMOD).

Main results

The results of the two models, MESOS and UFOMOD, an in good agreement concerning the mean-values of pollutants deposited at different distances from the source. While at a distance of 900 km from the source only 20 % (MESOS) or 10 % (UFOMOD), resp. of the originally released reactive gas survive in the atmosphere, the fraction of aerosols remaining airborne amounts to still 50 % at the same distance.

The main difference between the two diffusion models consists in the way the dispersion directions of the emitted puff are taken into account. In the Trajectory

Puff model MESOS the dispersion direction of the emitted pollutant puff is calculated for any point in space and at any time period. The mean ground-level concentration in Ci m^{-2} from a 3 hours release of 1 Ci at the site of Jülich between 44^{0} and 62^{0} northern latitude and 10^{0} western and 20^{0} eastern longitude on the basis of meteorological data of the year 1973/74 shows a complex spatial distribution.

The way in which the isolines of the ground-level contamination along the Alps approach each other closely in southernly and south-easterly directions from the source is striking. The comparison of the spatial distribution of the mean ground-level contamination for the nuclides Caesium (aerosol) and Iodine (reactive gas) shows the more complex structure for the case of the aerosol due to the importance of the wet deposition for ground-level aerosol contamination. While at shorter distances from the source areas with high contamination for Iodine-131 are more expanded compared to those for Caesium-137, at larger distances from the source ($\geq$ 200 km) areas with a contamination higher than a certain value for aerosol contamination are much larger.

As the diffusion model used in the UFOMOD model assumes the same probabilities for each dispersion direction of an emitted pollutant puff it can be expected that significant differences between the diffusion models will emerge, esp. with regard to the spatial distribution of mean ground-level contamination.

Plans for the near future

The comparison of the MESOS model and the diffusion model from UFOMOD will be continued and finished during 1983. A final report about the results will be published.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01. - 31.12.1982	5.4	06.03.02/15A (PNS 4822)
Vorhaben/Project Title Investigation of the Atmospheric Dispersion of Radioactive Substances in the Mesoscale (more than 15 km distance) Untersuchung der Ausbreitung radioaktiver Stoffe im regionalen Bereich (> 15 km Entfernung)		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe GmbH (KfK) Projekt Nukleare Sicherheit (PNS) - HS
Arbeitsbeginn/Initiated 01.03.1977	Arbeitsende/Completed 1984	Leiter des Vorhabens/Project Leader Dr. Hübschmann
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1982	Bewilligte Mittel/Funds

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

The tetroons (Helium-filled balloons of constant volume) are equipped by a passive radar-reflector or by a transponder.

5. Progress to Date

A tetroon flight series has been carried out from 12th through 17th of May. Five tetroons have been launched and radar tracked at the Rohrbacher Hof (east side of the Rhine valley) during NE winds, 3 tetroons at Minfeld (west side of the Rhine valley in the Pfalz area) during south-western wind. All of the tetroons were equipped with transponders and were tracked up to 83 km, at an average height of 400 m above ground. Four out of the eight transponders have been sent back to KfK.
Two other flight series took place at Gundremmingen and at Essen. The tetroons were tracked by a radar of the German Meteorological Service. Accordingly, the tetroons were equipped by radar reflectors and were tracked up to between 14 and 31 km.

6. Results

The 29 tetroon flights of the PUKK campaign carried out in Sept./Oct. 81 have been plotted. From the successive flight series the horizontal dispersion parameters σ_y as dependent on the source distance has been derived. These σ_y are somewhat greater than those derived from diffusion experiments and extrapolated to greater distances.

The average tracking distance of transponder equipped tetroons is 2.2 times that of reflector equipped tetroons.

7. Planned Future Work

Tetroon flight series will be continued. It is investigated, whether multiple tracking of several tetroons simultaneously is feasible.

8. Relation with Other Projects

PNS 4824, PNS 4825.

9. References

H. Kiefer, W. Koelzer, L. A. König, editors;
Jahresbericht der Hauptabteilung Sicherheit 1982, KfK 3535.

10. Availability of the Reports

Reports are available through Kernforschungszentrum Karlsruhe GmbH, Karlsruhe, Zentralbücherei.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01. - 31.12.1982	5.4	06.03.02/16A (PNS 4824)
Vorhaben/Project Title Atmospheric Diffusion Models for Particular Meteorological Situations Ausbreitungsmodelle für besondere meteorologische Situationen		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe GmbH (KfK) Projekt Nukleare Sicherheit (PNS) - HS
Arbeitsbeginn/Initiated 01.01.1979	Arbeitsende/Completed 31.12.1985	Leiter des Vorhabens/Project Leader Dr. Hübschmann
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating Dec. 1982	Bewilligte Mittel/Funds

1. General Aim

Development of atmospheric diffusion models for particular meteorological situations.

2. Particular Objectives

In particular the models are to be developed for the following situations:

- low wind velocity ($\bar{u}$ < 1 m/s; at small wind velocity the concentration distribution approximation by a double Gaussian distribution is no longer applicable).
- inhomogeneous wind field; this implies a. o. wind direction shear and transitional states like low elevated inversion layers.

3. Research Program

Tracer experiments are performed during weak wind velocities and inhomogeneous situations.

4. Experimental Facilities

Automatically operated, portable sampling device;
tracer evaporator;
gaschromatograph for the measurement of the tracer concentration.

5.+6. Progress to Date and Results

A planned SF_6 diffusion experiment to be performed in cooperation with JRC Ispra has been cancelled because of unfavourable meteorological conditions during the foreseen time period.

Diffusion experiments performed during very unstable stratification (category A, partly weak winds) and with emission heights of 160 m and 195 m have been reevaluated. A new evaluation code has been programmed according to a proposal of R. G. Lamb, USA. It takes into account the long-wave vertical air motion in predominantly convective turbulence. The plume axis, therefore, is deformed. For the category A experiments a satisfactory correlation of measured and calculated tracer concentrations has been achieved. It is to be expected, that this result will hold also for off-gas plumes released at greater emission height or with plume rise.

The plume trajectory code SPALT has been applied to a diffusion experiment with weak wind velocity and changing wind directions. Tracer concentrations around the concentration maximum are reasonably well approximated, if the actual wind directions and velocities measured at 40 m and 100 m height are applied and the long-wave rotations of the wind direction are used to evaluate the trajectory. The comparison gets less favourable beyond 1000 m distance from the source.

7. Plans for Future Work

The experiment evaluation with the computer code according to Lamb will be continued. The evaluations will be extended to experiments performed with source heights of 60 m and 100 m.

8. Relation with Other Projects

PNS 4822 and PNS 4825.

9. References

H. Kiefer, W. Koelzer, L. A. König, editors;
Jahresbericht der Hauptabteilung Sicherheit 1982, KfK 3535.

10. Availability of the Reports

Reports are available through Kernforschungszentrum Karlsruhe GmbH, Karlsruhe, Zentralbücherei.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01. - 31.12.1982	5.4	06.03.02/17A (PNS 4825)
Vorhaben/Project Title Invesigation of remote sensing methods in respect to their suitability to measure meteorological parameters in the atmospheric boundary layer Untersuchung meteorologischer Fernmeßmethoden auf ihre Verwendbarkeit für Messungen in der atmosphärischen Grenzschicht		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe GmbH (KfK) Projekt Nukleare Sicherheit (PNS) - HS
Arbeitsbeginn/Initiated 1981	Arbeitsende/Completed 1985	Leiter des Vorhabens/Project Leader Dr. Hübschmann
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

Investigation of meteorological remote sensing methods in respect to their suitability for the required meteorological measurements at nuclear power plant sites and to increase the limited range of meteorological towers.

2. Particular Objectives

Remote sensing devices, especially Doppler-SODAR systems (sound detection and ranging) are to be tested in respect to:

measurement range,
accuracy,
reliability.

3. Research Program

A remote sensing system, preferably a Doppler-SODAR-system, is installed near the meteorological tower and operated parallel to the tower for a representative time period.

4. Experimental Facilities

- remote sensing system, for instance Doppler-SODAR,
- pertinent data recording on tape or disc.

5. Program to Date

Two Doppler-SODAR-systems have been installed at KfK for a one year period. They are located close to the meteorological tower. The SODAR transmits a sound pulse of 1600 Hertz every 3 seconds. The Doppler-shift of the backscattered signal is used to calculate the wind vector. The required range of detection is 420 m height.

The test period of one of the SODAR instruments (SODAR-R) was completed at the end of 1982. The test of the other instrument (SODAR-B1) was interrupted because of service and performance problems. An improved system (SODAR-B2) replaced this instrument at the end of 1982.

6. Results Obtained

The performance of the SODAR-R was satisfactory during summer and winter. Wind direction and wind velocity are well correlated to the respective parameters measured at the 200 m high meteorological tower.

7. Next Steps

Final evaluation of the SODAR-R measurements, operation of the SODAR-B2, and evaluation of SODAR-B2 measurements.

8. Relation with Other Projects

PNS 4822, PNS 4824.

9. References

H. Kiefer, W. Koelzer, L. A. König, editors;
Jahresbericht der Hauptabteilung Sicherheit 1982, KfK 3535.

10. Availability of the Reports

Reports are available through Kernforschungszentrum Karlsruhe GmbH, Karlsruhe, Zentralbücherei.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.1982-31.12.1982	5.4	06.03.01/08A PNS4813
Vorhaben/Project Title Microbiological influences on the mobility and bioavailability of radionuclides in soils and sediments Mikrobiologische Einflüsse auf die Mobilität und Bioverfügbarkeit von Radionukliden in Böden und Sedimenten		Land/Country Fördernde Institution/Sponsor BMFT/EG-BIO-B-484-82-D Auftragnehmer/Zuwendgsempf./Contractor Kernforschungszentrum Karlsruhe GmbH (KfK) Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated 01.01.1982	Arbeitsende/Completed 31.12.1984	Leiter des Vorhabens/Project Leader Dr. S. Strack
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds

1. General Aim

1.1 Evaluation after radioactivity releases from nuclear facilities of the radiation exposure to people living nearby. This work is to make a contribution to modeling the behavior of nuclides which are radio-ecologically relevant in ecosystems. The work was initiated by the long-term changes observed in the vicinity of the Karlsruhe Nuclear Research Center (KfK) as regards the bioavailability in the soil of I-129 and Pu-nuclides. Microbiological processes in the layer of humus might account for it.

1.2 Only little information has been collected so far about the role of microorganisms in transfer and retention processes of radionuclides, both in terrestrial and aquatic systems. Therefore, it is intended to clarify in experimental studies how migration, retention and changes in the biological availability of radionuclides in soil can be influenced by microbiological activity. It is planned to concentrate these studies initially to the radionuclides currently used in biological tracer tests such as H-3, C-14 and P-32, but to extend them very soon to include radioecologically relevant nuclides such as Sr-90, I-129, Cs-137 and Pu-139.

2. Particular Objectives

2.1 Investigation into the migration behavior of radionuclides applied to soil columns as various chemical compounds. It is intended to determine the consequences of specific interferences in the composition and metabolic activity of the microflora.

2.2 Investigation into the fine distribution of the immobilized nuclides in the soil material.

3. Research Program

3.1 Development of methods of studying the fine distribution in the soil material of the nuclides employed.

3.2 Development of extended methods of determining the germ content and the microbiological metabolic activity in the soil columns.

3.3 Measurement of migration rates in untreated soil columns.

3.4 Observation of the changes in the migration behavior after specific treatments of the microflora.

3.5 Investigation into the fine distribution of the nuclides left in the soil columns.

3.6 Whole-year study of the microflora of a forest soil at a point of the Nuclear Research Center where the soil material required for the experiments is permanently collected.

4. Test Facilities, Computer Codes

Development and testing of suitable percolator apparatuses.

5. Work Performed

5.1 Various methods of quantitative determination of the microflora and its metabolic activity were examined.

5.2 Preliminary tests were performed with percolator apparatuses in order to find out suitable experimental conditions.

5.3 Conditioned soil samples, incubated with H-3 and C-14 compounds, were examined using microautoradiography.

5.4 Monthly determinations of the number of germs in the soil of a test field performed since August this year.

6. Results Obtained

The preparatory work started this July has advanced so far that the laboratory experiments proper can begin early next year.

7. Plans for Future Work

8. Relations with Other Projects

9. Literature

10. Availability of Reports

unrestricted

148.2.03		5 - 4
TITRE ETUDE DES TRANSFERTS ATMOSPHERIQUES		*Pays* FRANCE
		Organisme directeur CCE (pour partie du programme) CEA/DSN
TITRE en anglais STUDIES ON ATMOSPHERIC TRANSFERS		*Organisme exécuteur* CEA/DSN/SAER /FAR
		Responsables B.CRABOL/DSN/SAER
Date de démarrage 01.01.72	*Etat actuel* En cours	*Scientifiques* G.DEVILLE-CAVELIN DSN/SESTR
Date d'achèvement 31.12.84	*Dernière mise à jour* Décembre 1981	

1) OBJECTIF GENERAL

Cette étude a pour objectifs 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 d'un polluant minoritaire résultant des rejets permanents ou accidentels des installations industrielles, nucléaires notamment.

2) OBJECTIFS PARTICULIERS ET PLANNING

2.1 - Objectifs théoriques

a) Développement et perfectionnement du modèle de transfert atmosphérique utilisé au DSN par la prise en compte de :
- la variation spatiale et temporelle du vent (1983)
- la surélévation des nuages ou panaches (1983)
- l'effet de terrain construit (1984)
- les dépôts secs par turbulence et les dépôts gravitaires (1983)

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) (1983),
- 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, étude de la corrélation des paramètres météorologiques de basse couche sur le site avec les paramètres synoptiques) (1984)

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) (1983).

b) Adaptation éventuelle du modèle DSN pour des situations météorologiques et/ou orographiques complexes (vents faibles, terrain construit, tâche thermique des villes, relief) afin d'étendre son domaine d'application (en particulier

ajustement éventuel des écarts-types (1983).
c) Développement et perfectionnement des deux techniques expérimentales précédentes (traceurs sur le terrain et maquette en veine hydraulique) (permanent).

3) INSTALLATIONS EXPERIMENTALES

Les moyens expérimentaux sont essentiellement :

- deux mâts télescopiques de 30 m* qui permettent de constituer deux stations météorologiques simplifiées mais mobiles.
- Le matériel de prélèvement et d'analyse des traceurs* (SF_6 et Fréon 13 B 1).
- Une veine hydraulique ** (30 m de long, 3m de large, 1 m de haut), permettant des vitesses hors couche-limite de $0{,}03 ms^{-1}$ à $0{,}7 ms^{-1}$.

Cette installation est utilisée pour les études de simulation sur maquette de la dispersion dans l'hypothèse d'une atmosphère neutre.

4) ETAT D'AVANCEMENT - PRINCIPAUX RESULTATS

Objectif particulier 2.1.a

- Développement de codes de calcul permettant la prise en compte de la variation spatiale et temporelle du vent (codes ICAIR 3 et TRAIR 3).
- Développement d'un modèle de jet dans l'atmosphère avec prise en compte de la décroissance radioactive des produits de fission qui y sont contenus (CEN G/STT).
- Développement d'un modèle de calcul du dépôt sec au sol des particules transportées par l'atmosphère. Ce modèle a été étendu au cas des aérosols sodés polydispersés.

Objectif particulier 2.1.c

La comparaison des résultats du modèle DSN avec ceux que l'on obtient avec d'autres modèles français ou étrangers a été poursuivie dans le cadre des groupes de travail 8B et 10 de la Commission Franco-Allemande (SCSIN-BMI).
Cette comparaison s'est exercée dans un premier temps au niveau des données de base (météorologie et paramètres de dispersion) servant aux calculs : une compilation et une mise en forme de ces données ont été effectuées pour les site de Fessenheim (F) et Wyhl (RFA).

Objectifs particuliers 2.2.a et 2.2.b

. Par des expériences de simulation en veine hydraulique et des considérations sur le spectre de turbulence atmosphérique, on a pu préciser les lois d'évaluation des écarts-types de dispersion des particules en atmosphère, en fonction de la vitesse du vent et du temps de transfert (champ proche) ou du temps de transfert seul (champ lointain).
. Qualification de modèles de prévision des transferts atmosphériques ICAIR 3 et TRAIR 3 dans les cas de vent pouvant varier fortement en vitesse et en direction (situation de vent faible notamment).
. L'interprétation des campagnes de simulation in situ par traceurs pour des conditions de vents faibles a été poursuivie. Elle a notamment donné lieu à la création de nouveaux jeux d'écarts-types de dispersion longitudinale et transversale correspondant au temps d'échantillonnage de 2mn utilisé lors des expériences.

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* matériel implanté au CEN/CADARACHE/SESTR

** exploitée par la Société SECURIPOL à EVIAN.

Cette étude est réalisée dans le cadre d'un contrat passé avec la Communauté Européenne de l'Energie Atomique . Elle comporte principalement les 3 points suivants :

. interprétation des campagnes de simulation in situ par traceur pour des conditions de vent faible (CADARACHE),

. adaptation des écarts-types σ_x et σ_y à la durée T sur laquelle sont moyennées les données météorologiques pour le calcul des trajectoires,

. étude de la sensibilité des résultats aux paramètres de l'acquisition des données météorologiques (durée de l'acquisition, nombre de stations de mesure) pour différentes situations météorologiques.

Des expérimentations réalisées in situ et sur maquette pour un terrain construit (site de Pierrelatte) ont permis de tester la validité des principales formulations classiquement utilisées pour évaluer les transferts sur ce type de terrain.

Objectif particulier 2.2.c.

. La méthode de simulation sur maquette des transferts en atmosphère neutre a été qualifiée dans certaines limites que l'on a pu définir par une étude à la fois théorique et expérimentale.

. Le prototype de l'appareil de prélèvements à 10 voies avec commutation automatique d'une voie à l'autre et durée des prélèvements réglable a été testé sur le terrain. 20 appareils ont déjà été réalisés (20 autres seront disponibles fin 83). Ils sont actuellement utilisés pour les expérimentations en cours (objectifs particuliers 2.2.a et 2.2.b).

. Participation à la mise au point et à la qualification du système d'acquisition SODAR en tant qu'instrument de surveillance atmosphérique des sites d'installations nucléaires.

5) PROCHAINES ETAPES

Objectif particulier 2.1.a

Prise en compte de la surélévation des nuages et panaches (1983).

Objectif particulier 2.1.b

Participation à la mise au point d'une méthodologie pour déterminer, sur un site donné, la corrélation entre les paramètres météorologiques à échelle locale et les paramètres à échelle synoptique, dans l'optique des situations accidentelles (1984).

Objectif particulier 2.1.c

Poursuite de la comparaison du modèle avec les autres modèles français ou étrangers.

Objectifs particuliers 2.2.a et 2.2.b

. Poursuite des expériences "vents faibles" sur le site de CADARACHE pour d'autres conditions météorologiques (neutre, instable). Interprétation à l'aide des codes ICAIR 3 et TRAIR 3.

. Interprétation des expériences "vents faibles" réalisées en situation stable sur un terrain plat (plateau de Saclay) (1983).

. Participation à la définition d'actions ultérieures (programme 84-87) dans le cadre du groupe C-DG XII de la CEE.

Objectif particulier 2.2.c

. Comparaison des deux techniques de simulation à échelle réduite : veine hydraulique et soufflerie. Poursuite de l'interprétation des résultats obtenus sur la maquette du site de Paluel.

6) RELATIONS AVEC D'AUTRES ETUDES

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. MAIGNÉ . Séminaire sur la dispersion en milieu physique naturel S.F.R.P. Cadarache/Mars 1978.

/ 4 / Study of pollutant dispersion in water air. R. PORTE et J.P. MAIGNÉ. Conférence ENS-ANS "The nuclear power reactor safety" - Bruxelles 16-19 octobre 1979.

/ 5 / Contribution à l'étude de la simulation en laboratoire des transferts de masse en atmosphère neutre. Thèse de 3ème Cycle - B.CRABOL.

/ 6 / Détermination des lois d'évolution des écarts-types de distribution de polluant à partir de considérations sur le spectre de turbulence atmosphérique et sa possibilité de simulation en laboratoire. B.CRABOL, Séminaire CEE - RISØ (Danemark) - Avril 1980.

/ 7 / Transfert de pollution par voie atmosphérique sur un site dont le vecteur vent est rapidement variable (vents faibles) J.P. MAIGNÉ - Séminaire CEE/RISØ (Danemark) - Avril 1980.

/ 8 / Un modèle de calcul du dépôt sec des particules et des gaz dans l'atmosphère - Z. WAN - Note SAER 81/243 - Mai 1981.

/ 9 / Modélisation des transports des aérosols sodés polydispersés dans l'atmosphère avec prise en compte du dépôt sec - Z. WAN - Note SAER 81/244 - Juin 1981.

/ 10/ Rapports d'avancements du contrat CEEA - B.CRABOL - juin 1981, Janvier 82, Juillet 82, Janvier 83.

/ 11 / Présentation de deux codes de calcul TRAIR 3 et ICAIR 3 de la dispersion de polluant dans l'atmosphère - B. CRABOL - Note SAER N°83/337 - Janvier 1983.

8) DEGRE DE DISPONIBILITE

Documents disponibles :
/2/, /3/, /4/, /5/, /6/, /7/.
Les autres documents sont internes, non disponibles sauf accord au CEA.

	CLASSIFICATION: 5.4 - 5.5 - 5.6
TITLE (ORIGINAL LANGUAGE): Valutazione quantitativa del rilascio di sostanze radioattive naturali nell'ambiente	COUNTRY: ITALY SPONSOR: ENEA
TITLE (ENGLISH LANGUAGE): Quantitative evaluation of the release of natural radioactive substances into the environment	ORGANISATION: ENEA PROJECT LEADER: M. Dall'Aglio
INITIATED: January 1974 / COMPLETED:	SCIENTISTS: R. Gragnani, C. Orlandi, G. Paganin
STATUS: in progress / LAST UPDATING: July 1980	

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.

	CLASSIFICATION : 5.4-5.5-5.6
TITLE (ORIGINAL LANGUAGE) : Rilascio e circolazione nell'ambiente di elementi stabili e radioattivi naturali dall'estrazione mineraria di uranio e dalla fabbricazione di elementi di combustibile.	**COUNTRY :** ITALY **SPONSOR :** ENEA
TITLE (ENGLISH LANGUAGE) : Release and circulation in the environment of natural radioactive and stable elements from Uranium Mining upto fuel fabrication.	**ORGANISATION :** ENEA **PROJECT LEADER :** G.G. MASTINO
INITIATED : January 1974 / **COMPLETED :**	**SCIENTISTS :** R. GRAGNANI, C. ORLANDI, G. PAGANIN, G. SCIOCCHETTI, F. SCACCO
STATUS : in progress / **LAST UPDATING :** January 1982	

1. General Aim

Researches on the environmental pollution from Uranium Mining activities and from fuel fabrication.

2. Particular Objectives

In the surrounding of two uranium mines (Novazza and Valvedello) and of a fuel fabrication plant (Bosco Marengo - Al)field and laboratory research are in progress in order to evaluate the man-induced pollution as regards as natural radioactive and stable trace elements. The first necessary step in order to get to a reliable assessment of pollution is to establish the natural level of aforesaid elements before the mines and fuel fabrication plants are in production.
Analyzed elements are U, Ra, Rn, Pb-Po, Pb, Zn, Cu, Cd, F and As with major emphasis to natural water, stream sediment and soil samples.

3. Experimental Facilities and Programme

A well equipped geochemical standard laboratory plus alfa and gamma spectrometry, neutron analysis equipment. Fluorimetric instrumentation for analysis.

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). Environmental surveys have been carried out at Novazza and Valvedello Mine areas and at the Bosco Marengo Site.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Release and circulation in the environment of natural radioactive and stable elements from Uranium Mining upto fuel fabrication.	5.4 - 5.5 - 5.6

5. Next Steps

Surveys are planned at two mine sites after the start of mine production.

6. Reference Documents

-Dall'Aglio M., De Cassan P., 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)".
Società Italiana di Mineralogia e Petrologia, Rendiconti, Vol. XXXII(1), pag.437-459, 1976.

-Clemente G.F., Dall'Aglio M., Gragnani R., Mastino G.G., Santaroni G.P., Scacco F., Sciocchetti G. "Pre-operational environmental survey for two uranium mine sites in northern Italy".
IAEA -"International Symposium on management of wastes from uranium mining and milling". Alburquerque, New Mexico, USA, 10-14 May 1982.

7. Degree of Availability

Free available by G.G. Mastino, ENEA, Istituto della Casaccia, C.P. 2400, I-00100 Roma.

8. Additional Information

Personnel: 6 persons

Budget (1982): 20 millions Lit.

	CLASSIFICATION: 5.4
TITLE (ORIGINAL LANGUAGE): TRASFERIMENTO DI ALCUNI ELEMENTI STABILI DALL'AMBIENTE ALL'UOMO E SUL LORO METABOLISMO IN UOMO	COUNTRY: Italy SPONSOR: Ist.Naz.Nutrizione
TITLE (ENGLISH LANGUAGE): THE TRACE ELEMENT PATHWAYS FROM THE ENVIRONMENT TO MAN AND ON THEIR METABOLIC BALANCE IN MAN	ORGANISATION: ENEA PROJECT LEADER: G.INGRAO
INITIATED: Ian.1969 / COMPLETED:	SCIENTISTS: G.P.SANTARONI, G.F.CLEMENTE, G.INGRAO
STATUS: In progress / LAST UPDATING: April 1982	

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 certan 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 relefence 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 elements 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, Cr, Co, Cs, Eu, Fe, Hg, Ni, Rb, Sb, Sc, Se, Sn and Zn. The trace elements content of the human body and their distribution in the various organs should be evaluated in members of the general population.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
THE TRACE ELEMENT PATHWAYS FROM THE ENVIRONMENT TO MAN AND ON THEIR METABOLIC BALANCE IN MAN.	5.4

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. 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 trace element content has been measured in human milk samples and related to various parameters (regional difference, dietary habits, days post partum, different feeds during the same day).
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.

5. Next Steps:
The trace element contentin various organs of human body will be measured by means of activation analysis of autopsy samples.

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" Proc.of the Conf.on Trace Substances in Environmental Health XII pp 23-30, 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" Reviews on Environmental Health, vol.III, No.1, pp 19-42

6.4 Mariani A., Clemente G.F., Santaroni G.P. (1979) "Mercury levels in food and its intake in high risk population groups". Biblthca Nutr.Dieta, vol.29, pp 32-38 (Kurger, Basel 1980)

6.5 Clemente G.F., Cigna Rossi L., Santaroni G.P. (1979) "Nickel in food and dietary intake of nickel". Biogeochemistry of Nickel,book to be published by J.Wiley and Sons.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
THE TRACE ELEMENT PATHWAYS FROM THE ENVIRONMENT TO MAN AND ON THEIR METABOLIC BALANCE IN MAN	5.4

6.6 Clemente G.F., Ingrao G., Santaroni G.P. (1981) "Trace element in human milk in Italy" in course of publication on The Sciences of the Total Environment.

7. Degree of Availability: Free
G.Ingrao, ENEA, CSN Casaccia, C.P. 2400, I-00100 Roma

8. Additional Information: Number of persons involved: 4.Budget (1982): 50.000.000

Trasferimento del plutonio e tritio dall'ambiente all'uomo.		CLASSIFICATION : 5.4
TITLE (ORIGINAL LANGUAGE) : Trasferimento del plutonio e tritio dall'ambiente all'uomo.		COUNTRY : ITALY SPONSOR : ENEA
TITLE (ENGLISH LANGUAGE) : Plutonium and Tritium Transfer from the environment to man.		ORGANISATION : ENEA PROJECT LEADER : G.F. Clemente
INITIATED : Jan. 1979	COMPLETED :	SCIENTISTS : P. Belloni G. Ingrao G. Santori
STATUS : in progress	LAST UPDATING :	

1. General Aim

The production and the need to dispose of transuranium elements, particularly plutonium, have been considered in the last decade as serious drawbacks to the develpment of nuclear energy. The discussion has recently been exacerbated by the controversy on development and construction of fast breeder reactors which use plutonium as nuclear fuel.
The risk of exposure to plutonium is associated with its radiotoxicity once deposited in the human body.
As a result of the presence of plutonium in the fallout of nuclear weapon tests in the 60's and in a few local accident releases, the environment and man contain small amounts of plutonium.
Tritium is one of the most interesting radionuclides of global character that may be released to the environment through the development of nuclear and fusion power.
Tritium is of particular interest as it is naturally produced and also present in large amounts in the fallout of nuclear weapon tests in the 60 's.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Plutonium and Tritium Transfer from the Environment to Man -2-	5.4

The risk of exposure to tritium is mainly associated with its radiotoxicity once deposited in human tissues as organically bound tritium.
The knowledge of the actual levels of both tritium and plutonium in th environment and in man and of the various mechanisms responsable of their transfer from the environment to human beings is then of a particular relevance to evaluate any possible effects of unplanned releases of tritium and plutonium into the environment.

2. Particular Objectives

The aim of the research programme is precisely to obtain information on the actual environmental levels, pattern of movement, distribution in typical ecosystem and burden of the general Italian population of tritium and plutonium in order to determine the radioprotectional significance of any release of such radionuclides under various environmental conditions.

3. Experimental Facilities and Programme

The tritium is measured in biological and environmental samples under the water (HTO) and bound form by means of liquid scintillation counting and a sample oxidizer technique.
The plutonium is measured by selective radiochemical separation of plutonium in dissolved samples and high resolutions solid state detector alpha spectrometry.

4. Project Status

The tritium content has been measured in many environmental samples representative of different continental ecosystems. The results obtained have indicated that the organic bound tritium may play a very important role on the long term behaviour of tritium in the environment.
The fallout plutonium content measured in Italian diet samples is in good agreement with the plutonium dietary levels found in other countries of the northern emisphere.

5. Next Steps

The tritium and fallout plutonium content is being measured in autopsy samples collected from individual of the general Italian population.
The fallout plutonium and tritium organ distribution in skeleton liver, kidneys, spleen and gonads as a function of various parameters (sex, age, etc.) is under study.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Plutonium and Tritium Transfer from the Environment to Man -3-	5.4

7. Reference Documents

Belloni P., Clemente G.F., Di Pietro S., Quaggia S. 1979: Dati preliminari sullo studio del trasferimento di bassi livelli dall'ambiente all'uomo. CNEN-Techinical Report RT/PROT (79) 5.

Clemente G.F., Belloni P., Di Pietro S., Santori G., Santaroni G.P.: Tritium and plutonium content in the Italian diet and transfer to man.In biological implications of radionuclides released from nuclear industries p.p. 257-264, IAEA, Vienna. (1979)

Belloni P., Clemente G.F., Di Pietro S., Quaggia S., Santaroni G.P. (1979) : Livelli di contaminazione da tritio in diete ed alimenti italiani. CNEN Technical Report RT/PROT (79) 5.

Belloni P., Clemente G.F., Di Pietro S. Quaggia S. (1980): Livelli di tritio nel sangue e nelle urine della normale popolazione e di soggetti contaminati. CNEN-Technical Report RT-PROT (80) 37.

G. Santori (1980) : Contenuto di $^{239,240}Pu$ nella dieta italiana, in Proceedings of the XXII AIRP National Conference, pp. 345-350.

A. Delle Site, Marchionni V., Santori G. (1979): Un metodo sensibile per la determinazione del plutonio in campioni ambientali. CNEN Technical Report RT/PROT (79) 21.

P. Belloni, G.F. Clemente, Di Pietro S., Merli S, G. Santori, Sgarbazzini M., (1982): Tritium and plutonium content in Human Tissues of the general population in Italy, in Proceedings of Third Internation Symposium of Radiation Protection, Advances in Theory and Practive, Inverness, Scotland 6-11 June 1982.

8. Free. Please contact G.F. Clemente

Environmental Science Division
ENEA - C.P. 2400 - 00100 Roma
Italy

9. Additional Information :

Personnel: 4 scientists and
2 technicians

Budget for 1982: Lit. 150.000.000= for technical equipments
Lit. 70.000.000= for expendable supplies.

	CLASSIFICATION: 5.4
TITLE (ORIGINAL LANGUAGE): LOCAL SCALE ATMOSPHERIC DIFFUSION AT A COASTAL SITE IN THE PRESENCE OF BREEZE EFFECT	COUNTRY: ITALY SPONSOR: C.E.C.
TITLE (ENGLISH LANGUAGE):	ORGANISATION: ENEA PROJECT LEADER: P.CAGNETTI, V.FERRARA

INITIATED: Jun.1981	COMPLETED: (expected) 31 Dec.1983	SCIENTISTS:
STATUS: In progress	LAST UPDATING:	

1. GENERAL AIM

   a) To evidentiate correlations in typical land/sea breeze situation between parameters which are readily and inexpensively accessible to measurement and recording (I.E. Solar radiation, insolation and cloud cover, etc.) and those relevant parameters which are more complex to record (I.E.turbulence, boundary layers, etc.).

   b) To predict by an appropriate and simplified model the diffusion into the atmosphere of hypothetical releases of radioactive material for LWR power plants located at a coastal site.

2. PARTICULAR OBJECTIVES
   Studies about a particular diffusion model to predict the environmental consequences of hypothetical accident in sea/land breeze conditions.

3. EXPERIMENTAL FACILITIES

   Data collection at a coastal site both at ground level and in altitude by meteorological instrumentation (net radiometer, anemometers, tethered balloons, etc.). Data elaboration with desk computer (HP-85) and IBM computer.

4. PROJECT STATUS
   Work in progress. Two experimental campaigns on two sampling points at a coastal site (Cangari plain Italy) carried out. Essential results: characteristic profile of wind speed and direction in altitute detected, the height of internal boundary layer evaluated and typical profile of temperature and humidity sampled.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
LOCAL SCALE ATMOSPHERIC DIFFUSION AT COASTAL SITE IN THE PRESENCE OF BREEZE EFFECT	5.4

5. NEXT STEPS

To complete the experimental campaigns (one for every season), to elaborate statistical and experimental data on the coastal site and to set up an appropriate diffusion model.

6. RELATION TO OTHER PROGRAMS

Connected to other European programs within area C of the "Indirect action" of the Commission of the European Communities.

7. REFERENCE DOCUMENTS

Bibliographical review.

8. ADDITIONAL INFORMATION

Personnel: 8 persons
Budget: 226 millions Lit.

		CLASSIFICATION: 5.4 5.5
TITLE (ORIGINAL LANGUAGE): RICERCHE SUI RADIONUCLIDI NELL'AMBIENTE		COUNTRY: ITALY
		SPONSOR: ENEA
TITLE (ENGLISH LANGUAGE): RESEARCHES ON RADIONUCLIDES IN THE ENVIRONMENT		ORGANISATION: ENEA
		PROJECT LEADER: F.GIORCELLI
INITIATED: Jan.1961	COMPLETED:	SCIENTISTS: F.GIORCELLI L.MONTE
STATUS: In progress	LAST UPDATING: April 1982	

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.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
RESEARCHES ON RADIONUCLIDES IN THE ENVIRONMENT	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
Irrigation water	2	Sr^{90}	Quarterly
Milk	11	Sr^{90}, Cs^{137}	Quarterly
Vegetable (1)	1	Sr^{90}, gamma spectr.	Quarterly
Bovine meat	2	Cs^{137}	Quarterly
Whole diet	3	Sr^{90}, Cs^{137}	Quarterly

(1) Brassica oleracea, Lactuca sativa, Lycopersicum esculentum, Malus domestica, Solanum tuberosum

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
   Data referring to 1981 will be available within few months.
   4.2 The present levels of radioactive contamination on country-wide scale, due to fall-out from atmospheric nuclear tests, are very low.

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 Environment Laboratory, ENEA, to the "Reti Nazionali per la Sorveglianza della Radioattività Ambientale" (National Network for the Surveillance of Environmental Radioactivity) co-ordinated by ENEA under auspices of the Italian Health Ministery, with the co-operation of the following Organizations:

   CCR-EURATOM
   (Centro Comune Ricerche Euratom)
   Servizio di Protezione
   ISPRA (Varese)

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
RESEARCHES ON RADIONUCLIDES IN THE ENVIRONMENT	5.4 5.5

CISE
(Centro Informazioni Studi Esperienze)
SEGRATE (Milano)

ENEA
(Comitato Nazionale per la Ricerca e per lo Sviluppo dell'Energia Nucleare e delle Energie Alternative)
Viale Regina Margherita, 125
ROMA

MDA-SERV-METEO-GNMA
(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

ENEA
(Comitato Nazionale per la Ricerca e per lo Sviluppo dell'Energia Nucleare e delle Energie Alternative)
Laboratorio per lo Studio dell'Ambiente Marino
S.TERESA
LERICI (La Spezia)

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.ENEA, Roma

8. Degree of Availability

The series of reports quoted before is available from ENEA (Direzione Centrale per la Sicurezza Nucleare e le Protezione Sanitaria) Viale Regina Margherita 125, 00198 Roma

9. Additional Information

Number of persons involved: 6. Budget (1982): It.L.70.000.000

	CLASSIFICATION : 5.4 5.5
TITLE (ORIGINAL LANGUAGE) : IMPLICAZIONI AMBIENTALI E RADIOPROTEZIONISTICHE DEGLI IMPIANTI NUCLEARI SITUATI IN ZONE COSTIERE	COUNTRY : Italy SPONSOR : ENEA - CCE
TITLE (ENGLISH LANGUAGE) : ENVIRONMENTAL AND HEALTH PROTECTION IMPLICATIONS FROM NUCLEAR PLANTS DISCHARGING INTO COASTAL MARINE ECOSYSTEMS	ORGANISATION : ENEA (°) PROJECT LEADER : A. CIGNA
INITIATED : 1957 | COMPLETED :	SCIENTISTS : Boniforti, Buffoni, Marri, Peroni, Schulte, Scoppa, Zattera, Zurlini
STATUS : In progress | LAST UPDATING : April 1982	

(°) ENEA, S. Teresa (La Spezia)

1. General Aim:

The research programme is essentially related to the need to improve predictive capability for longer-lived radionuclides, whose behaviour now needs to be considered over much longer time scales in order to give effect to the new ICRP emphasis on estimation of collective dose equivalent commitment.

2. Particular Objectives:

The central problems surround the need to examine those physical and chemical factors which determine the initial fractionation of radioactivity and its subsequent transport and distribution within the marine environment. Particular attention will be paid to water, sediments and suspended matter, as these compartments are the prime determinants of any initial distribution and act as reservoirs for uptake of radionuclides into food-chains leading to man. The research programme should also provide a basis for a description of the radiological status of the Italian coastal marine environment with reference to the radioactive fallout from nuclear test explosions and to natural radioactivity. Laboratory experiments under controlled conditions will be carried on to evaluate the uptake and release of important radionuclides in some marine food-chains.
The results obtained will be used to determine the dose equivalent to critical groups and the collective dose equivalent commitment in the areas studied, referring to different release conditions of real or hypothetical situations.

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3. Experimental Facilities and Programme:

- The main task of the Laboratory since its foundation in 1957 has been the study of the marine environment under the health protection view point. Therefore, there is available the basic instrumentation to carry out research in the fields of physical oceanography, chemistry, sedimentology, botany, zoology, microbiology, radioecology, etc.
  In March 1982 the research group moved to a new building. The advantages (space, facilities, aquaria, new personnel, etc.) will considerably improve the efficiency of the work.

- The research programme is actually divided in two projects:
  1) Description and classification of typical marine coastal ecosystems.
  2) Laboratory studies on the behaviour of long-lived radionuclides in the marine environment.

4. Project Status (progress to date and essential results):

Chemical, thermal and radioactive discharges of nuclear power plants have been taken into consideration, according to the title of the programme. Laboratory experiments and mathematical models provided further information on the speciation of heavy metals and chlorine in seawater. Equilibrium concentrations of the most representative chemical species have been calculated by using a thermodynamic approach. Kinetic considerations have been applied to evaluate the persistence of haloamines. It has been shown that thermal discharges have negligible effects on natural populations of primary producers. However, entrainment in cooling circuits of power plants results in high mortality, caused by mechanical, thermal and chemical stresses. Laboratory experiments, carried out simulating temperature conditions occurring in steam condensers, showed that thermal shock can be lethal to a large number of marine organisms. Collection and elaboration of physical and oceanographic data referring to specific areas were directed not only to the study of coastal currents and transport, but also to the acquisition of a better knowledge on the distribution of fission and activation products in sites where nuclear power reactors have been operative for many years.

TITLE (ENGLISH LANGUAGE): ENVIRONMENTAL AND HEALTH PROTECTION IMPLICATIONS FROM NUCLEAR PLANTS DISCHARGING INTO COASTAL MARINE ECOSYSTEMS - 3 -	CLASSIFICATION: 5.4 5.5

The determination of radioactivity in the marine environment has continued to extend the large series of data obtained in the last two decades. Measurements of radioactivity along the Italian coasts have been made on seawater, sediments and a number of organisms relevant to marine food chains. Investigations on the behaviour of radionuclides in the various compartments of the marine environment have been extended to the long-term transfer, by setting up a new subproject dealing with the distribution of metals in sediments and their exchange with seawater.
The global approach of the Laboratory to the evaluation of the impact of nuclear plants on the coastal environment improved the predictive capability needed for the siting of new power plants.

5. Next Steps:

Future work will be in accordance with the general aim and the objectives described under 1. and 2.

6. Relation to other Projects and Codes:

None.

7. Reference documents:

BONIFORTI, R., M. CAMBIAGHI, E. CROCE, P. FRIGIERI, AND R. RUGGIERO 1981
CARACTERISATION DES EAUX INTERSTITIELLES DES SEDIMENTS MARINS
PRESENTED AT: RIUNIONE FRANCO ITALIANA SULLE "CARATTERISTICHE COMPARATE DEI METODI D'ANALISI PER SPETTROMETRIA ATOMICA". 26-27 MAY VALBONNE (ANTIBES). (IN STAMPA).

BONIFORTI, R., M. MEDARO, A. MOAURO, AND R. RUGGIERO 1981
DETERMINATION INSTRUMENTALE PAR ACTIVATION NEUTRONIQUE DES ELEMENTS TRACES DANS LE MATERIEL PARTICULE SEPARE DES EAUX NATURELLES
PRESENTED AT: RIUNIONE FRANCO ITALIANA SULLE "CARATTERISTICHE COMPARATE DEI METODI D'ANALISI PER SPETTROMETRIA ATOMICA. 26-27 MAY, VALBONNE (ANTIBES). (IN STAMPA).

BONIFORTI, R., O. FERRETTI, F. BO, AND P. CAVALLI 1981
ALCUNI RISULTATI PRELIMINARI RIGUARDANTI L'APPLICAZIONE DI UN METODO DI ESTRAZIONE SELETTIVA ALLA ANALISI DI METALLI IN TRACCE IN SEDIMENTI MARINI
CNEN RT/CHI (81) 5..

BRUSCHI, A., G. BUFFONI, A.J. ELLIOT, AND S. MANZELLA 1981
A NUMERICAL INVESTIGATION OF THE WIND-DRIVEN CIRCULATION IN THE ARCHIPELAGO OF LA MADDALENA
OCEANOLOGICA ACTA, 4,3.

TITLE (ENGLISH LANGUAGE): ENVIRONMENTAL AND HEALTH PROTECTION IMPLICATIONS FROM NUCLEAR PLANTS DISCHARGING INTO COASTAL MARINE ECOSYSTEMS - 4 -	CLASSIFICATION: 5.4 5.5

BRUSCHI, A., O. LAVARELLO, C. PAPUCCI, G. RASO, M. RICCOMINI, S. SGORBINI, AND G. ZURLINI 1981
DISTRIBUZIONE DEI RADIONUCLIDI NELL'AMBIENTE MARINO ANTISTANTE LA CENTRALE DEL GARIGLIANO
ATTI DEL XXII CONGRESSO DELLA ASSOCIAZIONE ITALIANA DI FISICA SANITARIA E PROTEZIONE CONTRO LE RADIAZIONI. (IN STAMPA).

ESPOSITO, A. 1981
PROGRAMMA INTERATTIVO MATCROSS. MANUALE PER L'UTENTE.
CNEN RTI/LSAM (81) 1.

ESPOSITO, A., AND G. MANZELLA 1981
AN ANALYSES OF LIGURIAN SHELF CIRCULATION. PART 1. OBSERVATION AND THEORY.
CNEN RT/FI (81) 3, EUR. ART./24.179 .

FOWLER, S.W., G. BENAYOUN, P. PARSI, M.M.A. ESSA, AND E.H. SCHULTE 1981
BEHAVIOUR AND FATE OF TECHNETIUM IN MARINE BIOTA
ATTI 27TH CONGRESS AND PLENARY ASSSEMBLY OF ICSEM, CAGLIARI, 9-13/10/1980. (IN STAMPA).

MAESTRINI, S.Y., AND .M. G. KOSSUT 1981
IN SITU CELL DEPLETION OF SOME MARINE ALGAE ENCLOSED IN DIALYSIS SAC AND THEIR USE FOR THE DETERMINATION OF NUTRIENT-LIMITING GROWTH IN LIGURIAN COASTAL WATERS (MEDITERRANEAN SEA).
J. EXP. MAR. BIOL. ECOL., 50, 1-19. RT/BIO(81) 21

PERONI, C., AND R. RUGGIERO 1981
DETERMINAZIONE DELL' ATTIVITA' MICROBICA E DEL CONTENUTO DI SOSTANZA ORGANICA IN ALCUNI CAMPIONI DI SEDIMENTO RACCOLTI VICINO AL GOLFO DI LA SPEZIA
CNEN RT/BIO (81) 1.

KAMPI, L., AND M. BERNHARD 1981
CHIAVE PER LA DETERMINAZIONE DELLE COCCOLITOFORIDEE MEDITERRANEE
CNEN RT/BIO (81) 13

SCHULTE, E.H., A. SECONDINI, AND P. SCOPPA 1981
TRASFERIMENTO DEL TECNEZIO ATTRAVERSO LE CATENE ALIMENTARI MARINE
ATTI DEL XXII CONGRESSO DELLA ASSOCIAZIONE ITALIANA DI FISICA SANITARIA E PROTEZIONE CONTRO LE RADIAZIONI. (IN STAMPA).

SCOPPA, P., A. SECONDINI, AND E.H. SCHULTE 1981
INDAGINI SULLA STABILITA' DELL'ANIONE PERTECNETATO NELL'AMBIENTE MARINO
ATTI DEL XXII CONGRESSO DELLA ASSOCIAZIONE ITALIANA DI FISICA SANITARIA E PROTEZIONE CONTRO LE RADIAZIONI. (IN STAMPA).

TITLE (ENGLISH LANGUAGE): ENVIRONMENTAL AND HEALTH PROTECTION IMPLICATIONS FROM NUCLEAR PLANTS DISCHARGING INTO COASTAL MARINE ECOSYSTEMS - 5 -	CLASSIFICATION: 5.4 5.5

ANSELMI, B., A. BRONDI, O. FERRETTI, C. PAPUCCI, 1982
ARCIPELAGO DI LA MADDALENA. DATI SUL QUADRO SEDIMENTOLOGICO DELLE PARTICELLE SOTTILI E CORRELAZIONI CON LA DISTRIBUZIONE LOCALE DI ALCUNI RADIONUCLIDI.
ATTI DEL CONVEGNO AIOL - STRESA, 19-22 MAGGIO 1982.

ANSELMI, B., A. BRONDI, O. FERRETTI, C. PAPUCCI 1982
VERIFICA DEL RUOLO ESERCITATO DA I FATTORI GEOMORFOLOGICI E SEDIMENTOLOGICI COSTIERI NEL COMPORTAMENTO DI ALCUNI RADIONUCLIDI. STUDIO SULLA BASE DI RILEVAMENTI NELLE AREE DI: GAETA, LATINA E LA SPEZIA
ATTI DEL CONVEGNO AIOL - STRESA 19-22 MAGGIO 1982.

BERNHARD, M., AND G. BUFFONI 1982
MERCURY IN THE MEDITERRANEAN. AN OVERVIEW
CNEN RT/BIO 82 ..

BONIFORTI, R., A. BRUSCHI, E. FRIGIERI 1982
CONTENUTO DI MN, AI, CO, CR, E SR IN ACQUE MARINE, FLUVIALI E LACUSTRI
ATTI DEL CONVEGNO AIOL, STRESA 19-22 MAGGIO 1982.

BONIFORTI, R., AND A. MOAURO 1982
COMPARISON OF TRACE ELEMENT CONTENT IN MARINE ORGANISMS COLLECTED FROM THE LA MADDALENA ARCHIPELAGO AND OTHER MEDITERRANEAN AND PACIFIC OCEAN SITES
ENEA RT/AMB (82)...

BONIFORTI, R., R. RUGGIERO, F. RO, AND J.C. TOUSSAINT 1982
CHEMICAL AND MINERALOGICAL ANALYSIS OF MARINE LACUSTRINE, AND RIVERINE SEDIMENTS COLLECTED FROM THE ZONE DELIMITED BY LAKE MASSACCIUCOLI AND MESCO CAPE (NORTHERN TYRRHENIAN SEA)
RT/CHI (82)...

CIGNA, A.A. 1982
IL PROBLEMA DEGLI INQUINAMENTI PERSISTENTI
RT/PROT (82)..

ESPOSITO, A., AND G. MANZELLA 1982
CURRENT CIRCULATION IN THE LIGURIAN SEA
ELSEVIER'S PUB. CO. ED. NIHOUL.

PERONI, C., AND G. ROSSI 1982
STIMA DELL'ATTIVITA' MICROBICA IN SEDIMENTI MARINI TRAMITE RIDUZIONE DELLA RESAZURINA
ATTI DEL CONVEGNO AIOL - STRESA 19-22 MAGGIO 1982.

8. Additional information

Personnel: 8 researchers

	CLASSIFICATION: 5.2 5.3 5.4
TITLE (ORIGINAL LANGUAGE): CRITICAL ANALYSIS OF ACCIDENT SCENARIOS AND CONSEQUENCES MODELLING APPLIED TO LWR POWER PLANTS, FOR ACCIDENT CATEGORIES BEYOND THE DESIGN BASIS ACCIDENT	COUNTRY: Italy SPONSOR: CEC
TITLE (ENGLISH LANGUAGE):	ORGANISATION: ENEA PROJECT LEADER: G.PETRANGELI

INITIATED: Nov.1981	COMPLETED: (expected) 1982	SCIENTISTS: C.BROFFERIO P.CAGNETTI V.FERRARA A.MANILIA
STATUS: In progress	LAST UPDATING:	

		CLASSIFICATION : 5:1, 5:3, 5:4, 5:5 och 18
TITLE (ORIGINAL LANGUAGE) : Transuraner i reaktoravfall		COUNTRY : Sweden
		SPONSOR : S.S.I., National Institute of Radiation Protection
TITLE (ENGLISH LANGUAGE) : Transuranic Nuclides in Low-Level Wastes from Power Reactors in Sweden.		ORGANISATION : Dept. of Nuclear Chemistry Chalmers Univ. of Techn.
		PROJECT LEADER : J.O. Liljenzin
INITIATED : 1983-01-12	COMPLETED : 1984-09-30	SCIENTISTS : S. Johnson
STATUS : In progress	LAST UPDATING : 1983-06-01	Tech. ass.: I-M. Ehrlund

	CLASSIFICATION : 5.4
TITLE (ORIGINAL LANGUAGE) : Modeller avseende spridning och exposition i terrestriska ekosystem, dos via livsmedel	COUNTRY : Sweden SPONSOR : National Institute of Radiation Protection
TITLE (ENGLISH LANGUAGE) : Models concerning dispersion and exposition in terrestrial eco systems, dose via food-stuffs	ORGANISATION : Studsvik Energiteknik AB PROJECT LEADER : Ulla Bergström

INITIATED : 1982-07-01	COMPLETED : 1984-12-31	SCIENTISTS : Bo Röjder, Bob Gardner, Ulla Bergström
STATUS : In progress, phase 2	LAST UPDATING : July 1983	

General aim: To make

- a detailed study of involved processes concerning the turnover of nuclides in the biosphere
- a technical review of some selected models and their data bases
- develop a dynamic version of the model
- study ranges and uncertainty interval for the ingoing parameters
- make comparative calculations concerning concentration in food-stuffs and dose burden.

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	CLASSIFICATION : 5.4
TITLE (ORIGINAL LANGUAGE) : Variationsanalys - BIOPATH	COUNTRY : Sweden
	SPONSOR : National Institute of Radiation Protection
TITLE (ENGLISH LANGUAGE) : Variation analysis - BIOPATH	ORGANISATION : Studsvik Energiteknik AB
	PROJECT LEADER : Ulla Bergström

INITIATED : 1982-07-01	COMPLETED : 1983-12-31	SCIENTISTS : Bo Röjder Bob Gardner
STATUS : In progress	LAST UPDATING : July 1983	

1 General aim:

Uncertainty analysis of the BIOPATH code.

2 Particular objectives:

Decisions of

- the total uncertainty for the individual dose burden for a specific eco system
- the contributions from the ingoing parameters to the total uncertainty

3 Programme BIOPATH

BIOPATH - A computer program using compartment analysis for calculation of biosphere turnover and radiation doses

	CLASSIFICATION : 5.4
TITLE (ORIGINAL LANGUAGE) : Kulturväxters transuranupptag	COUNTRY : Sweden
	SPONSOR : SSI National Inst. of Rad. Protection
TITLE (ENGLISH LANGUAGE) : Uptake óf Transuranics by Ciltivated Crops	ORGANISATION : Dep. of Radioecology, Sw. Univ. of Agric. Sciences
	PROJECT LEADER : Åke Eriksson

INITIATED : 1975	COMPLETED :	SCIENTISTS : Åke Eriksson
STATUS : In progress	LAST UPDATING : June 1983	

1. General Aim: Investigation of the transfer of transuranics from agricultural soils to cultivated crops.

2. Particular Objectives: To study the long term aspects, and the effects of complexants on the transfer of plutonium.

3. Experimenatl Facilities and Programme: The investigation is carried out in an open air lysimeter installation with 200 lysimeter vessels of stainless steel, 1 m in height and 0.5 m in diam. 8 respresentative Swedish soils, 2 crops, clover and wheat, and the transuranics Pu-238, Np-237 and Am-241 are used.

4. Project status:
    1. Progess to Date: Data from six years of experimental work have been accumulated and inofficially reported. To be published in 84/85.
    2. Essential Results: A slow decrease with the years in transfer from soil to plants as well as considerable differences between nuclides and between soils and crops have been observed. Complexing agents, DTPA, to a high degree enhanced the Pu-uptake.

5. Next Steps: Continue the experiments and investigate the influence of an increased precipitation level on the relative plant uptake and compare the plant uptake with the extractability of the transuranics from the soil matrix.

8. Degree of Availability: Material is not distributed until publishing.

Mailing address: The Swedish Univ. of Agricultural Sciences
Department of Radioecology
S-750 07 UPPSALA Sweden

	CLASSIFICATION : 5.4 Environmental effects
TITLE (ORIGINAL LANGUAGE) : Sedimentundersökningar utanför Oskarshamn.	**COUNTRY :** Sweden **SPONSOR :** SSI National Institute of Radiation Protection
TITLE (ENGLISH LANGUAGE) : Sediment investigations outside Oskarshamns nuclear power plant.	**ORGANISATION :** SNV The National Swedish Environmental protection boarc **PROJECT LEADER :** Manuela Notter
INITIATED : 5 May 1982	**COMPLETED :** autumn 1983 **STATUS :** in progress	**LAST UPDATING :** June 1983	**SCIENTISTS :**

1. General aim: The investigated area is the bottom outside the nuclear powerplant Simpevarp at the eastcoast of Sweden.
   About 55 sediment samples from the bottom between the swedish coast and out to the island Gotland and south of the island Öland has been sampled.

2. Particular objectives: The aim of the investigation is to estimate the amount of some radionuclides attached to the bottom sediment.

3. Experimental facilities and programme: A Niemistö sedimentsampler has been used. The sediment cores are silced in 2 cm layers, before analyse.

4. Project status: To date, The upper layer 0-2 cm of the sediment has been analysed concerning γ-radiation.
   Essential results: The nuclides Cs-137,134, Sb-125 and Co-60 has bee detected in most samples from the near-lying stations.

5. Next steps: Some supplement sediment samples between Öland and Gotland will be taken in the summer 1983.Deeper layers of some sediment cores are going to be measured to calculate the sedimentation velocity.

6. /
7. /
8. Contact persons: Manuela Notter, Statens Naturvårdsverk, 171 20 Solna.
   Ragnar Boge, Statens Strålskyddsinstitut, Fack, 104 01 Stockholm

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	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: GD KAISER
INITIATED: 1974 | COMPLETED:	SCIENTISTS: BY UNDERWOOD, PJ COOPER, C HARTER, A PICKUP
STATUS: CONTINUING | LAST UPDATING: SEPTEMBER 1980	

GENERAL AIM

To provide a 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, especially the treatment of rain.

3. Automation of input and output.

4. Miscellaneous problems.

5. Sensitivity Studies and Comparative Exercises.

REFERENCES

SRD R62: SRD R63: SRD R85: SRD R120 SRD R134

DEGREE OF AVAILABILITY

Freely

	CLASSIFICATION: 5.4
TITLE (ORIGINAL LANGUAGE): THE CONSEQUENCES OF THE ACCIDENTAL RELEASE OF TOXIC OR FLAMMABLE VAPOURS TO THE ATMOSPHERE	COUNTRY: UK
	SPONSOR: UK HEALTH AND SAFETY EXECUTIVE
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD, UKAEA
	PROJECT LEADER: GD KAISER

INITIATED: 1976	COMPLETED:	SCIENTISTS: CJ WHEATLEY, SF JAGGER
STATUS: CONTINUING	LAST UPDATING: SEPTEMBER 1980	

GENERAL AIM

To develop methods for predicting the consequences of the accidental release of toxic or flammable vapours to the atmosphere.

PARTICULAR OBJECTIVES

To develop and understand models of heavy vapour dispersion.

PROGRESS TO DATE

The computer program DENZ is a completed package for the prediction of the behaviour of puff releases of heavy vapours. The computer program TIRGAS may be used to calculate concentration isopliths from arbitrary sources. The report SRD R154 is a sytematic attempt to classify possible release modes from various forms of containment SRD's work has been successfully applied to the accidental escape of anhydrous ammonia.

NEXT STEPS

1. Development of a model for continuous releases

2. Development of detailed numerical models

3. Study of Chemically interacting plumes

4. Analysis of large scale data which is likely to be available soon

5. Study of transition from deflagration to detonation in heavy vapour clouds

REFERENCES

SRD R150, SRD R152 SRD R154,
Atmospheric Environment

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: SEPTEMBER 1980	

BACKGROUND

It is fundamental that the requirements of the European Directive on radiation protection standards are followed with rega. 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 factor 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 feature 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 analyti 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 furthe 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 evalı 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.

5.5. DETECTION AND MEASUREMENTS

	CLASSIFICATION : 5.4 5.5
TITLE (ORIGINAL LANGUAGE) : IMPLICAZIONI AMBIENTALI E RADIOPROTEZIONISTICHE DEGLI IMPIANTI NUCLEARI SITUATI IN ZONE COSTIERE	COUNTRY : Italy
	SPONSOR : ENEA - CCE
TITLE (ENGLISH LANGUAGE) : ENVIRONMENTAL AND HEALTH PROTECTION IMPLICATIONS FROM NUCLEAR PLANTS DISCHARGING INTO COASTAL MARINE ECOSYSTEMS	ORGANISATION : ENEA (•)
	PROJECT LEADER : A. CIGNA

INITIATED : 1957	COMPLETED :	SCIENTISTS : Boniforti, Buffoni, Marri, Peroni, Schulte, Scoppa, Zattera, Zurlini
STATUS : In progress	LAST UPDATING : April 1982	

		CLASSIFICATION: 5.4 5.5
TITLE (ORIGINAL LANGUAGE): RICERCHE SUI RADIONUCLIDI NELL'AMBIENTE		COUNTRY: ITALY SPONSOR: ENEA
TITLE (ENGLISH LANGUAGE): RESEARCHES ON RADIONUCLIDES IN THE ENVIRONMENT		ORGANISATION: ENEA PROJECT LEADER: F.GIORCELLI
INITIATED: Jan.1961	COMPLETED:	SCIENTISTS: F.GIORCELLI L.MONTE
STATUS: In progress	LAST UPDATING: April 1982	

CLASSIFICATION :
5:1, 5:3, 5:4, 5:5 och 18

TITLE (ORIGINAL LANGUAGE) :
Transuraner i reaktoravfall

COUNTRY :
Sweden

SPONSOR : S.S.I., National Institute of Radiation Protection

TITLE (ENGLISH LANGUAGE) :
Transuranic Nuclides in Low-Level Wastes from Power Reactors in Sweden.

ORGANISATION :
Dept. of Nuclear Chemistry
Chalmers Univ. of Techn.

PROJECT LEADER :
J.O. Liljenzin

INITIATED : 1983-01-12	COMPLETED : 1984-09-30
STATUS : In progress	LAST UPDATING : 1983-06-01

SCIENTISTS :
S. Johnson

Tech. ass.:

I-M. Ehrlund

5.6. DOSES EMANATING FROM RELEASE ACTIVITIES

	CLASSIFICATION : I.5.6
TITLE (ORIGINAL LANGUAGE) : Mätmetod vid friklassning av lågkontaminerat reaktoravfall	COUNTRY : Sweden SPONSOR : Statens Strål-skyddsinstitut
TITLE (ENGLISH LANGUAGE) : Measuring methods for low-contaminated reactor wastes to be released for unrestricted use	ORGANISATION : Studsvik Energiteknik AB PROJECT LEADER : Göran Carleson
INITIATED : 1983-06-01 — COMPLETED : June 1983	SCIENTISTS : Göran Carleson, Hans Tovedal
STATUS : — LAST UPDATING : June 1983	

1. <u>General Aim:</u> Large amounts of low-contaminated or potentially contaminated material or wastes are generated during the ordinary operation of nuclear power stations. Firm and fixed norms as to the applicable radiation levels which would automatically allow an unrestricted use and/or recycling of various types of such wastes are investigated and their implications as to the maximum dose contribution to an individual and to the population determined.

2. <u>Particular Objectives and Experimental Programme:</u> Simple methods will be developed which allow rapid and unambiguous determinations of the average surface dose rate for bulk quantities of various homogeneous and inhomogeneous solid wastes. Empirical connections between bulk activity and surface dose rate for different geometrical waste forms will be established.

	CLASSIFICATION : I.5.6
TITLE (ORIGINAL LANGUAGE) : Dosmätningar vid återcykling av ett friklassat metallparti	COUNTRY : Sweden SPONSOR : Statens Strålskyddsinstitut
TITLE (ENGLISH LANGUAGE) : Dose measurements during the release and recycling of a low-contaminated metallic scrap quantity	ORGANISATION : Studsvik Energiteknik AB PROJECT LEADER : Göran Carleson
INITIATED : During 1983 / COMPLETED :	SCIENTISTS : Göran Carleson
STATUS : / LAST UPDATING : June 1983	

1. General Aim: Large amounts of low-contaminated or potentially contaminated material or wastes are generated during the ordinary operation of nuclear power stations. Firm and fixed norms as to the applicable radiation levels which would automatically allow an unrestricted use and/or recycling of various types of such wastes are investigated and their implications as to the maximum dose contribution to an individual and to the population determined.

2. Particular objective and experimental program: Measurements of dose contributions to the environment and to the individuals involved will be performed during the principal steps of the commercial recycling of a low-contaminated stainless steel scrap quantity released from a reactor station, and compared to earlier theoretical calculations.