COMMISSION OF THE EUROPEAN COMMUNITES

Directorate-General for Research, Science and Education

XII/D/3

# NUCLEAR SCIENCE AND TECHNOLOGY

## European Community
## Water reactor
## Safety Research Projects

## VOLUME II

July 1978

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## European Community
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## VOLUME II

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XII/816/78-EN

# 7. CONTAINMENT AND ASSOCIATED SYSTEMS

| 142-1 -05/4111-01 | | 7 |
|---|---|---|
| **Titre**<br>Etude du comportement du puits de cuve des réacteurs PWR 900 MWe en cas de rupture limitée de cuve. | | **Pays**<br>FRANCE<br>**Organisme directeur**<br>CEA/DSN |
| **Titre (anglais)**<br>Primary shield wall behaviour of PWR's in case of restricted pressure vessel rupture. | | **Organisme exécuteur**<br>CEA/DEMT<br>**Responsable**<br>: (DSN-FAR)<br>(DEMT) |
| Date de démarrage 1/75 | Etat actuel<br>en cours | **Scientifiques** |
| Date prévue d'achèvement 12/80 | Dernière mise à jour 12/77 | |

1 - Objectif général :

L'étude concerne le comportement de structures en béton soumises à l'éclatement de circuit d'eau pressurisée : problème des structures soumises à des impulsions de pression.

2 - Objectifs particuliers :

- L'étude vise essentiellement à déterminer le comportement du puits de cuve (écran biologique ) des réacteurs PWR 900 MWe dans le cas d'une rupture de la cuve du circuit primaire principal.

Du point de vue de la sûreté il est nécessaire de vérifier les points suivants :

- Le puits de cuve doit continuer à assurer le supportage de la cuve.
- Le puits de cuve ne doit pas engendrer de projectiles pouvant mettre en cause l'intégrité de l'enceinte de confinement.

Cette étude devrait permettre de définir des règles et des guides pour juger de la conception du puits de cuve.

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## 3 - Installations expérimentales et programme :

L'étude et la mise au point de l'installation expérimentale ont été confiées au DEMT.
Le programme expérimental comprend les étapes suivantes :

- Essais préliminaires à petite échelle (1/75) pour déterminer les ordres de grandeur correspondant à une rupture brutale de cuve (rupture longitudinale qui correspond au chargement maximum sur le puits de cuve).

- Essais plus représentatifs à une échelle plus importante (1/20) et en prenant en compte des ruptures limitées de cuve.

## 4 - Etat de l'étude :

1) Avancement à ce jour :

- Fin des essais à petite échelle (1/75) sur maquettes en béton armé et précontraint.

- Campagne d'essais pour mettre au point à l'échelle 1/75 des brèches équivalentes aux nouvelles sections de rupture de cuve (10 000, 5000, et 1000 cm2).

- Etablissement d'un avant projet de maquette de circuit primaire au 1/20.

- Calculs préliminaires d'évolution de pression dans le puits de cuve en fonction de la modélisation des structures internes de la cuve.

2) Résultats essentiels :

- La campagne d'essais à petite échelle (1/75) simulant une rupture complète de cuve (rupture longitudinale suivant une génératrice ou rupture circonférentielle) permet de dégager les points suivants :
  - Le puits de cuve en béton armé est complètement pulvérisé et donne lieu à l'émission de projectiles.
  - Le renforcement du puits de cuve par précontrainte circonférentielle externe (frettage) permet d'obtenir des structures capables de résister à ce type d'accident.

- Les calculs préliminaires d'évolution de pression pour l'avant projet au 1/20 du puits de cuve, fonction de la modélisation des structures internes de la cuve, de la localisation de brèche (virole ou fond de cuve) conduisent aux conclusions suivantes :
  - Le degré de modélisation des structures internes (avec ou sans internes) n'apporte pas de différence dans les évolutions de pression.
  - La localisation de la brèche est pratiquement également sans influence sur les pressions.

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5 - Prochaines étapes :

Les étapes prévues en 78 sont les suivantes :

- Construction de la maquette au 1/20 du circuit primaire.
- Etablissement du dossier d'avant-projet de la maquette béton du puits de cuve.
- Construction de maquettes en béton au 1/20.

6 - Relation avec d'autres études :

Néant.

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

| 142-2 -01/4111-6-11 | | 7 |
|---|---|---|
| Titre<br>Piégeage de l'iode dans le béton. | | Pays<br>FRANCE<br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br>Iodine - trapping in concrete. | | Organisme exécuteur<br>CEA/DTech/STA<br>Responsable<br>(DSN-FAR)<br>(DTech / STA ) |
| Date de démarrage 1/75 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement fin 79 | Dernière mise à jour 12/77 | |

1 - Objectif général :

Cette étude a pour but de déterminer l'effet de filtre du béton des enceintes de confinement afin de mieux évaluer les conséquences radiologiques en cas d'accidents de perte de réfrigérant primaire des réacteurs à eau PWR.

2 - Objectifs particuliers :

Déterminer les lois de rétention des produits de fission, en particulier des iodes dans les cas suivants :

- conditions normales de fonctionnement.
- conditions accidentelles (140°C et 4 bars relatifs).

3 - Installations expérimentales et programme :

On utilise le banc d'essai mis au point par la STA pour la mesure des coefficients de perméabilité des bétons dans différentes conditions de pression et de température.

Le programme comprend les étapes suivantes :

- étude quantitative de rétention de l'iode stable (iode pénétrant et moléculaire) en fonction des différents paramètres à prendre en compte (type et épaisseur du béton ; pression et température, points singuliers).

- étude quantitative de la rétention de l'iode actif (éventuellement).

## 4 - Etat de l'étude :

### 1) Avancement à ce jour :

Les principaux travaux réalisés en 1977 sont les suivants :

- montage du dispositif d'analyse des composés iodés ($ICH_3$) mis au point par le service d'études Analytiques
- Conception et réalisation du dispositif d'injection d'iode par la SESTR.
- Adaptation du dispositif d'injection d'iode au perméamètre de la STA à Saclay.
- Essais de réception et de mise au point (cinq) du dispositif d'injection de l'iode dans les conditions de l'accident de perte de réfrigérant primaire (4 bars, 140°C).

### 2) Résultats essentiels :

Pas de résultats sur la diffusion de l'iode ($ICH_3$), car les deux essais préliminaires avec injection d'iode ne peuvent être encore jugés comme représentatifs. (Volume d'eau entraîné au moment de l'injection trop important).

## 5 - Prochaines étapes :

1) Les conditions initiales d'injection étant maintenant atteintes, il reste encore à effectuer des modifications au générateur de vapeur de façon à atteindre les buts suivants :

   - évolution de pression et de température dans la partie supérieure du perméamètre se rapprochant des conditions existantes dans une enceinte après l'accident.

   - régler le débit de vapeur de façon à réduire au maximum le volume d'eau entraîné dans le perméamètre.
     Ce point est considéré comme très important dans la mesure où l'eau de condensation risque de piéger préférentiellement les iodes.

   - Le prochain essai (N° 6) de diffusion d'iode sera effectué sur une éprouvette disposée verticalement.

2) Réalisation par le SEA d'un dispositif de prélèvement d'un échantillon gazeux dans la chambre amont du perméamètre pour vérifier les concentrations d'iode avant diffusion dans le béton.

3) Conception d'un dispositif d'analyse de l'iode moléculaire.

4) Essai de piégeage d'iodure de méthyle par du béton normal (type Bugey) - les essais seront effectués avec les concentrations libérées lors de l'accident de perte de réfrigérant primaire - (LOCA).

5) Essais de piégeage de l'iode moléculaire par du béton normal (type Bugey) - les essais seront également effectués avec les concentrations libérées lors du LOCA.

6 - <u>Relation avec d'autres études</u> :

Néant.

7 · <u>Documents de référence</u> : - rapports internes non disponibles .

| Title 1 : Eléments absorbeurs d'énergie pour la conception des bridages de tuyauteries. | Country BELGIUM |
|---|---|
| Title 2 : Energy absorber elements for pipe whip restraints | Organization TRACTIONEL |
| Initiated : 1972<br>Completed : 1975<br>Status : completed<br>Last updating : - | Project Leader P.HERNALSTEEN |

1. General aim :

Experimental investigation aimed at characterizing the dynamic behaviour of materials used for energy absorber design, under representative geometries and loading conditions. The major part of the programme was related to stainless steel rods acting in tension, and commercial cellular concrete, acting in compression. In addition, several tests were performed on copper bumpers and special mixtures of light weight concrete.

2. Particular objectives :

- Development and testing of a forged head design for anchorage of stainless steel bars

- Comparison of dynamic versus static strength of tested materials, and determination of the scatter in dynamic stress.

- Effect of physical parameters such as humidity and temperature.

3. Experimental facilities and programme :

- Dynamic testing including an air reservoir, rupture disks and a sliding piston (driving force : 40 kN)

- Drop weight machin with a 1 ton hammer.

4. <u>Project status</u> :

The results have been published in the paper : " The use of energy absorbers to protect structures against impact loading 4, P. Hernalsteen and C.Leblois . Nuclear Engineering and Design 37 (1976) 373.406.

5,6,7 -

8. Degree of availability : Contact TRACTIONEL - BRUSSELS.

Classification : 7.1

| | |
|---|---|
| Title 1: Programme VAPON. Evaluation des sollicitations des tuyauteries de vapeur en cas de brèche | Country BELGIUM |
| Title 2 : Programme VAPON. Evaluation of the pipe forces resulting from a steamline break | Organization TRACTIONEL |
| Initiated : January 1977<br>Operational : June 1977 | Project leader E.STUBBE<br>Scientist DUPLAT |

1. General aim :

Pipe restraints are usually installed around steamlines in order to prevent severe pipe movement and pipe whip in case of a severe break. The general aim is to evaluate the hydraulic forces acting on the pipe at different break locations in order to choose the proper locations and dimensions of the pipe restraints.

2. Particular objectives.

Evolution of the pipe force coefficient and mass flow rates during the short time interval following a break of a steam line.

4. Project status

1. Progress to date : The calculations are based on the method of characteristics in order to treat the wave propagation phenomena during the short period following the accident. Step by step calculation of the wave force and the blowdown force gives the evolution of the force coefficient at several locations.

The program can handle pipes equipped by venturis, sudden contractions or expansions, elbows.

The flow regimes extend to the supersonic flow with the possible development of standing shocks in the pipe.

The vapour is treated as an ideal gas and specified by its proper isentropic index. As such, the program can handle any ideal gas.

2. Essential results :

The results obtained are in close agreement with the results from similar codes and published data for similar problems.

5. Next step.

In order to treat the break of high energy lines filled with subcooled water or staturated water and steam, a programme is under development using the same basic method of characteristics in order to evaluate the pipe force coefficient and break mass flow rate.

7. Reference documents.

1. A.H. SHAPIRO " Dinamics and thermodynamics of compressible fluid flow " Ronald 1953

2. F.J.MOODY Time-dependent pipe forces caused by blow down and flow stoppage.
Transactions of the ASME-September 1973

3. P.PANA, J. ROHDE Stationary and transient Mass flow rates and jet thrust forces following pipe breaks
IAEA Meeting COLOGNE 1976

3. Degree of availability

Contact TRACTIONEL-BRUSSELS

Classification : 7.1

| | |
|---|---|
| Title 1 : Programme LOCA-2 : Evolution des pressions à court terme dans les logettes de l'enceinte d'un réacteur PWR en cas d'un accident LOCA | Country BELGIUM |
| Title 2 : Programme LOCA-2 : A computer code to estimate the short term pressurization in the subcompartiments surrounding the primary system, in case of a loss of coolant accident | Organization TRACTIONEL |
| Initiated : June 72<br>Completed : April 73<br>Last update : December 76 (version 3,4,5) | Project Leader E.J. STUBBE |

1. General aim :

   The programme LOCA 2 evaluates conservatively the short term pressure evolution in the subcompartments of a containment following a LOCA or a HELB, in order to ensure the integrity of the concrete structures surrounding the break location.

2. Particular objectives :

   Three versions of the programme exist in order to treat different break locations and a wide variety of interconnected volumes.

   LOCA 2 V 3 : 10 nodes, and 20 interconnections.
   Mainly used for simple geometries. This programs contains a bubble rise option to treat the depressurization of a steam generator.

   LOCA 2 V 4 : 20 nodes, and 60 interconnections.
   Mainly used for calculations of overpressurization in the multiple compartments surrounding the primary and the secondary lines.

LOCA 2 V 5 : 50 nodes and 120 interconnections.
Mainly used for estimating the overpressures in the pressure vessel cavity following a LOCA at the inlet or outlet nozzles of the reactor.

4. Program Status :

1. Progress to date.

The program contains an inertia option for problems where inertia and frictional effects are important, and an orifice option for which quasi steady state compressible flow can be assumed.
The program rigosously treats the thermodynamics of two-phase two-component mixtures of water and an inert gas and contains three flow model options for estimating the critical mass flow rates.

1. Henri-Fauske model (water, vapour+air)
2. Moody model (for water-vapour only)
3. The homogeneous equilibrium model (water, vapour+air)

The effect of water entrainment can be simulated by specifying a water entrainment factor for each interconnection.

A code option is available to simulate fly-out panels and movable plugs between volumes.

2. Essential results

Extensive validation of the program models was performed by comparison of the results on benchmark problems and real configurations obtained from equivalent codes such as TMD, RELAP, COMPRESS, DDIFF.
The results indicate generally good agreement.

5. Next steps :

The code LOCA 2 is presently subjected to an objective validation exercise by participation in the USNRC standard subcompartment problem program. The 13 standard problems are treated and the results are being submitted to the NRC for evaluation.

7. Reference Documents :

D. BROSCHE : ZOCO V, a computer program for the calculation of time and space dependent pressure distribution in reactor containments.
Nuclear Engineering and Design. Vol. 23 (1972)

K.V. MOORE ET AL
Relap-IV : Computer program for transient thermohydraulic analysis. IDO-83401 (1973)

F.J. MOODY : Maximum Flow rate of a single Component, two-phase Mixture.
Transactions of the ASME - February 1965.

R.E. HENRY, H. FAUSKE : The two-phase Critical flow of a one-component Mixtures in Nozzles, orifices and short tubes.
Journal of heat transfer - May 1971.

DDIFF-1 Code : A description of the DDIFF-1 digital computer code for reactor plant subcompartment Analysis.
Combustion Engineering Power Systems CENPD-141 February 1976.

8. Degree of availability : Contact TRACTIONEL-BRUSSELS.

Classification : 7.1

| | |
|---|---|
| Title 1 : PROGRAMME LOCA-3:Evolution de la pression à long terme dans l'enceinte d'une centrale nucléaire suite à une rupture du circuit primaire. | Country : BELGIUM |
| Title 2 : PROGRAMME LOCA-3:Long term pressure evolution in the containment of nuclear power plants, following a loss of coolant accident. | Organization TRACTIONEL |
| Initiated : July 1974<br>Completed : July 1975<br>Last update : January 1976 | Project Leader : E. STUBBE |

1. General aim : To calculate the pressure history in the containment following a loss of coolant accident. The computer code LOCA-3-V4 unables one to :

   a. estimate the maximum pressure for which the containment integrity must be assured ;

   b. estimate a conservatively low containment back pressure to evaluate the efficiency of the ECCS ;

   c. evaluate the efficiency of different safeguard systems (spray, ventilation) ;

   d. evaluate the temperature gradients in the containment structure in order to estimate the stress levels in the concrete.

2. Particular objectives : The code was developped for calculating containment loading for actual power plants. This requires th[illegible]ation of the various components that influence the pressure history such as :

- Detailed description of all passive heat sinks available. The code is dimensionned for a maximum of 10 structures, each of which can contain up to 150 nodes with a variable spacing.

  Four different options are built in to calculate the internal heat transfer coefficient in case of LOCA, two of which are the widely used Tagami-Ushida correlation for integrity and ECCS calculations.

- Simulation of the operational safeguard systems, such as spray and cooling coils.

- Evaluation of the sump water temperature resulting from such sources as the spray, the spill-over flow rates, condensing flux, and the flashing fraction that goes to the sump. This temperature is important to determine the stress in the sump concrete structure and to evaluate the depressurization rate in the recirculation phase.

- During the recirculation phase, a proper evaluation of the temperature of the component cooling water is necessary in order to estimate the heat absorption capacity of the cooling coils and the cooling capacity of the residual heat removal heat exchangers.

4. Project Status :

   1. Progress to date : The fourth version (LOCA-3-V4) is fully operational and provides graphical output for the most important parameters (pressure, temperatures).

2. Essential results : Extensive validation of the code was performed and the results show good agreement with results from other codes such as CONTEMPT, COMPATE, COCO and ZOCO V.

5. Next steps : The code LOCA-3-V4 is continually being updated to follow the evolution in the models used to conservatively estimate the pressure evolution (e.g. FLASH options, including "pressure flash" or "temperature flash").
Work is proceeding to include the treatment of the post-reflood phenomena with FROTH. As the input data are usually given for a fixed downstream pressure, the post reflood mass and energy release rates must be adjusted to actual downstream pressure in the containment.

7. Reference documents :

D. BROSCHE : ZOCO V, a computer code for the calculation of time-and-space dependent pressure distributions in reactor containments.

Nuclear Engineering and Design 23 (1972)

L. RICHARDSON ET AL.
"CONTEMPT", A computer programme for predicting the containment pressure-temperature response to a Loss-of-coolant accident. IDO-17220 (1967).

F. BORDELON ET AL.
Containment Pressure Analysis Code (COCO)
WCAP 8326

8. Degree of availability : CONTACT TRACTIONEL - BRUSSELS

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 7.1 | RS 50 |

| Vorhaben/Project Title | |
|---|---|
| Untersuchung der Vorgänge in einem mehrfach unterteilten Containment beim Bruch einer Kühlmittelleitung wassergekühlter Reaktoren<br><br>Investigation of the Phenomena Occurring within a Multi-Compartment Containment after Rupture of the Primary Cooling Circuit in Water-Cooled Reactors | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: BATTELLE-INSTITUT E.V. Frankfurt am Main<br>Abt. Energietechnik |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| May 4/14, 1971 | June 30, 1978 | Dr. T. F. Kanzleiter |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | December 1977 | 20.275.200,-- DM |

1. General Aim

The objective of this research project is to simulate in large-scale experiments rupture of the primary cooling circuit within the containment of a water-cooled reactor and to investigate the phenomena occurring within the containment. The experimental results are to be compared with the results of model calculations and will finally serve to improve the computer codes.

2. Particular Objectives

Problems to be investigated experimentally:

- Flow rate and jet forces at the site of rupture,
- differential pressure between compartments,
- pressurization in the containment during the LOCA,
- depressurization after the LOCA,
- loads on containment structures.

3. Research Program

3.1 Integral LOCA experiments in a scale-model PWR containment with nine compartments. The volumetric model scale is about 1 : 64 relative to the 1200 MW reactor plant Biblis A.

3.2 Basic LOCA experiments with steam line breaks and a simplified containment geometry.

3.3 Jet force experiments leading to an extremely high load on special concrete structures.

3.4 Additional LOCA experiments.

4. Experimental Facilities, Computer Codes

The experimental facilities consist essentially of

- a model containment (approx. 600 $m^3$, 6 bar),
- a model coolant circuit (approx. 6 $m^3$ 140 bar, 300 °C)
- instrumentation (approx. 200 channels for pressure, differential pressure, temperature, density, mass flow, force, strain and water level),
- data collecting and processing systems with 120 and 256 channels, respectively.

For comparison with the experimental data, the GRS-code ZOCO 6 is used with small modifications; for details see research program RS 50 A. In addition, several external institutions are using the experimental data to verify their own codes.

5. Progress to Date

Ad 3.1 The first series of PWR containment LOCA experiments (Nos.C1 to C16) has been completed in 1976.

Ad 3.2 A series of basic containment LOCA experiments (Nos.D1 to D15) has been performed during January through December 1977. Main features of these experiments are:

- steam line breaks, 150 mm in diameter, 70 bar/285 °C initial conditions, saturated steam flow during the first 2 or 3 seconds, afterwards two phase flow.
- simplified containment geometry: a series of 3 to max. 6 adjoining compartments, which are interconnected alternatively by orifices, nozzles or channels of uniform effective cross sectional flow area.

One of these experiments has been chosen for an international comparison with theoretical results (Containment Analysis Standard Problem CASP).

6. Results

Ad 3.1 An evaluation of jet impingement data from some C-experiments, obtained with an impact plate at the site of rupture (guillotine water line break dia. 100 mm, 290 °C/140 bar) showed that a jet of initially subcooled water can remain in a superheated, metastable state (at least within a compact core) after leaving the nozzle until it reaches the impact plate (installed in 240 mm distance from the pipe nozzle). Thereby the mass flow rate is relatively high (max. 80,000 kg/m$^2$s) and the profile of the local jet force distribution is acute with a maximum in the center up to 55 bar.

If only a slight portion of void (5 or 10 Vol. %) is present already upstream in the inflowing medium, the jet "breaks open" immediately after the rupture cross section and produces a flat wide profile of the jet pressure distribution (max. 6 bar). Thereby the mass flow rate reaches maxima of 40,000 kg/m$^2$s.

Ad 3.2 The analysis of the measured containment pressurization with the ZOCO 6 code indicated - at least in the case of the model containment with its high ratio of internal surface to volume - a significant influence of the heat tranfer and condensation processes even in the short term behavior (2 s). For details see research project RS 50 A. A comparison of experiments with similar compartment configuration but with different vent geometries showed that the discharge coefficent of the sharp edged orifices (dia 750 mm) used in some experiments is approximately 0.7, if for nozzle vents (dia 600 mm) a dicharge coefficient of 1 is assumed.

7. <u>Next Step</u>

Ad 3.2 Evaluation of the results will be continued and additional reports will be prepared.

Ad 3.3
and 3.4 Specification and preparation of additional experiments.

8. <u>Relation to Other Projects</u>

RS 50 A: Analysis of the D-Series Experiments of Research Project RS 50

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## 9. References

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

(5) GKSS 77/I/43:
H. Schwan, "Erste Nachrechnungen einiger Containmentversuche des Forschungsvorhabens RS 50, Reihe D, im Battelle-Institut, Frankfurt am Main". Gesellschaft für Kernenergieverwertung in Schiffbau und Schiffahrt mbH, Geesthacht 1977.

(6) BF-RS 50-21-5
"Beschreibung des Programmpakets RS DV zur Meßdatenverarbeitung", January 1977

(7-9) BF-RS 50-24-3 through 5
"Abschlußbericht Förderungsvorhaben BMFT RS 50 DWR.
Kennwort: Containmentversuche Battelle, Unterauftrag DWR-Versuche".
Vol. 1: "Gebäudeauslegung und theoret. Versuchsbetreuung"
Vol. 2: "Versuche C1 - C12, C14, C16"
Vol. 3: "Versuche C13, C15"
Kraftwerk Union Erlangen, September 1977

(10) BF-RS 50-30-D1
"Quick Look Report Experiment D1", March 1977

(11) BF-RS 50-30-D3 "Quick Look Report Experiment D3", September 1977

(12) BF-RS 50-30-D6 "Quick Look Report Experiment D6", May 1977

(13) BF-RS 50-30-D7 "Quick Look Report Experiment D7", June 1977

(14) BF-RS 50-30-D8 "Quick Look Report Experiment D8", July 1977

(15) BF-RS 50-30-D9 "Quick Look Report Experiment D9", November 1977

(16) BF-RS 50-32-C7 "Vorläufiger Versuchsbericht C7", February 1977

(17) BF-RS 50-32-C14 "Vorläufiger Versuchsbericht C14" February 1977

(18) BF-RS 50-32-D1 "Ergänzende Versuchsdokumentation D1", March 1977

(19) BF-RS 50-32-D3 "Ergänzende Versuchsdokumentation D3", October 1977

(20) BF-RS 50-32-D6 "Ergänzende Versuchsdokumentation D6", May 1977

(21) BF-RS 50-32-D7 "Ergänzende Versuchsdokumentation D7", June 1977

(22) BF-RS 50-32-D8 "Ergänzende Versuchsdokumentation D8", August 1977

(23) BF-RS 50-32-D9 "Ergänzende Versuchsdokumentation D9", November 1977

(24) BF-RS 50-62-5 "LOCA-Experiments with a PWR Multi-Compartment Model Containment". October 1977

10. Degree of Availability of the Reports

Reports are availbale through GRS-FB.
Documents (5) to (23) can be made available only by special agreement.

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 7.1 | RS 50 A |
| Vorhaben/Project Title: Begleitende theoretische Arbeiten zu den D-Versuchen des Forschungsvorhabens RS 50<br><br>Analysis of the D-Series Experiments of Research Project RS 50 | | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: BATTELLE-INSTITUT E.V. Frankfurt am Main<br><br>Abt. Energietechnik |
| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
| May 2, 1977 | June 30. 1978 | Dr. T. F. Kanzleiter |
| Stand der Arbeiten/Status | Berichtsdatum/Last Updating | Bewilligte Mittel/Funds |
| Continuing | December 1977 | 399.850,-- DM |

1. General Aim

Experimental results obtained in RS 50 containment LOCA experiments are to be compared with results of computer code calculations. The objective is to demonstrate, to quantify and to analyze the safety margins inherent in the computer codes of today in order to get a basis for future best estimate codes.

2. Particular Objectives

Analysis of the results of the D-series experiments using available computer codes with and without modifications to support the initial evaluation of experimental results (Quick Look Report) and assist in the detailed planning of following experiments.

3. Research Program

3.1 Calculation of the containment pressurization immediately after the performance of an experiment using the experimental mass flow data. Documentation of the calculational results in comparison with experimental data in Quick Look Reports (combined plots of experiment and computation, remarks, conclusions).

3.2 Analysis of experimental results by
- parametric studies
- variation of program options
- calculation with separated calculational models (e. g. overflow model) using experimental data as boundary conditions.

4. Experimental Facilities, Computer Codes

To evaluate experimental RS 50 data, the GRS-code ZOCO 6 is mainly used with the following modifications:

- original version with additional options for isentropic vent flow and for channel flow.
- special version for separated subcompartment analysis by using additional experimental input data as boundary conditions.
- special version to determine unknown input parameters by continuously controlling the difference between computational results and experimental input data and adjusting the input parameter accordingly.

5. Progress to Date

Ad 3.1 Seven of the 10D-experiments planned for post-calculation have been evaluated. The evaluation of two other experiments is under way.

Ad 3.2 Analysis of experimental data to obtain data for
- heat tranfer coefficients
- discharge coefficients for the vents

have been performed.

6. Results

Ad 3.1 and 3.2 The analysis of the measured containment pressurization with the ZOCO 6-code indicated - especially for the model containment with its high ratio of internal surface to volume - a significant influence of the heat transfer and condensation processes even in the short term behavior (2 s). Using the ZOCO 6-code a good agreement between computational and experimental data can be obtained with the following heat transfer coefficients:

- long, narrow compartments near the break
  - -- with longitudinal flow 10 - 20,000 $W/m^2K$
  - -- with transversal flow 1 - 2,000 $W/m^2K$
- large compartments far from the break 100 - 500 $W/m^2K$

A first preliminary evaluation of data measured with a special heat transfer measuring device (concrete block with thermocouples in different depths) could not confirm heat transfer coefficients higher than 1,000 $W/m^2K$. It is therefore possible that the above mentioned extremely high values contain also a factor which accounts for an unkown additional effect not sufficiently considered by the computer code.

A comparison of experiments with similar containment configuration but with different vent geometries showed that the discharge coefficient of the sharp-edged-orifice-type vents (dia 750 mm) is approximately 0.7 if for nozzle-type vents (dia 600 mm) a discharge coefficient of 1 is assumed.

For the analytical description of the flow through a channel with circular cross section empirical data of Frössel* were used and lead to a good agreement with the experimental results.

## 7. Next Steps

Ad 3.1
and 3.2 Evaluation and analysis of experimental results will be continued and additional reports will be prepared.

## 8. Relation to Other Projects

RS 50: Investigation of the Phenomena Occurring within a Multi-Compartment Containment after Rupture of the Primary Cooling Circuit in Water-Cooled Reactors.

## 9. References

(1) BF-RS 50-30-D3 "Quick Look Report Experiment D3",, September 1977

(2) BF-RS 50-30-D6 "Quick Look Report Experiment D6", May 1977

(3) BF-RS 50-30-D7 "Quick Look Report Experiment D7", June 1977

(4) BF-RS 50-30-D8 "Quick Look Report Experiment D8", July 1977

*Frössel, Strömung in glatten, geraden Rohren mit Über- und Unterschallgeschwindigkeit.Forschung, 7 (2), March/April 1936

(5) BF-RS 50-30-D9 "Quick Look Report Experiment D9", November 1977

(6) BF-RS 50-62-5 "LOCA-Experiments with a PWR Multi-Compartment Model Containment". October 1977

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

Reports are available through GRS-FB

Documents (1) to (5) can be made available only by special agreement.

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 30.11.77 | 7.1 | RS 195 |

| Vorhaben/Project Title | |
|---|---|
| Meßsystem zur Analyse des Dampf-Wasser-Luft-Gemisches in Containment-Überströmöffnungen<br><br>Measuring System to Analyze the Steam-Water Air Mixture in Containment Overflow Openings | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>BATTELLE-INSTITUT E.V.<br>Frankfurt am Main<br>Abt. Energietechnik |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| February 15, 1976 | November 30, 1977 | W. Zirnig |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| completed | November 30, 1977 | 373.150,-- DM |

1. General Aim

Development of a measuring system to determine the composition of the steam-water-air stream in containment overflow openings in LOCA experiments.

2. Particular Objectives

The measuring system is to be used directly in the steam atmosphere of the pressurized containment compartments. It consists of commercial components for density, pressure, and temperature measurement which have to be adapted to the specific conditions of application. The system is to be employed first for experiments in the Battelle model containment under project RS 50.

3. Research Program

3.1 Development of a housing to protect and cool the electronic components of the beta radiation absorption densitometer.

3.2 Selection and calibration of the densitometer and error analysis for the complete measuring system.

3.3 Selection of suitable pressure and temperature transducers by preliminary investigations.

3.4 Construction of an experimental facility to simulate a containment compartment and evaluation of the measuring system under the intended conditions of application.

4. Experimental Facilities, Computer Codes

The experimental facility extended under this project consists of an autoclave with a filling volume of 0.17 $m^3$ which is connected via a pipe to a low-pressure vessel. In this low-pressure vessel - a former reactor air lock of 10 $m^3$ - the test instrumentation can be mounted so as to be easily accessible. In oder to simulate the conditions existing in the containment compartments during a loss-of-coolant accident, a pipe rupture is initiated by breaking a rupture disk. LOCA conditions can be simulated for both a rupture compartment and the adjoining compartments.

5. Progress to Date

Ad 3.1 The housing with covered front side was tested and modified.

Ad 3.2 Preliminary investigations were carried out to determine the best measuring arrangement and the attainable precision.

Ad.3.3 The selected measuring chains were investigated under the intended conditions of application to find out their accuracy.

Ad 3.4 Additional experiments were conducted in the experimental facility to check the results of the improvement of the housing and, after the complete measurement system was installed, a final experiment was run.

6. Results

Ad 3.1 The detector housing with covered beta ray detector proved to be ready for use. The detector is protected by a steel foil of 0.05 mm thickness.

The housing, in which the detector is protected from exposure to steam by air flowing in the opposite direction, needed an air flow rate which would affect the composition of the measured mixture considerably.

With the closed housing the required cooling capacity for protection of the electronic equipment was sufficient. However there is a shift of the density signal caused by fluctuations in temperature inside the housing. Therefore a minor improvement of the housing design is necessary.

Ad 3.2 The strontium-90 source used in this investigation allows a measuring depth of 200 mm. The boundary conditions are: activity of the source 500 mCi, uncertainty < 10 %, time constant

10 ms, density range 1 - 8 kg/m$^3$.

The measuring depth might be enlarged by using a stronger source.

Ad 3.3 The pressure and temperature measurements can be performed as required with the selected transducers and mounting arrangements. The thermal shock and long-time thermal effects on the transducers lead to uncertainties of 1 - 1.5 % full range.

Ad 3.4 Containment blow down conditions with values between 4.2 bar, 136 °C and 6.1 bar, 152 °C were achieved in the low-pressure vessel of the experimental facility. The pressure build up is corresponding to the measurements in a rupture compartment under project RS 50. The final test of the complete measuring system was successful: The mixture density range (depending on the steam content) determined from pressure and temperature measurements was in good agreement with density measurement during the phase of increasing pressure. After the highest temperature in the low-pressure vessel was reached, however, a slight rising of temperature inside the housing led to a significant shift of the density signal.

7. Next Steps

The investigations conducted under this project are concluded. It is considered to use the measuring system in containment experiments (RS 50, RS 0123).

8. Relation with Other Projects

9. References

Final report BF-R-62.994-1

10. Degree of Availability of the Reports

The report is available through GRS-FB.

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 7.1 | PNS 4221 / 4223 |
| **Vorhaben/Project Title** | | **Land/Country** FRG |
| Auslegung, Vorausberechnung und Auswertung der HDR-Blowdown Experimente, sowie Weiterentwicklung und Verifizierung fluid-strukturdynamischer Codes zur Beanspruchung von RDB-Einbauten beim Blowdown<br>Design and Pre-calculation of the HDR-Blowdown-Experiments as well as Development and Verification of Codes in Coupled Fluid-Structural Dynamics for Stress Analysis of Reactor Vessel Internals Under Blowdown Loading | | **Fördernde Institution/Sponsor** BMFT<br>**Auftragnehmer/Contractor** Kernforschungszentrum Karlsruhe GmbH<br>Projekt Nukleare Sicherheit IRE |
| **Arbeitsbeginn/Initiated** | **Arbeitsende/Completed** | **Leiter des Vorhabens/Project Leader** |
| Oct. 74 | 1980 | Dr.Krieg/Dr.Schlechtendahl |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | December 1977 | |

1. General Aim

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

2. Particular Objectives

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

3. Research Program

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

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

3.3 Performance of small scale tests in addition and relation to the HDR-blowdown experiments.

## 4. Experimental Facilities, Computer Codes

Ad 3.1: Hydraulic mechanism for snap-back tests.

Codes in production:

| | |
|---|---|
| TURBIT-2 | (Turbulence, Natural Convection) |
| YAQUIR, STRUYA | (2D fluid dynamics) |
| STRUDL/DYNAL | (finite element structural dynamics) |
| CYLDY2 | (shell dynamics for the core barrel) |
| FLUX-incompressible | (3D fluid dynamics coupled with CYLDY2) |

Ad 3.2 Codes in development:

| | |
|---|---|
| STRUYA/CYLDY2 | (2D fluid dynamics with CYLDY2 structure) |
| FLUX | (3D fluid dynamics with CYLDY2 structure) |
| DRIX-2D | (2D non equilibrium two phase flow) |
| FLUST | (fluid dynamics system code with 2D models) |
| CYLDY3 | (improved shell dynamics for the core barrel) |
| SING-S | (Boundary integral equation code for 3D fluid structure coupling) |

Ad 3.3 Test section "Düse" for steam/water loop of IRB (Karlsruhe).
Fast pressure relief apparatus for autoclave.
Flat water table for analogy experiments.

## 5.+6. Progress To Date and Results

Ad 3.1 The structural behaviour of the core barrel has been analysed with STRUDL/DYNAL and CYLDY2. Results agreed satisfactorily but deviated from results obtained by another laboratory. Development of improved model CYLDY3 was started to clarify the discrepancies. Pressure fields for blowdown were computed with YAQUIR and compared with DAPSY and WHAMMOD results. With the incompressible FLUX code version the snap-back tests have been analysed. Detail plans for the natural convection test (for determining the temperature distribution prior to blowdown) are developed. With the TURBIT-2 code it was shown that radial temperature variation in the downcomer will decay to 25 % due to turbulent exchange. First tests of data transfer from the HDR facility to KFK were initiated.

Ad 3.2 The YAQUIR code and the FLUX code were applied to DWR blowdown analysis. It was shown that the HDR tests give a good representation of the real reactor situation. The DRIX-2D code (based upon the LASL SOLA-DF approach) was developed and successfully applied to detail analysis ot the highly transient region of the blowdown nozzle. A new equation of state permitted complete modelling of downcomer, nozzle flow and free jet stream with FLUST. The incompressible FLUX code development was completed. The 3D FLUX code and 2D STRUYA were successfully coupled to the CYLDY2 structural model of the core barrel.Results were produced for blowdown response of the core barrel with and without coupling. Furthermore, in CYLDY2 a modul has been added for calculation of stresses and strains due to the dynamic deformations. The development of the code SING-S for coupled fluid-structural dynamic problems under any 3D geometry has been started. In this code the fluid dynamics is discribed by an advanced singularity or boundary integral equation method, which leads to modifications of mass and stiffness matrices of the shell structure surrounding the fluid regions.

Ad 3.3 The detail specification of two phase flow tests which are able to determine the essential parameters of the drift flux model was completed. Test equipment has gone into fabrication.

The test conceptions for small scale blowdowns with an autoclave and the fast pressure relief apparatus have been developed in detail. A special goal of these tests is to study the dynamic effects of core mock ups.

7. Next Steps

Ad 3.1 Installation of the HDR core barrel and the natural convection tests are due next year. The natural convection test result will be analysed in order to finalyse the planning of the blowdown test. Predicitve calculations will be carried out with FLUX and STRUYA and will be documented.

Ad 3.2 FLUX (incompressible version) and STRUYA will be documented. Development of the code CYLDY3 will be finished and investigations of the discrepancies between same precalculations for the core barrel dynamics will be carried through. Work will start on integrating the DRIX-2D and FLUX models in FLUST. First applications of the coupled fluid structure dynamic code SING-S to other problems in light water reactor safety (for instance the pressure suppression system) will be made.

Ad 3.3 The two phase flow tests will be performed and analysed.

8. Relation with other Projects

The project is closely related to PNS 4.22.2. It is coordinated with all other projects of the HDR blowdown program.

9. References

R. Krieg, E.G. Schlechtendahl, K.-H. Scholl:
Design of the HDR Experimental Program on Blowdown Loading and Dynamic Response of PWR-Vessel Internals
Nucl. Eng. and Design 43(1977), 419-435

A. Ludwig, R. Krieg:
Dynamic Response of a Clamped/Ring-Stiffened Circular Cylindrical Shell under Non-Axisymmetric Loading
Nucl. Eng. and Design 45(1977), 437-453

R. Krieg, E.G. Schlechtendahl, K.-H. Scholl, U. Schumann:
Full-Scale HDR Blowdown Experiments as a Tool for Investigating Dynamic Fluid-Structural Coupling
SMIRT 4, Session B

E.G. Schlechtendahl, F. Katz, R. Krieg, A. Ludwig, K. Stölting:
2D Fluid FLOW in the Downcomer and Dynamic Response of the Core Barrel During PWR Blowdown
SMIRT 4, Session B

U. Schumann:
Three-dimensional dynamic fluid-structure interactions during reactor-vessel blowdown
Euromech Colloquium No 96
Numerical Analysis of Dynamic Interaction of Structures with Fluids
Swansea, 12th - 15th Sept. 1977

U. Schumann:
Three-dimensional Poisson-solver for reactor-vessel geometry.
2. GAMM-Conf. on Numer. Math. in Fluid Mech., Köln, (1977), S. 192-199

U. Schumann:
Instationäre Potentialströmung in komplexer Geometrie am Beispiel von DWR-Blowdown Strömungen.
KFK-2324 (Dezember 76)

R. Krieg, H. Zehlein:
Coupled problems in transient fluid and structural dynamics with application to nuclear engineering.
Internat. Symposium on Innovative Numerical Analysis in Applied Engineering Science, Versailles, May 23-27, 1977

H. Löffler:
Konstruktion und dynamische Auslegung einer Vorrichtung zum schlagartigen Öffnen eines Druckbehälters.
Diplomarbeit, Univ. Karlsruhe, Institut für Reaktortechnik, 1977

Ferner je zwei Beiträge zu den PNS-Halbjahresberichten

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| | | |
|---|---|---|
| Berichtszeitraum/Period<br>Jan. 1. 1977 - Dec. 31, 1977 | Klassifikation/Classification<br>7.1 | Kennzeichen/Project Number<br>PNS 4222 |
| Vorhaben/Project Title<br>Experimental Data Acquisition and Processing of the Dynamic Behavior of the Pressure Vessel Test Internals in the HDR-Blowdown-Experiments<br><br>Meßtechnische Erfassung und Auswertung des dynamischen Verhaltens der Versuchseinbauten im Reaktordruckbehälter (RDB) des HDR im Rahmen der HDR-Blowdown-Versuche | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Kernforschungszentrum Karlsruhe GmbH<br>Projekt Nukleare Sicherheit, IRE |
| Arbeitsbeginn/Initiated<br>Jan. 1, 1975 | Arbeitsende/Completed<br>1978 | Leiter des Vorhabens/Project Leader<br>K.D. Appelt |
| Stand der Arbeiten/Status<br>Continuing | Berichtsdatum/Last Updating<br>December 1977 | Bewilligte Mittel/Funds |

## 1. General Aim

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

## 2. Particular Objectives

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

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

3. Research Program

The experimental program includes the development, testing and provision of the instrumentation for the planned large-scale experiments as well as the measurements and evaluation. It 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 data for the transducers by laboratory basic tests and dynamic calibration of the original measurement chains prior to their installation in the reactor pressure vessel of the HDR.

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

3.8 Conduct of tests and acquisition of measuring data.

3.9 Evaluation of measured data.

4. Experimental Facilities, Computer Codes

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

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

5. Work Performed

The test facility for the dynamic investigation of transducers (under blowdown conditions) was completely installed at IRE and has started operation.

The operating behavior of the aggregate facility (autoclave system, shaker and auxiliary system), was evaluated and the necessary improvements were made. Moreover, appropriate methods were developed to determine the required data on the thermohydraulic and dynamic behavior of the transducers which are permanently improved.

A report of work on a measurement error correction method for transient measurement signals inclusive of a computer code were completed.

As work went on, four different displacement transducer types and two acceleration transducers were studied in the IRE autoclave testing facility under thermohydraulic conditions similar to that in the reactor pressure vessel of HDR. Another topic of investigations was the acceleration behavior of the transducers (in this case the displacement transducers) and above all the dynamic behavior. The signal, noise and error behavior, respectively, of the transducer was determined. Parallel with these activities appropriate verification methods and computer codes were prepared for the acquisition and evaluation of experimental data (and are used on the existing process computer and TDA 33 analyzer).

A new activity became the supervision of the development work for a high temperature induction type transducer, the first prototypes of which have been delivered to IRE. Preliminary investigations have started.

A proposal was worked out to realize modal investigations at the HDR core barrel, which is to provide knowledge of the vibrational behavior of the HDR core barrel mockup.

6. Results

First investigations of acceleration transducers, above all induction type displacement transducers, were performed with the shaker and the autoclave system. In the investigations of induction type displacement transducers, on which the attention had been concentrated, one transducer type survived all the thermo-hydraulic and mechanical dynamic tests and exhibited an error behavior which makes it suitable for the anticipated tests. All the rest of prototype transducers did not qualify and were returned to the supplier firms for modification. The causes of errors were: non-reproducible behavior because of alterations in the insulation caused by thermal stresses and high noise levels during accelerations at the transducer casing.

The studies of the acceleration transducers had to be discontinued because of defects occurring at the transducers (loss of insulation resistance). New prototypes have been ordered.

The first investigations of induction type high temperature, pressure and differential pressure transducers do not yet allow statements to be made about the suitability of these transducers.

7. Plans of Future Work

Investigations of the modified and new prototype displacement transducers and acceleration transducers, respectively, will be continued and terminated, respectively, with the objective of their qualification so that the best suited transducers can be ordered as soon as possible with the suppliers. The same applies to the high temperature, pressure and differential pressure transducers, with various modifications certainly still required in this case in the course of studies.

The planned high temperature strain gauges have mainly been studied at MPA of Stuttgart University; a final test on the dynamic behavior will be performed at IRE.

8. Relation with Other Projects

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

9. Literature

Appelt, K.D.:
"Aufbau und Übertragungsverhalten der verschiedenen Meßketten der HDR-RDB-Instrumentierung nach dem derzeitigen Kenntnisstand "
Interner Arbeitsbericht Nr. IRE 3/4222/119/76, PNS-Bericht Nr. 104/76

Eberle, F.:
"Beitrag zur Entwicklung von Meßfehlerkorrekturverfahren für transiente Meßsignale "
Interner Arbeitsbericht Nr. IRE 3/4222/137/76

1. Halbjahresbericht,PNS 1/77
Kadlec, J.:
"Realisierungsvorschlag für die Modaluntersuchungen am HDR-Modell-kernmantel mit der PRODERA-Anlage "
Interner Arbeitsbericht Nr. IRE 3/4222/182/77

Appelt, K.D., Eberle, F., Lang, G., Philipp, P.:
"Untersuchung des Hochtemperatur-Beschleunigungsaufnehmers Type Z 10315 von der Fa. Kistler zur Qualifizierung für die HDR-Instrumentierung "
Interner Arbeitsbericht Nr. IRE 3/4222/201/77

2. Halbjahresbericht, PNS 2/77

10. Degree of Availability
Unrestricted distribution

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. 77 - 31. 12. 77 | 7.1 | RS 93 A |

| Vorhaben/Project Title | |
|---|---|
| Weiterführung der Untersuchungen über die Auswirkungen des Ausströmens von Dampf-Wasser-Gemischen aus Rohrleitungslecks<br><br>Investigations on critical two-phase flow with regard to transient, scaling and subcooling effects | Land/Country<br>FRG<br><br>Fördernde Institution/Sponsor<br>BMFT<br><br>Auftragnehmer/Contractor<br>KRAFTWERK UNION AG<br>Reaktortechnik<br><br>R 513, Erlangen |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1. 2. 76 | 31. 12. 77 | W. Kastner |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | 31. 12. 77 | 456.400,-- DM |

1. General Aim

The theoretical models for calculations of critical mass flow rates and thrust forces have to be checked relating to the influence of subcooling and transient start-up procedures at discharging from pipe leaks. Investigations for applying this models up to pipe diameters of 800 mm have to be done, too.

2. Particular Objectives

When the calculation models have been confirmed, the thrust and jet forces on baffle plates of two-phase jets can be determined.

3. Research Program

3.1 Experimental investigations to the influence of subcooling on steady-state critical two-phase flow

3.2 Theoretical study on the difference of steady-state versus transient critical two-phase flow with respect to flow rates and jet forces

3.3 Theoretical study on scaling of experimental results on critical two-phase flow up to reactor coolant size.

4. Experimental Facilities

Preliminary experimental tests will be carried out at a test facility with 10 mm discharge diameter. Then some tests will be done at a big facility, to investigate the influence of discharge diameters up to 50 mm.

5. Progress to Date

Two literature studies were completed:

a) A theoretical study on scaling of experimental results on critical two-phase flow to full-scale plants

b) A theoretical study on the difference of steady-state versus transient critical two-phase flow with application of theoretically calculated and published experimental data for critical mass flow and reaction forces during the blowdown-phase.

Several tests with subcooled fluid were done at the test facility with 10 mm discharge diameter. The results were evaluated.

6. Results

For pipe length > 300 mm certain of the theoretical models agree well with the measurements (homogeneous phase distribution, thermodynamic equilibrium). This agreement persists irrespective of pipe diameter. The applicability of experimental data to full-scale plants then is successful.

The theoretical model for calculations of the transient behaviour of critical mass flow rates and reaction forces describes the experimental properties rather good. The following conditions have to be met:

| | | |
|---|---|---|
| stagnation pressure | $\geq$ | 20 bar |
| pipe length | $\geq$ | 30 cm |
| pipe diameter | $\geq$ | 25 mm |

The measured critical flow rates, jet and thrust forces at the facility with 10 mm discharge diameter agree well with experimental data from the literature and a modified Bernouilli-model.

7. Next Steps

The test facility with discharge diameters up to 50 mm will be modified to reduce the hydraulic resistance of the discharge pipe. Then measurements will be done.

8. Relation with Other Projects

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9. References

Abschlußbericht RS 93 A, Teil 1 (Jan. 1977)

10. Degree of Availability

Available from GRS-FB

Classification: 7.1 7.2

| | |
|---|---|
| Title: | Country: DENMARK |
| | Sponsor: Risø National Laboratory |
| Title: MACON. A Containment Multiroom Transient Analysis Code. | Organization: Risø National Laboratory |
| Initiated date: 1975 Completed date: <br> Status: Being tested. | Scientists: V.S. Pejtersen, M.B. Andersen |

## 1. General aim

Development of a multiroom containment computer code for pressure and temperature transients during a loss-of-coolant accident.

## 2. Particular objectives

The code, MACON, written in FORTRAN IV for the Burroughs B6700 computer, is developed to predict the behaviour of water-cooled nuclear reactor containments subjected to loss-of-coolant accidents. It can also be used to describe responses in experimental containments.

It calculates the time variation of room pressures, temperatures, mass and energy inventories of air, steam and water, mass and energy exchange between phases and adjacent rooms, too. Heat exchange between room constituents abd internal as well as external (wall) structures.

The model is one-dimensional, having separate continuity equations for air/steam/water, common momentum equation, but separate energy equations for air/steam and water. Air and steam are completely mixed and in thermal equilibrium, while thermal non-equilibrium may be experienced by steam and water. A drift flux model is used, together with suitable constitutive equation for interfacial mass and energy transfer and heat transfer to and from structures.

3. Experimental facilities and programme

The code is being tested against available experimental data from the Marviken MXI and MXII experiments.

4. Project status

A one-dimensional version is being tested, A sparse technique for solution of the equation system has been used with a speeding up effect on the calculation time.

5. Next system

Heat conduction in structures is to be coded in. Wetwell pool condensation and water level are to be modelled and built in.

6. Relation with other projects

The code is in some respects based on the experience gained during the Marviken experiments as well as experience with the CONTAC code.

7. Reference documents

No reports available yet.

8. Degree of availability

| | |
|---|---|
| Classification: 7.1, 7.2 | |
| Title: | Country: DENMARK |
| | Sponsor: Risø National Laboratory |
| Title: CONTAC-II. A containment transient analysis code | Organization: Risø National Laboratory |
| Initiated date: January 1969 Completed date: Feb. 70<br>Status: In use | Scientists:<br>Aksel Olsen<br>N. Kjær-Pedersen<br>V.S. Pejtersen |

## 1. General aim

Development of a computer code for the calculation of pressure- and temperature transients in a PS-containment during a loss-of-coolant accident.

## 2. Particular objectives

The present code, CONTAC II, is written in FORTRAN IV for the Burroughs 6700 computer, and it is a revised version of the CONTAC-code written for the IBM-7094 computer.

It calculates the transient pressures and temperatures in the two compartments of a PS-containment following a rupture of the primary system. The blow down mass flow rate and corresponding enthalpy may either be supplied to the code as input functions of time or be calculated internally in a one-node vessel representation using Moody's model for critical two-phase flow.

The code uses one-node representation of drywell and wetwell. Core spray, drywell spray, wetwell spray and wetwell pool cooling as well as heat transfer from and to walls and structures are incorporated.

## 5. Experimental facilities and programme

The CONTAC II code was used during the planning and experimental phase of the Marviken containment experiments, and a comparison between experimental data and calculations were undertaken and is reported in the Marviken report, MXA-2-205.

## 4. Project status

A comparison between a number of calculations and corresponding data from the Bodega Bay, Humboldt Bay and Marviken experiments show an overprediction of the maximum drywell pressure ranging from 5 to 25 per cent.

## 5. Next steps

A new updated version, CONTAC III, including an improved vent flow model, was completed in 1975.

## 6. Relation with other projects

## 7. Reference documents

N. Kjær-Pedersen:
CONTAC II, A Containment Transient Analysis Code
Risø, RD-Memo nr. 47 (1972)

N. Kjær-Pedersen:
SIMPLI II, A Simulation Program Using Implicit Integration
Risø, RD-Memo nr. 46 (1972)

## 8. Degree of availability

Available.

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| Classification: 7.1 7.2 | |
|---|---|
| Title: | Country: DENMARK |
| | Sponsor: Risø National Laboratory |
| Title: CONTAC-III. A containment transient analysis code. | Organization: Risø National Laboratory |
| Initiated date: Completed date: March 1975<br>Status: Tested. | Scientists: V.S. Pejtersen<br>F. Cortzen<br>K.L. Thomsen |

## 1. General aim

Development of a computer code for the calculation of pressure- and temperature transients in a PS-containment during a loss-of-coolant accident.

## 2. Particular objectives

The present code, CONTAC III, is written in FORTRAN IV for the Burroughs 6700 computer, and it is a revised version of the CONTAC-II code.

It calculates the transient pressures and temperatures in the two compartments of a PS-containment following a rupture of the primary system. The blow down mass flow rate and corresponding enthalpy may either be supplied to the code as input functions of time or be calculated internally in a one-node vessel representation using Moody's model for critical two-phase flow.

The code uses one-node representation of drywell and wetwell. Core spray, drywell spray, wetwell spray and wetwell pool cooling as well as heat transfer from the walls are incorporated (as one-dimensional heatconducting discretized structures, two for each node). A revised version of the vent-flow correlation is incorporated.

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## 3. Experimental facilities and programme

The CONTAC III code has been tested against the Marviken containment experiments, and a comparison between experimental data and calculation has been made.

## 4. Project status

In spite of improvements to the vent flow model, the predictions using CONTAC-III show no marked improvement compared to CONTAC-II. This is undoubtedly due to the assumption of homogeneity resulting in a unrealistic water-carry-over.

## 5. Next steps

Problems concerning the water-carry-over in vent flow is being considered.

## 6. Relation with other projects

## 7. Reference documents

N. Kjær-Pedersen:

CONTAC II, A Containment Transient Analysis Code
Risø, RD-Memo nr. 47 (1972)

N. Kjær-Pedersen:

SIMPLI II, A Simulation Program Using Implicit Integration
Risø, RD-Memo nr. 46 (1972).

## 8. Degree of availability

| 142-1 -C4/4111-06 | | 7.1 |
|---|---|---|
| Titre<br>Comportement local des enceintes en béton sous l'impact d'un projectile rigide. | | Pays<br>FRANCE<br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br>Local behaviour of reinforced concrete walls under hard missile impact. | | Organisme exécuteur<br>CEA/DEMT-CEA/CESTA<br>Responsable<br>(DSN-FAR)<br>(DEMT) |
| Date de démarrage 1974 | Etat actuel en cours | Scientifiques |
| Date prévue d'achèvement 1979 | Dernière mise à jour Décembre 77 | |

1 - Objectif général :

- Cette étude est destinée à mieux faire connaître la tenue d'une paroi en béton armé sous le choc d'un projectile. Les conditions de résistance limite sont systématiquement recherchées afin de permettre la mise au point d'une formule empirique permettant d'évaluer la tenue à la perforation de différentes structures (NB : le projectile est "dur").

- Cette étude doit permettre également de mettre au point un programme de calcul utilisable dans un code aux éléments finis.

2 - Objectifs particuliers :

1) Calculs d'évaluation préliminaires.
2) Recherche de conditions typiques pour maquettage.
3) Etude de l'influence du diamètre et de la masse de projectile.
4) Etude de l'influence du ferraillage.
5) Cas représentatifs de cas réels.
6) Influence de différents rapports géométriques.
7) Influence des caractéristiques mécaniques des parois.
8) Condition extrême (très grande vitesse).
9) Forme du projectile.

.../

3 - Installations expérimentales et programme :

- Les essais sont réalisés au Centre d'Etudes Scientifiques et Techniques d'Aquitaine (CESTA) à l'aide d'un canon à air comprimé de Ø 300 mm sur des dalles de 1,46 x 1,46 m.
L'utilisation d'un sabot en bois à l'arrière du projectile permet d'obtenir les possibilité suivantes :
$100 < \emptyset < 300$ mm
$15 < M < 300$ kg

Dans chacun des essais on cherche à obtenir une vitesse d'impact voisine de celle qui est juste nécessaire pour perforer la dalle en béton.

- Le programme proprement dit a pour but, dans un premier temps, d'établir une formulation permettant de relier la vitesse critique (vitesse d'impact minimale pour laquelle le projectile traverse la dalle) aux caractéristiques de la dalle (épaisseur, densité et répartition du ferraillage, résistance du béton) et du projectile (masse, diamètre et vitesse).

- L'étape suivante (fin 79) consistera à mettre au point un programme de calcul utilisable dans un code aux éléments finis.

4 - Etat de l'étude :

1) Avancement à ce jour :

Les essais déjà effectués (une cinquantaine) avaient pour but d'étudier l'influence des paramètres suivants :

- épaisseur des dalles.
- masse et diamètre du projectile.
- densité et répartition du ferraillage.
- vitesse du projectile.

2) Résultats essentiels :

- L'ensemble des résultats expérimentaux permet d'établir la relation donnant la vitesse critique de perforation :

$$V_c^{\frac{3}{2}} = 1,7\ \sigma . \rho^{\frac{1}{3}} \left( \frac{\emptyset\ e^2}{M} \right)^{4/3}$$

avec $\sigma$ : résistance en compression du béton
$\rho$ : densité du béton
Ø, M : diamètre et masse du projectile.
$e$ : épaisseur de la dalle en béton

- Le domaine de validité de la vitesse de juste perforation est le suivant :
  - ferraillage du béton 150 à 300 kg/m3
  - résistance en compression du béton 300 à 500 bars
  - vitesse du projectile $< 200$ m/s

5 - Prochaines étapes :

Les prochaines étapes visent les buts suivants :

1 - au niveau de la formulation proprement dite :

- Influence de la résistance du béton pour apprécier la fragilisation due au vieillissement.

- Influence de la section du projectile : cas particulier du projectile turbine pour évaluer le diamètre équivalent à une section rectangulaire donnée.

- Influence de vitessessupérieuresà 200 m/s.

2 - au niveau des modèles de calcul :

- mise au point d'un modèle béton à introduire dans un code aux éléments finis pour calculer la pénétration et la perforation de dalles en béton non armé tout d'abord.

- deuxième étape ensuite avec prise en compte du ferraillage.

6 - Relation avec d'autres études :

L'étude CEA-DSN est en liaison directe avec les essais effectués par EdF sur des dalles en béton armé de 5 x 5 x 0,4 m.

La relation concernant la vitesse critique de juste perforation donnée au § 4.2 tient compte des résultats des essais EdF.

- Documents de référence disponibles :

- Communication au 4ème SMIRT de San Francisco

Cette communication sera disponible sous forme de rapport DSN et sera publiée également dans Nuclear Engineering an Design .

| 145-1 -08/4160-05 | | 7.1 |
|---|---|---|
| **Titre** Conséquences d'un LOCA sur l'enceinte de confinement. Etude de la condensation sur un mur d'un mélange air-vapeur dans des conditions transitoires. Programme ECOTRA. | | **Pays** FRANCE |
| | | **Organisme directeur** CEA/DgCS - EDF/SEPTEN |
| **Titre (anglais)** Study on the condensation on a wall of an air steam mixture in transient conditions in a LOCA accident. ECOTRA project | | **Organisme exécuteur** CEA/DTCE-STT(GRENOBLE |
| | | **Responsable** (STT) STT - Grenoble |
| **Date de démarrage** 01/01/76 | **Etat actuel** en cours | **Scientifiques** |
| **Date prévue d'achèvement** 31/12/78 | **Dernière mise à jour** 1/78 | |

1 - Objectif général :

Etude des transferts de chaleur dans le cas de condensation de vapeur sur un mur en présence d'air dans des conditions transitoires afin de pouvoir déterminer l'évolution de la pression dans l'enceinte d'un réacteur au cours de l'accident de perte de réfrigérant primaire.

2 - Objectifs particuliers :

Développement de modèles physiques pour l'interprétation des expériences.

3 - Installations expérimentales et programme :

Installation ECOTRA : Une section d'essai d'un diamètre de 25 cm est brutalement mise en contact avec un mélange air-vapeur. La mesure des températures internes et externes de la section d'essai permettra de déterminer le coefficient d'échange.

.../

4 - Etat de l'étude :

1) Avancement à ce jour

Installation opérationnelle depuis Avril 1977.
Essais en cours sur une section d'essai en acier inoxydable.

2) Résultats essentiels

Influence de la vitesse, de la pression et de la teneur en air sur les coefficients d'échanges de condensation.

5 - Prochaines étapes :

Etude de l'influence de l'orientation de la plaque.
Essais sur une section d'essai en béton.

| 145-1 -09/4151-11 | 7.1 |
|---|---|

| | | |
|---|---|---|
| Titre<br>Conséquences d'un LOCA sur l'enceinte de confinement.<br>Etude de l'écoulement du brouillard entre les casemates d'une enceinte de confinement : programme REBECA. | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA /DgCS - EDF/SEPTEN |
| Titre (anglais)<br>Mist flow between subcompartments of a containment : REBECA project. | | Organisme exécuteur<br>CEA/DRE - STRE Cadarache<br><br>Responsable<br>-id- |
| Date de démarrage 01/01/75 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 31/12/81 | Dernière mise à jour 1/78 | |

1 - Objectif général :

Etudier les écoulements d'un mélange eau-air-vapeur afin de déterminer la mise en pression des casemates de l'enceinte d'un réacteur pressurisé après l'accident de perte du réfrigérant primaire.

2 - Objectifs particuliers :

Développement de modèles physiques pour l'interprétation de l'expérience (modèle d'écoulement axial).

3 - Installations expérimentales et programme :

Installation REBECA : le mélange réalisé dans un mélangeur est envoyé dans une section d'essai qui peut être une tuyère ou un diaphragme.

Paramètres importants : pressions amont et aval, titres vapeur et air, tailles des gouttes.

.../

4 - Etat de l'étude :

1) AVancement à ce jour

- Etude du mélangeur au moyen de l'expérience TUYERE et mise au point de l'instrumentation. (écoulement en eau, air et eau/air)
  Essais en eau et air réalisés.

5 - Prochaines étapes :

Essais en eau/air sur l'expérience TUYERE.
Construction au cours de 78.
Début des essais,fin 78.

6 - Relation avec d'autres études :

Etudes théoriques sur l'écoulement entre casemates dans une enceinte de confinement.

7 - Documents de référence: rapports internes non disponibles

CLASSIFICATION
7.1

| | | |
|---|---|---|
| TITLE 1 | CALCUL DE LA PRESSION DANS L'ENCEINTE DE CONFINEMENT LORS D'UNE RUPTURE DE TUYAUTERIE PRIMAIRE (PWR). | COUNTRY FRANCE<br>SPONSOR E.D.F. SEPTEN<br>ORGANIZATION E.D.F. |
| TITLE 2 | CALCULATION OF THE PRESSURE IN THE CONTAINMENT, FOLLOWING A LOSS OF COOLANT ACCIDENT (PWR). | Project Leader<br>E.D.F./SEPTEN/T |
| Initiated 1973 | Completed 1974 | Scientists<br>H. ROUX. |
| Status Codes opérationnels | Last updating : 20.01.75 | |

I - GENERAL AIM

• Dimensionnement de l'enceinte de confinement des réacteurs PWR.

II - PARTICULAR OBJECTIVES

L'analyse des sollicitations mécaniques et thermiques des enceintes de confinement des tranches nucléaires PWR en cas de rupture de tuyauterie primaire ou secondaire est effectuée à l'aide des codes développés à E.D.F. :

PAREO 5 : Calcul des évolutions de pression et de tuyauterie dans l'enceinte à court terme et à long terme en phase de recirculatio

CATEN 6 : Calcul des évolutions de température et pression à l'intérieur des divers compartiments de l'enceinte pendant les premiers instants suivant la rupture.

FORMON : Calcul des forces appliquées aux parois extérieures de la cuve en cas de rupture de tuyauterie dans le puits de cuve.

III - EXPERIMENTAL FACILITIES AND PROGRAMME

Néant.

| | |
|---|---|
| | Classification : 7.1 |
| **Title 1** (original language)<br><br>Thermal shock experimental program | Country : FRANCE<br>Sponsor : EDF FRAMATOME<br>Organization<br><br>FRAMATOME |
| **Title 2** (english)<br><br>Thermal shock experimental program | **Project leader :**<br>Mr. DOYEN<br><br>**Scientists :** |
| Initiated (date): 1975<br>Completed (date): 1977<br>Status: PROGRESSING<br>Last updating (date): | |

## 1. GENERAL OBJECTIVE

The main objective of this program is to gain a validation of LEFM (Linear Elastic Fracture Mechaniely) applied to a reactor vessel core shell submitted to thermal stress similar to those occuring after a LOCA. At the end of the reactor life.

## 2. PARTICULAR OBJECTIVES

2.1 Validation of LEFM

2.2 Test a SA 508 cl 3 shell with suberitical cracks and critical cracks with liquid nitrogen in order to verify the crack ini--tiation and the crack arrest conditions.

## 3. PROJECT STATUS

Some preliminary tests have been carried out for determining the heat transfer coefficient between a steel and liquid nitrogen.

Based on experimental data, finite elements calculations have been performed for determining the crack size which is susceptible to propagate under thermal stress induced by the cool down of the internal surface of the test piece by liquid nitrogen. The test piece is ordered.

## 4. NEAR TERM PLANNING

A detailed fracture toughness characterization of the material will be done prior to perform thermal shock test in order to determine the crack sizes.

## 5. RELATION WITH OTHER PROJECTS

Complementary of HSST program (thermal shock tests) carried out by ERDA (USA).

## 6. AVAILABILITY OF "RESULTS"

Property of FRAMATOME.

Classification : 1.1.2 / 7.1

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

| PROJECT TITLE : Development of a code for transient analysis | CLASSIFICATION 7.1 |
|---|---|
| SPONSORING COUNTRY : ITALY | ORGANISATION : NIRA-Cirene |
| DATE INITIATED : 1974<br>DATE COMPLETED : end of 1976 | PROJECT LEADER : NIRA |

Description :

Scope of the work is the development of the TILT code for the analysis of the transient following a LOCA in a pressure tube heavy water reactor. The work is performed in the framework of thecontract awarded from CNEN to NIRA for the design of the CIRENE prototype Nuclear Island.

Reference documents

NIRA Report
"Il Programma di calcolo TILT NIRA" - 01300 - RT - 19

| | |
|---|---|
| ENERGIEONDERZOEK CENTRUM NEDERLAND | CLASSIFICATION : 7.1 |
| TITLE : CHARME-IM Berekening van de impulsbelasting door uitstroming bij pijpbreuken | COUNTRY: THE NETHERLANDS |
| | SPONSOR : ECN |
| | ORGANIZATION : ECN |
| TITLE (ENGLISH LANGUAGE): CHARME-IM Blowdown jet impingement process | PROJECTLEADER : Speelman, J.E. |
| INITIATED : 1976 / LAST UPDATING : May 1978 | SCIENTISTS : Putten, A.P.W.M. van der; Bogaard, J.P.A. van den; Koning, H. |
| STATUS : progressing / COMPLETED : 1980 | |

General aim

A special subroutine as part of the CHARME code is developed, to calculate the fluid parameters in a jet beyond the system opening in view of the determination of the impact forces of this jet striking a wall.

Particular objectives

The jet subroutine of CHARME was modified in order to incorporate the geometrical aspects of the wall and the deviating impingement jet.

Experimental Facilities and program: none

Project status

The development of a two-dimensional jet model has been started.

Next steps: Introduction of special geometries

Relation with other projects: See CHARME-code (ECN)

Reference documents: -

Degree of availability: Upon mutual agreement at ECN-Petten

Budget:

Personnel:

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| ENERGIEONDERZOEK CENTRUM NEDERLAND | | CLASSIFICATION : 7.1 |
|---|---|---|
| **TITLE :** CHARME-DIS<br>Berekening van het thermohydraulische proces in de uitstroomleiding van veiligheidskleppen | | COUNTRY: THE NETHERLANDS |
| | | SPONSOR : ECN<br>ORGANIZATION : ECN |
| **TITLE (ENGLISH LANGUAGE):** CHARME-DIS<br>Determination of thermohydraulic process in safety relief discharges pipes | | PROJECTLEADER : Speelman, J.E. |
| INITIATED : June 1976 | LAST UPDATING : May 1978 | SCIENTISTS : Putten, A.P.W.M. van der<br>Bogaard, J.P.A. van den<br>Koning, H |
| STATUS : Progressing | COMPLETED : 1980 | |

General aim

The development of a calculation model to predict the thermohydraulic process in a discharge pipe of a safety relief valve.

Particular objectives

The opening of safety relief valves may lead to pressure waves in the discharge lines and back flow from the containment pressure suppression tank. This back flow may lead to underpressure in the containment suppression tank. Special subroutines for the CHARME-code will be developed to study this phenomenon.

Experimental facilities and program: Not foreseen

Project status:

A subroutine to calculate shockwaves has been developed.

Next steps

Calculators on special applicators will be performed.

Relation with other projects: See CHARME-code (ECN)

Reference documents: -

Degree of availability: Not yet applicable

Budget: -

Personnel: -

847

| | |
|---|---|
| TNO-IBBC | CLASSIFICATION : 3.2.3.3.7.1 |
| **TITLE :** Responsieberekeningen voor reactorgebouw | COUNTRY: THE NETHERLANDS |
| | SPONSOR : Ministry of Social Affairs; Ministry of Public works (DEF); TNO-IBBC |
| | ORGANIZATION : TNO-IBBC |
| **TITLE** (ENGLISH LANGUAGE): Dynamic response of reactor structures (building and containment) | PROJECTLEADER : Kusters |
| INITIATED : June 1974 — LAST UPDATING : June 1978 | SCIENTISTS : Kusters, de Groot, de Witte |
| STATUS : Progressing — COMPLETED : 1979 | |

General aim

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

Classification

7.1

| Title 1 | COUNTRY |
|---|---|
| EXPERIMENTAL WORK TO EXAMINE THE EFFECT OF MISSILE IMPACT ON REACTOR CONTAINMENT STRUCTURES | UNITED KINGDOM |
| | SPONSOR UKAEA |
| | ORGANIZATION FOULNESS |
| Title 2 | Project Leader |
| | T P O'BRIEN |
| Initiated 1976 Completed : 1978 | Scientists: |
| Status : Last updating | |

Description:

1. General Aim

Experimental work is required to supplement the information available at present to determine the effect of missile impact on containment structures. The missiles of interest can impact externally, as for example, a crashing aircraft or aircraft debris, or internally, such as those resulting from a steam strum or pressure vessel rupture, or the projection of a large pump casing following an adjacent pipe rupture.

2. Particular Objectives

2.1 Aircraft Impact

US and German design codes require the containment structure to withstand the effects of a specified aircraft impact. A similar requirement has now been proposed by the CEGB. A review of existing data has indicated that a research programme is necessary to determine how a number of structural parameters modified:

i the static strength characteristics under concentrated loading

ii the variation of load with time during loading

iii the variation of deflections with time during the following loading

iv the development and propagation of cracking and yielding

2.2 Impact of Other Missiles

In the consideration of aircraft impact, penetration of the structure is not expected to be important as the impact pressures are probably less than the compressive strength of the concrete. However, for the type of missiles resulting from steam drum or pressure vessel failure, local penetration must be considered as well as the gross effect of the applied load.

contd.....

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Classification

| Title 1 | COUNTRY |
|---|---|
| | SPONSOR |
| | ORGANIZATION |
| Title 2 | Project Leader |
| Initiated    Completed : | Scientists: |
| Status :    Last updating | ...... |

Description:

3. Experimental Facility and Programme

The test facility will consist of a missile launcher (which is already available in the UK), to project model missiles at various speeds against various concrete targets.

The basic target size at 1/25th scale will be 1.8 m dia. and 60 mm thick corresponding to 45 m dia. and 1.5 m thick full size. Targets will be constructed from unreinforced, reinforced and pre-stressed concrete. Variations in tensile and shear reinforcement, target thickness and diameter will be made.

A 'standard' missile of 1.6 kg at 215 m/s would be used as well as a crushable missile to simulate on the model scale the calculated load-time variation for an MRCA aircraft impacting at 215 m/s. Larger irregular missiles would be used for the penetration studies.

4. Project Status

Details of the number of tests, the rate of carrying out the test and the parameters to be varied have not been finally confirmed although proposals have been made. It is possible that priority would be given to internal missiles.

5. Relations with Other Projects

Studies concerned with assessing the velocity of a missile produced by failure of a component containing pressurised water or a pressurised steam/water mixture have been made. Experiments are in progress to validate these assessments.

Classification

7.1

| Title 1<br>INTERNAL MISSILE EFFECTS | COUNTRY<br>UNITED KINGDOM |
|---|---|
| | SPONSOR UKAEA |
| | ORGANIZATION<br>SRD CULCHETH |
| Title 2 | Project Leader<br>D L HUNT |
| Initiated 1974 Completed : | Scientists: |
| Status : Last updating | |

Description:

1. General Aim

To enable the possible missile effect to be predicted following an assumed circuit or vessel rupture, and thus to design containment.

2. Particular Objectives

For the assumed case of a ruptured pressure vessel, to predict the velocity which may be attained by a large fragment.

3. Experimental Facilities and Programme

A tube (circa 20 cm diameter) is pressurised to LWR conditions, then the end is removed by bursting a diaphragm. The acceleration of the tube is observed.

4. Project Status

The "steam rocket" experiment has been built to test SRD calculational methods. The flight of the pipe is measured over 15 ft mainly by fast photography (2000 frames/sec) against a grid, but acceleration over the first 6" is by a photo-electric method. Pressure variation in flight will be measured using transducers with flying leads.

The experimental programme of four tests has been completed using three tubes of masses 84 kg, 51 kg and 33 kg. Maximum velocities between 38.5 and 5 .2 m/s were measured.

A report comparing the experimental work with theoretical predictions is in preparation.

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. 77 - 31. 12. 77 | 7.2 | RS 78 D |

| Vorhaben/Project Title | |
|---|---|
| Kondensation V, Teil 1<br><br>Condensation V, Part 1 | Land/Country: BRD<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik<br>R 521, Karlstein |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1. 4. 76 | 30. 9. 1978 | Dr. Simon |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | 31. 12. 1977 | 4'089.830,-- DM |

## 1. General Aim

The condensation loads were investigated as a function of back pressure, with a well defined reservoir volume, at different containment stiffnesses and 600 mm vent diameter, corresponding to the actual pressure suppression system.

## 2. Particular Objectives

In a single cell (scale 1 : 1) which corresponds to the most important parameters (pipe diameter and length, partial volumes of pressure chamber, air and water room) of the condensation chamber, blowdown tests have been carried out with representative vapour mass flow rates.

## 3. Research Program

3.1 Condensation forces measurement at realistic back pressures

3.2 Influence of the containment stiffness simulated by a wall, which represents the condensation chamber stiffness

3.3 Visual observations of the phenomena

3.4 Investigations with changed end geometry on a vent of 600 mm diameter.

4. Test Facilities

The tests were carried out in the present test stand of the GKM. The test parameters were:

| | |
|---|---|
| Back pressure: | 1; 2; 2,8 bar |
| Vapour flow density: | Transient (simulated from a 100 $m^3$ reservoir) |
| Temperature range: | 30 - 65 °C |
| Containment stiffness: | 9 and 14 Hz. |

5. Progress to Date

In the evaluation of tests completed last year, frequency distributions of the forces occurring in the vent pipe supports were determined.

After reactivating the test equipment, the first series of four tests was conducted. In this series a) back pressure was reduced at low water temperature and b) initial water temperature increased to 75 °C at the regular backpressure of 2.7 bar in the condensation chamber.

Subsequently condensation tests were performed with transient mass flow and varying vent pipe end geometries. The first nozzle to be tested - a vent pipe with an open lower end and lateral openings (20 mm Ø) - showed in some instances the same behaviour as the open ended vent, and in others, stronger excitations were observed than in corresponding test sections of the open ended vent, due to pulsating condensation. This phenomenon was especially observed during the last test phase.

The second geometry investigated was a pipe with a plugged hemispherically-shaped end and discharge holes of 47 mm Ø.

After completion of the vent pipe end geometries two tests with increased steam mass flow (approx. 60 kg/$m^2$s) at the beginning of the test were conducted.

In agreement with the technical inspectorate the flexible wall was removed from the water side of the test container and tests performed whereby a) the 50 $m^3$ compression chamber was coupled to the condensation vent pipe, b) the compression chamber was by-passed by a DN 250 pipe. In addition, in two tests the condensation chamber was ventilated using the compression chamber so that no backpressure was built up.

A low initial water temperature (30 °C average) was used for the above tests, the mass flow density transient corresponds to average flow.

6. Results

Evaluation of the support forces exerted at 2 m submergence showed only slightly changed average values when using a wall with increased flexibility.

Increased water temperatures led to variations of the condensation process at the end of the vent pipe forming large steam bubbles. A significant influence of the pressure level at the container walls was not observed.

Tests with vent pipe end geometries showed in part - depending on steam mass flow and water temperature - varying behaviour compared with the open ended vent; considering the max. and average values, of all test events, however, no significant differences were observed.

Tests with increased initial mass flow rendered increased pressure amplitudes in the initial phase of aerated condensation, their max. values, however, were within the range (test 1 - 34) obtained from the first test series.

When the compression chamber was by-passed build-up of backpressure in the condensation chamber was eliminated.

"Chugging" processes were observed during the entire test period. Thereby sometimes the water in the vent was surged over the outside water level during the low-pressure phase in the tube. The max. and average pressure loads observed thereby at the container walls were higher than with preceeding tests.

As was expected with these tests, the highest dynamic pressure amplitudes occurred with a by-passed compression chamber after demounting the flexible wall. Due to absence of air volume, no backpressure occurred in the condensation chamber. With coupled compression chamber a slight increase of the average values was observed as compared to previous tests, no increases were detected with max. pressures.

7. Next Steps

Evaluation of measurement data

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. 77 - 31. 12. 77 | 7.2 | RS 78 E |

| Vorhaben/Project Title | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor |
|---|---|
| Kondensation V, Teil 2<br><br>Condensation V, Part 2 | KRAFTWERK UNION AG<br>Reaktortechnik<br><br>R 521, Karlstein |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1. 4. 76 | 31. 12. 1977 | Dr. Simon |
| Stand der Arbeiten/Status | Berichtsdatum/Last Updating | Bewilligte Mittel/Funds |
| Completed | 31. 12. 1977 | 417.600,-- DM |

1. General Aim

Verification of the single cell theory and determination of the condensation load in different water suppression pools.

2. Particular Objectives

Previous model tank experiments (Condensation IV) were continued; the evaluation of the results was carried out from modified (statistical) viewpoints.

3. Research Program

3.1 Construction of the test stand

3.2 Tests with two troughs (single cells)

3.3 Main test with other trough arrangements up to 5 vented pairs

3.4 Evaluation of the results.

4. Test Facilities

Two model troughs with different sizes and defined wall stiffnesses were built, including two vents and reservoir. The existing test stand was modified for the new troughs and geometries. The instrumentation was completed.

5. Progress to Date

Complementory statistical evaluations by means of a computer have been performed for the first 2-trough test series, whereby different methods used for event detection and their influence on the result were investigated.

Two differently sized concrete cells were constructed and fabricated in order to perform further tests.

The test sequence prepared for the planned condensation tests with multi-cell arrangements was assembled in a concrete trough with two separate condensation pools. From a common steam source (analogous to the drywell of reactor system), two and six condensation vent pipes respectively can lead into the two pools.

A steam supply connection was installed leading to the high-pressure vessel of the test plant.

Initial tests were performed with concrete cell 3. Thereby steam was conducted through DN 80 pipes into two separated pools. Steam supply could be supplied either separately or from a common drywell.

Separate tests were performed with 2 and/or 6 steam-carrying pipes in the single cells as well as tests with 8 pipes simultaneously loaded in both pools.

During these tests the drywell volume as specified for the pipes, was varied.

The next step was the assembly of a concrete cell which allowed for an arrangement of max. 10 vent pipes (DN 80). Tests were performed with 1, 2, 6 and 10 pipes. During these tests the specific drywell volume was maintained constantly.

The mass flow density of the steam as refers to the outlet diameter of the tubes was 16 kg/$m^2$s, and was common to all tests. During each test the water temperature increased from 20 to approx. 75 °C. For evaluation of the tests, (a) the max. values as observed on various pressure transducer positions were determined by analysis of the visicorder traces. In parallel, (b) the magnetic tape recordings were evaluated by a computer, especially with respect to evaluation of statistical characteristic data.

## 6. Results

The concrete cell evaluation showed the influence of the number of vent pipes on the height of the pressure amplitudes.

No significant influence on the measurement data could be observed with the 6-vent-tests using a varying drywell volume in ratios of 1/2 and/or 1:4.

The simultaneous steaming of two separate pools from a common source revealed the coupling over the steam path, in that the water level motion in the pipes of both cells often ran synchronously. Usually, however, the violence of the condensation phenomena varied strongly.

## 7. Next Steps

Test results are being evaluated.
Final report is being prepared.

## 8. Relation with Other Projects

RS 78 D

## 9. References

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## 10. Degree of Availability

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| Jan. 1,1977-Dec.31,1977 | 7.2 | PNS 4211 |
| Vorhaben/Project Title<br>Dynamische Beanspruchung von LWR-Druckabbausystemen<br>Dynamic Load of LWR Pressure Suppression Systems | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Kernforschungszentrum Karlsruhe (KfK)<br>Projekt Nukleare Sicherheit (PNS)<br>IRE |
| Arbeitsbeginn/Initiated<br>January 1972 | Arbeitsende/Completed<br>December 1977 | Leiter des Vorhabens/Project Leader<br>R.A. Müller |
| Stand der Arbeiten/Status<br>Completed | Berichtsdatum/Last Updating<br>December 1977 | Bewilligte Mittel/Funds |

## 1. General Aim

The pressure suppression systems used in boiling water reactor facilities are exposed to dynamic loads during specific conditions of operation and in a loss-of-coolant accident, the intensity of which have not been sufficiently taken into account in the original design. The aim consisted in enlarging knowledge of the extended steady-state condensation process so that the dynamic loads applied to the components could be determined. The implementation of this task had to be determined largely by the availability of suitable test facilities over the time required.

## 2. Particular Objectives

Experimental evaluation of the pressure pulsation field in the water pool of pressure suppression systems and of the dynamic response of the vessel walls. Simulation and recalculation of these phenomena by use of appropriate computer codes (essential parameters: mass flow density, water temperature, pressure level, air content of the steam, vent pipe diameter, number of vent pipes, drywell volume).

## 3. Research Program

Work concentrated on the following main topics:

- Gathering of experimental data on the pressure pulsations occurring during the process of condensation in the water pool of a pressure suppression system and on the resulting dynamic responses of the vessel structure.
- Evaluation and interpretation of experimental data, using available computer codes.

- Simulation and recalculation of these events using existing computer codes as well as computer codes to be developed.

4. Experimental Facilities, Computer Codes

Measurements have been performed in the following experimental facilities (in most cases as a supplement and in coordination with the measurements performed by other participating institutions and partly for the account of third parties):

- Marviken Nuclear Power Station (Sweden),
- Brunsbüttel Nuclear Power Station,
- Philippsburg Nuclear Power Station,
- Karlstein Large Vessel Test Facility,
- Condensation Test Facility at Mannheim Large Power Station.

The following computer codes were used:

- STRUDL/DYNAL, a finite element program for first calculations on the dynamical shell behavior.
- SPHER 1, a semi-analytical program for calculation of the dynamical deformation of a thin-walled spherical shell.
- KONDAS for calculation of the low-frequency events of pulsating condensation.
- WELLA for calculation of waive propagation in an anular wetwell with the simultaneous recording of also asynchronous pressure wave generation at the individual vent pipes.
- AKUDYN for calculation of the effects on the wall structure produced by the very short-term "pressure needles" observed in the experiments.

5. Work Performed

The final documentation was written on the infrared measuring equipment used in the Marviken II tests. Moreover, a quantitative comparison was made with the measured values determined by other measuring methods as regards the integral mass flow into the wetwell.

The controlling measurements at the Mannheim Large Power Station have been continued on behalf and for the account of TÜV Baden (Technical Inspectorate). The 24 tests as yet performed within a new test series referred to the design and geometry of the pressure suppression system of the Philippsburg Nuclear Power Station. All the measured data were recorded on magnetic tape, plotted at IRE, and transmitted to TÜV Baden.

The influence exerted by the water in the wetwell on the dynamic shell behavior was estimated analytically in a first step using a simple model. To calculate the dynamic deformation of a thin-walled spherical shell, a computer program (SPHER 1) was developed which is based on the shell theory by Flügge. It was used in test computations relating to local impact load. The load values employed were the transient wall pressures measured in the blowdown tests of the Brunsbüttel Nuclear Power Station.

The parameter calculations performed on behalf and for the account of TÜV Baden, using the computer programs specifically developed for this purpose, were terminated. The KONDAS computer code was also used to investigate pulsating condensation in the multi-pipe assembly. The results obtained together with directly derived test data were used to study the overlapping statistics in the reactor facility using the WELLA computer code. Finally, AKUDYN was employed to study the impacts of the very short-term "pressure needles" observed in the experiments with the help of a substitute model for coupling between the fluid and the wall structure. The KONDAS computer code has meanwhile been taken over by Gesellschaft für Reaktorsicherheit (GRS) Munich.

Results

The infrared measuring technique, as used so far, is suited above all for the measurement of the droplet velocity and for the determination of the steam density, practically without any delay. All the pressure variations having a high time resolution are recorded via the steam density.

The eigenfrequencies of the wetwell calculated with the water in the wetwell taken into account agree well with the frequencies observed during the blowdown tests performed in the Brunsbüttel Nuclear Power

Station. The calculations on the dynamical deformation of a thin-walled spherical shell exposed locally to impact load have shown that the local deflection propagates to the more distant spherical zones after a great many of cycles only. The local behavior calculated in terms of time agrees quite well with the structural vibrations measured in Brunsbüttel Nuclear Power Station tests.

The results of computations made with the KONDAS code and relating to the low-frequency events of pulsating condensation satisfactorily agree with the experimental data. Also the calculations performed for a multipipe assembly did not yield unfavorable coupling effects. The results of these computations have been directly integrated in the licensing procedure for German BWR nuclear power stations.

## 7. Plans for Future Work

Another 8 tests are planned to be performed in the Mannheim Large Power Station. They will be based on the conditions prevailing in the new KWU boiling reactor line, i.e., in the Gundremmingen II Nuclear Power Station. The independent accompanying controlling measurements will be carried out for these tests also to the extent previously applied.

Work on the dynamic shell behavior will be carried on in future under PNS 4223, since it involves a coupled fluid shell dynamics problem which has to be treated in its entirety.

## 8. Relation with Other Projects

An exchange of experience has been established with other groups in Germany working on the same problems as well as with TÜV Baden.

## 9. Literature

1st Seniannular Progress Report PNS 1/1977

2nd Semiannular Progress Report PNS 2/1977

Barschdorff,D., Neumann,M., Rabold,W., Weißhaupt,R., Erb,E., Philipp,P., Wolf,E.:
Entwicklung einer Meßeinrichtung zur Bestimmung der Einzelkomponenten in einer instationären Luft/$H_2O$-Zweiphasenströmung nach dem Infrarot-Absorptionsprinzip.
KFK 2550

Barschdorff,D., Erb,E., Neumann,M., Philipp,P., Wolf,E.:
Einsatz einer Infrarot-Meßeinrichtung bei den Marviken II-Blowdown Experimenten
KFK 2578

Krieg,R., Zehlein,H.:
"Coupled Problems in Transient Fluid and Structural Dynamics with Application to Nuclear Engineering"
International Symposium on Innovative Numerical Analysis in Applied Engineering Science, CETIM, Paris, May 1977

Göller,B., Hailfinger,G., Krieg,R.:
Vibrations of the Pressure Suppression System of a Boiling Water Reactor.
Int. Conf. Vibration in Nuclear Plant, 9 - 12 May, 78, Keswick UK

Krieg,R., Zehlein,H.:
Coupled Problems in Transient Fluid and Structural Dynamics with Application to Nuclear Engineering
Int. Symp. on Innovative Num. Analysis in Appl. Eng. Science, Versailles, May 77

Class,G.:
Übersicht über das Verhalten von SWR-Druckabbausystemen beim Kühlmittelverluststörfall
KFK-Nachrichten 1/1977, 9. Jahrgang

Class,G.:
Theoretische Untersuchungen der Druckpulsentstehung bei der Dampfkondensation im Druckabbausystem von Siedewasserreaktoren. Rechenprogramm KONDAS
KFK 2487 (Oktober 1977)

Class,G.:
Das Verhalten von Siedewasserreaktor-Druckabbausystemen beim Kühlmittelverluststörfall.
Seminar über neuere Arbeiten auf dem Gebiet der Reaktortechnik, RWTH Aachen, May 12, 1977

Müller,R.A., Class,G., Fränkel,E., Simon, U.:
Experimentelle und theoretische Untersuchungen zur Belastung des SWR-Druckabbausystems.
Reaktortagung, Mannheim, March 29 - April 1, 1977; Deutsches Atomforum e.V., Kerntechnische Gesellschaft im Deutschen Atomforum e.V., Leopoldshafen 1977 ZAED p. 237-40

10. Degree of Availability
Unrestricted distribution

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| | |
|---|---|
| Classification: 7.1, 7.2 | |
| Title: | Country: DENMARK |
| | Sponsor: Risø National Laboratory |
| Title: CONTAC-II. A containment transient analysis code | Organization: Risø National Laboratory |
| Initiated date: January 1969 Completed date: Feb. 70<br>Status: In use | Scientists:<br>Aksel Olsen<br>N. Kjær-Pedersen<br>V.S. Pejtersen |

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| | |
|---|---|
| | Classification: 7.1 7.2 |
| Title: | Country: DENMARK |
| | Sponsor: Risø National Laboratory |
| Title: CONTAC-III. A containment transient analysis code. | Organization: Risø National Laboratory |
| Initiated date: Completed date: March 1975<br>Status: Tested. | Scientists:<br>V.S. Pejtersen<br>F. Cortzen<br>K.L. Thomsen |

| Classification: | $\boxed{7.1}$ 7.2 |
|---|---|

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|---|---|
| Title: | Country: DENMARK |
| | Sponsor: Risø National Laboratory |
| Title: MACON. A Containment Multiroom Transient Analysis Code. | Organization: Risø National Laboratory |
| Initiated date: 1975 Completed date: <br> Status: Being tested. | Scientists: V.S. Pejtersen M.B. Andersen |

| Title 1 (Original language) | Classification |
|---|---|
| Heat transfer in pressure suppression | 7.2 |
| Title 2 (English) | Country ITALY<br>Sponsor / Organisation: CNEN |
| Date initiated 1975<br>Date completed 1977<br>Last updating April 1977 | Project Leader<br>G. E. Farello |

1 - General aim

Heat transfer coefficient (vapour to liquid) measurements.

2 - Particular objectives

Experimental determination of heat transfer coefficient (vapour to water) related to steam relief in subcooled pool water.

3 - Experimental facility

Visualized test section in an optical bench (small experimental loop); 0,2 m3/h, 10 kW.

4 - Project status

A final report collects the experimental data and provides a heat transfer correlation for direct condensation of steam.

| Title 1 (Original language)<br>SOPRE 1 - Ricerca sul comportamento del sistema di contenimento a soppressione di pressione in caso di LOCA. | Classification<br>7.2 |
|---|---|
| Title 2 (English)<br>SOPRE 1 - Research on the behaviour of pressure suppression containment system after LOCA. | Country ITALY<br>Sponsor CNR - CNEN<br>Organisation University of Pisa |
| Date initiated 1974<br>Date completed 1977 (first phase)<br>Last updating January 1977 | Project Leader<br>M. MAZZINI |

1) General aim
To acquire specific knowledge on basic phenomena in pressure suppression containment system after LOCA.
Assess the effectiveness of computer codes, like CONTEMPT series, for the prevision of pressure and temperature transients in a pressure suppression containment system during LOCA.

2) Particular objectives
The program is intended to investigate the pressure and temperature transients within a model of MARK II containment system, by varying blow-down flow rate and energy, number and submergence of vent pipes and pool temperature.

3) Experimental facilities
SOPRE 1 apparatus, which is a 1 : 13 scale model of a MARK II system. In the containment model there are located 17 points for measurements of pressure and 14 for measurements of temperature during the transient.

4) Project status
The first series of tests with blow-down nozzle diameter of 15 mm (8 runs) or 50 mm (8 runs) was carried out, varying pressure (from 20 to 85 $Kg_p/cm^2$) and mass (from 45 to 70 Kg) of the water in PIPER vessel.
The experimental results were compared with data from CONTEMPT-PS and CONTEMPT-LT codes.

5) Next steps
Next program includes 10 runs with blow-down nozzle of 50 mm diameter, 55 Kg of water inside the PIPER vessel and a starting pressure of 70 $Kg_p/cm^2$; further 3 runs with a starting pressure in PIPER vessel of 30 $Kg_p/cm^2$ are scheduled for safety reasons. In this second series of tests the pool temperature, the submergence and the number of the vent pipes shall be varied.

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| Title 1 (Original language) | Classification |
|---|---|
| SOPRE 1 - Ricerca sul comportamento del sistema di contenimento a soppressione di pressione in caso di LOCA. | 7.2 |

6) Relation to other projects
Blow-down tests by PIPER Apparatus (the facility and instrumentation are the same) - Project Leader P. VIGNI.
CNEN - Heat Transfer in Vapour Suppression Systems, etc.

7) Reference documents

1. B. GUERRINI, M. MAZZINI
L'apparecchiatura sperimentale SOPRE 1 per ricerche sui sistemi di contenimento a soppressione di pressione.
Ingegneria Nucleare N. 3 - Nuova Serie, Luglio-Dicembre 1976.

2. M. MARINELLI, M. MAZZINI
SOPRE 1: Reasearch on the Pressure Suppression Containment System. First Experiments.
Energia Nucleare N. 11, Vol. 23, Nov. 1976.

3. N. CERULLO et alii
Experimental Investigation of the Behaviour of Pressure Suppression Containment Systems by the SOPRE 1 Facility.
Paper presented at the 4th S.M.I.R.T. Conference, S. Francisco (USA), August 1977.

4. M. MAZZINI et alii
SOPRE 1: Ricerca sul sistema di contenimento a soppressione di pressione. Relazione sulla prima serie di prove.
Istituto di Impianti Nucleari dell'Università di Pisa, RL 254(76).

8) Degree of availability
The first references are free; the last one may be available with the authorization of the CNEN.

| PROJECT TITLE : | CLASSIFICATION |
| --- | --- |
| SOPRE 1 - Research on the behaviour of pressure suppression containment system after LOCA | 7.2 |
| SPONSORING COUNTRY : ITALY | ORGANISATION : UNIVERSITY OF PISA |
| DATE INITIATED : 1974 (actual phase)<br>DATE COMPLETED : 1976 (actual phase) | PROJECT LEADER : M. MAZZINI |

Description :

Research program:

Experimental investigation of pressure and temperature transients within a model of MARK II pressure suppression containment system. The effects of blow-down flow rate, inlet area and flow rate distribution among the vent pipes, are examined.

Assess of capability of computer codes, like CONTEMPT-PS, to fit experimental data.

In the next future it is possible that MARK III type systems will be studied.

Facilities:

SOPRE 1 apparatus, which is a 1:13 scale model of MARK II system. Steam is supplied by a pressure vessel; its main features are: design pressure 100 $Kg/cm^2$, design temperature 310 °C, capacity 100 litres, blow-down area variable up to 6 cm of equivalent diameter.

Reference documents:

1. B. GUERRINI, M. MAZZINI

: L'apparecchiatura sperimentale SOPRE 1 per ricerche sui sistemi di contenimento a soppressione di pressione.

To be issued.

Related projects:
1.1.1 - 7.2(CNEN)

Remark:

The research is performed on C.N.R. contract. Financial support by C.N.E.N. and ENEL is hoped for the future.

| Title 1 (Original language)<br>Analisi dei transitori termo-fluido-dinamici a seguito di LOCA nei sistemi di contenimento dei reattori ad acqua leggera. | Classification<br><br>7.2 |
|---|---|
| Title 2 (English)<br><br>Analysis of thermofluidynamic transients in LWR containment systems following a LOCA. | Country ITALY<br>Sponsor CNEN and CNR<br>Organisation University of Pisa |
| Date initiated 1975<br>Date completed 1977<br>Last updating May 1977 | Project Leader<br><br>N. CERULLO |

## DESCRIPTION

### Research Program

The program has the aim of investigating, using analytical tools, thermofluidynamic transients in Light Water Reactors Containment Systems following LOCAs. The main purpose of the work is to verify the capability of the computer codes used in this study, checking their calculations with experimental data obtained at the safety facilities of Scalbatraio Research Center, University of Pisa.

### Facilities

IBM, 370/168 and 370/158 Computer belonging to CNUCE of Pisa.
SOPRE 1: an experimental small scale facility representing a Mark II pressure suppression system which is at the Scalbatraio Laboratory of Nuclear Plant Institute, University of Pisa.

### Reference documents


1. R.J. WAGNER, L.L. WHEAT
   CONTEMPT-LT USERS MANUAL - Interim Report I-214-74-12.1
   Aerojet Nuclear Company - USA - August 1973.
2. M. MARINELLI, M. MAZZINI
   SOPRE 1: Ricerca sul sistema di contenimento a soppressione di pressione.
   Istituto di Impianti Nucleari, Università di Pisa, RP 210(75); 1975.
3. M. MARINELLI, M. MAZZINI, A. MAZZONI, P. TODISCO
   Evoluzione e stato attuale delle conoscenze sui fenomeni termo-fluido-dinamici nei sistemi di contenimento a soppressione di pressione degli Impianti Nucleari tipo BWR.
   Istituto di Impianti Nucleari, Università di Pisa, RL 242(76).
4. M. MAZZINI, A. MAZZONI, P. TODISCO
   SOPRE 1: Ricerca sul sistema di contenimento a soppressione di pressione: Relazione sulla prima serie di prove.
   Istituto di Impianti Nucleari, Università di Pisa, RL 254(76).

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| Title 1 (Original language) | Classification |
|---|---|
| Analisi dei transitori termo-fluido-dinamici a seguito di LOCA nei sistemi di contenimento dei reattori ad acqua leggera. | 7.2 |

5. N. CERULLO, A. DELLI GATTI, M. MARINELLI, M. MAZZINI, A. MAZZONI, A. SBRANA, P. TODISCO
   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.

| Title 1 (Original language)<br>Instabilità connesse con il rilascio del vapore attraverso le valvole di sicurezza | Classification<br>7.2 |
|---|---|
| Title 2 (English)<br>Instability phenomena related to steam relief through S.R.V. | Country ITALY<br>Sponsor C.N.E.N.<br>Organisation C.N.E.N. |
| Date initiated 3 - 1976<br>Date completed 6 - 1978<br>Last updating April 1977 | Project Leader<br>D. Pitimada |

1. General aim

Experimental study of air, water and steam discharge through a single safety relief valve.

2. Particular objectives

Determination of instabilities connected to air-water clearing, bubble dynamics and to steam flow pulsations.
Implementation of a computer code for the determination of chief parameters interesting the discharge.

3. Experimental facilities and programme

Facility consisting of: $2 m^3$ boiler ($70 kg/cm^2$), 2" relief valve, 70 m long, 1.5" SS. discharge pipe, $7 m^3$ suppression pool.

4. Project status

The facility is ready, preliminary results concerning steam flow pulsations are available.
A preliminary computer code, concerning transient phenomena of discharge, is implemented.

5. Next steps

Experimental determination of pressure, temperature and flow rate as functions of steam and water conditions.
Implementation of computer codes.
Comparison of experimental data with computer codes.

| TITLE 1 (original language) | Classification |
|---|---|
| Instabilità connesse con il rilascio del vapore attraverso le valvole di sicurezza | $\boxed{1.1.2}$ - 7.2 |
| TITLE 2 (english) | Country: ITALY |
| Instability phenomena related to steam relief through S.R.V. | Sponsor: CNEN<br>Organisation: CNEN |
| Date initiated 3-1976<br>Date completed 6-1978<br>Last updating June 1976 | Project Leader<br><br>D. Pitimada |

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

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. 77 - 31. 12. 77 | 7.3 | RS 223 |

| Vorhaben/Project Title | |
|---|---|
| Untersuchung und Entwicklung von Systemen zur Begrenzung der Wasserstoffkonzentration im Sicherheitsbehälter von Siedewasserreaktoren<br><br>Investigation and Development of Systems Limitating the $H_2$-Concentration in the BWR Containment | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik<br>R 312, Frankfurt |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1. 7. 1976 | 31. 5. 1978 | E. Queiser |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | 31. 12. 1977 | 310.600,-- DM |

1. General Aim

The purpose of the R+D-program is to improve our knowledge of hydrogen generation and distribution in the BWR containment during reactor operation and after LOCA, and especially to develop and test concepts and methods for measurement and limitation of $H_2$ concentrations in the containment atmosphere.

2. Particular Objectives

The specific aim will be verification of the applicability of apparatus, devices and catalysts for $H_2$-removal after LOCA. Catalytic recombiners in which hydrogen will oxidize with atmospheric oxygen to form water, will have to be tested under reference operating and LOCA conditions.

Tests will have to verify that decontamination procedures (e.g. for fission-product Iodine), which are to be expected in the post-LOCA atmosphere of the containment, will not impair the mode of operation of the catalysts.

3. Research Program

3.1 Theoretical Investigations

$H_2$-production

Survey of literature and possibly measurement data to improve knowledge of:

- metal-water reaction
- production rate by radiolysis ($G(H_2)$-value)

- gamma-absorption within the reactor core
- dependency on coolant quality

$H_2$-measurement

In connection with paragraph 3.2, suitable sampling and measurement methods will be selected, with special attention to post-LOCA conditions.

3.2 Experimental Investigations

Tests are planned in a suitable semi-technical lab-facility which is similar to the KWU off-gas system recombiner. The major components of this facility are: preheater, recombiner, cooler and the required measuring devices. The following investigations will be carried out:

Thermal capacity of the total system:
Evaluation of catalyst and recombiner material properties, both under normal and extreme temperature conditions of 50 to 600 °C consistent with possible failure modes.

Chemical performance:
Verification of functioning and regeneration-ability of the catalysts by specified additions of contamination products (e.g. Iodine) using different recombiner temperatures, $H_2$-concentrations and atmospheric conditions.

Optimization of catalyst quantities:
Evaluation of thermal capability and chemical properties under different ratios of catalyst-to-air-$H_2$-volume.

3.3 Development of a $H_2$-Reducing System

Based on theoretical and experimental test results, concepts and systems for measurement and control of $H_2$-concentration present in the containment atmosphere will be developed.

## 4. Experimental Facilities

The test facility represents in principal the recombination unit of the off-gas system at a reduced scale (1 : 30). The model laws indicate that the same hydrodynamic and geometric conditions are present in the model.

The arrangement consists of a proportioning apparatus for varying the hydrogen enrichment of the atmosphere. The air-$H_2$ mixture will be saturated with steam at 80 °C in a moistening chamber and then heated in a heater. Then a heated temperature-controlled reaction container will be introduced, in which by means of a catalyst the recombination will be initiated. The hydrogen-free gas will then be released into the atmosphere.

Addition of various contaminations is made prior to entering the reaction container by means of a heated tube and/or injection device.

Residual quantities of the catalyst poisoning Iodine will be determined by volumetric analysis. Hydrogen analysis will be performed by a continuously operating thermal conductivity measuring device.

## 5. Progress to Date

Pretest series were run in order to verify the effects caused by oil mists on the activity of the catalyst.

Longtime tests (33 days) were run with

- Iodine
- Fission gas follow-products
- concrete dust
- oil moisture
- boric acid

The catalyst poisons were added corresponding to the expected freeset-mechanisms after LOCA.

The semi-technical lab-facility was installed. The heating of the recombiner was tested; the characteristic of the temperature increase was measured.

6. Results

The pretests showed that the efficiency of the Hydrogen-recombiner was influenced by oil moisture at 200 °C process temperature. A regeneration was reached by increasing the process temperature to 250 °C.

The longtime tests confirmed that the Hydrogen-recombiner efficiency depends strongly on the temperature of the catalyst. Therefore a homogeneous and constant temperature field in the catalyst will be necessary.

The process temperature of 200 °C was reached after about 1 day without flow. When condensators and preheaters were used the temperature of 200 °C was reached after 2 hours in the enter region of the gas. The equilibrium temperature reached 300 °C after 8 hours.

7. Next Steps

Heat-up characteristics after failure of preheater or external chamber heater .

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 |
|---|---|---|
| 1.1.77 - 31.12.77 | 7.3 | RS 246 |

| Vorhaben/Project Title | Land/Country |
|---|---|
| Experimentelle Untersuchung der Wasserstoffverteilung im Containment eines Leichtwasserreaktors nach einem Kühlmittelverlust-Störfall<br><br>Experimental investigation of the hydrogen distribution in the containment of a light-water reactor following a loss-of-coolant accident | FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: BATTELLE-INSTITUT E.V. Frankfurt am Main<br><br>Abt. Energietechnik |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.8.1977 | 30.9.1978 | G. Langer |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | 31.12.1977 | 648.900,-- DM |

1. General Aim

Various processes taking place during and after a loss-of-coolant accident in a light-water reactor (e. g. zirconium-steam reaction, radiolysis) may lead to the generation of hydrogen, which then spreads in the containment by diffusion and convection. These transport mechanisms influence the local hydrogen concentrations. The objective of the planned investigations is to measure the hydrogen distribution in a multi-compartment containment. The model containment built at Battelle under project RS 50 is to be used for this purpose. The experimental results will be used to determine whether existing computer models (e. g. RALOC) describe the transport processes within a real containment geometry with sufficient accuracy.

2. Particular Objectives

-

3. Research Program

The scheduled research program includes seven basic experiments to be performed at room temperature. The hydrogen, also at room temperature, is fed from a single large-area source into a central containment compartment. The local hydrogen concentration is measured by numerous detectors. The experiments are scheduled to last 5 hours. The experimental configurations include one- and two-compartment models that extend vertically or horizontally. The following experimental parameters are varied:

- location of the hydrogen source
- hydrogen feed rate
- geometry of the containment compartments and the connecting overflow openings.

The purpose of the basic experiments is to determine the variation of the hydrogen concentration within one or two compartments as a result of diffusion and convection.

In experiments planned for a later date (not included in the above work), the influence of elevated temperatures and thermal stratifications will be investigated, and measurements will be made during and after a blowdown.

4. Experimental Facilities, Computer Codes

The model containment built for the RS 50 experiments is the most important component. The containment compartments used in the present investigations are equipped with detectors wich measure the local hydrogen concentration. The measuring principle is based on the change in resistance of a bridge resistor following a temperature rise. The temperature rise results from the catalytic combustion of the hydrogen. The measured signals are processed and stored with the existing data acquisition system.

The hydrogen supply system is operated with a bottle battery.

Pre- and post-experiment calculations are performed at the Gesellschaft für Reaktorsicherheit in Munich with the computer code RALOC.

5. Progress to Date

The following work has been completed in the report period:

- Planning of the instrumentation system, collection of bids, purchase of the detectors
- Planning of the hydrogen supply system, procurement activities.

The scheduled work was delayed because of unexpected difficulties in the RS 50 experiments, for which the same experimental facilities are used.

6. Results

Major results are not yet available since experiments have not yet been conducted.

7. Next Steps

The procurement and set-up of the instrumentation should be completed in the next report period. In addition, the hydrogen supply system will be installed.

The preliminary experiments will be planned in detail, e. g. modification of the data acquisition system.

8. Relation with Other Projects

Research activities at the Gesellschaft für Reaktorsicherheit in Munich towards further development of the computer code RALOC. Program RS 50 "Investigation of the phenomena occurring within a multi-compartment containment following the rupture of a coolant pipe in a water-cooled reactor".

9. References

-

10. Degree of Availability of the Reports

-

CLASSIFICATION

7.3

| | |
|---|---|
| TITLE 1 CORROSION EN AMBIANCES NATURELLES ET ARTIFICIELLES | COUNTRY FRANCE |
| | SPONSOR E.D.F. |
| | ORGANIZATION E.D.F. |
| TITLE 2 CORROSION OF MATERIALS INSIDE OF THE CONTAINMENT | Project Leader E.D.F./DER/TEGAEE |
| Stated février 1975 Completed 1976 | Scientists MM. BUREAU BERGE |
| Status Last updating : 20.01.75 | |

I - GENERAL AIM

II - PARTICULAR OBJECTIVES

Détermination de la quantité d'hydrogène produite lors de l'aspersion dans l'enceinte de confinement d'une chaudière à eau sous-pression d'une solution d'acide borique et de soude après une rupture de tuyauterie primaire.

III - EXPERIMENTAL FACILITIES AND PROGRAMME

Etude de la corrosion de l'alliage léger AG 5 et des zincs de galvanisation Z6 et Z9. Influence de la température du pH, du mode de contact métal-solution et de la composition des alliages légers.

IV - PROJECT STATUS

Début des essais en 1975.

V - NEXT STEPS

- essais en immersion,

- essais en aspersion.

...

VI - RELATION WITH OTHER PROJECTS

Néant.

VII - REFERENCE DOCUMENTS

Aspersion de l'enceinte de confinement d'une chaudière à eau sous-pression. Aspects chimiques. Note E.D.F./SEPTEN (restreinte E.D.F.)

IX - DEGREE OF AVAILABILITY

E.D.F.

| Title 1 (Original language) | Classification |
|---|---|
| Controllo della concentrazione di idrogeno nel contenitore dopo LOCA | 7.3 |
| **Title 2 (English)** | **Country** ITALY |
| Control of Hydrogen Concentration in Containment Following a LOCA | Sponsor CNEN and CNR<br>Organisation University of Pisa - CAMEN |
| Date initiated 1975<br>Date completed 1978<br>Last updating April 1977 | **Project Leader**<br>S. LANZA (University)<br>S. MANFREDINI (CAMEN) |

1. General Aim

To study hydrogen transfer inside a containment with the goal of evaluating the capability of proposed devices to keep hydrogen concentration in reactor containments below the flammability limit in post--LOCA conditions

2. Particular Objectives

a) Molecular diffusion coefficient evaluation for air-water vapour mixtures in post-LOCA conditions
b) Solution of the diffusion equation for PSICO-10 geometry
c) Influence of the turbulence due to thermal gradients on hydrogen mixing

3. Experimental Facilities and Program

3.1 - Facilities:
- PSICO-10 facility
- hydrogen detection assembly
- hydrogen injection system

3.2 - Program:
a) a set of runs to study hydrogen diffusion at room temperature by injecting hydrogen-nitrogen mixtures .
b) a set of runs to study how thermal gradients affect hydrogen diffusion

4. Project Status

Item 2a was done.
Item 2b was executed for one-dimensional problems.
A first arrangement of the experimental equipment was carried and a part of the item 3a was done.

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| Title 1 (Original language) | Classification |
|---|---|
| Controllo della concentrazione di idrogeno nel contenitore dopo LOCA | 7.3 |

5. Next Steps

Improvement of the detection performance, i.e. type and number of detecting devices.
To end the R.T. experiments and carry out runs with thermal gradients.

6. Reference Documents


1. FINESCHI F., LANZA S.
   Una proposta per la valutazione del coefficiente di diffusione molecolare dell'idrogeno in aria satura di vapor d'acqua
   Istituto Impianti Nucleari - Pisa RP 286(77)
2. ISTITUTO IMPIANTI NUCLEARI
   Ricerche sui problemi relativi al rilascio di effluenti gassosi negli impianti nucleari - Parte IV
   Pisa, RL 252(76)
3. ISTITUTO DI IMPIANTI NUCLEARI
   Ricerche sui problemi relativi al rilascio di effluenti gassosi negli impianti nucelari - Parte IV
   Pisa, RL 281(77)

| NETHERLANDS ENERGY RESEARCH FOUNDATION (ECN) | | CLASSIFICATION : 7.4 |
|---|---|---|
| TITLE : Advies inzake periodieke kontrole van insluitvaten op lekkages | | COUNTRY: THE NETHERLANDS |
| TITLE (ENGLISH LANGUAGE): Advice on periodic supervision on leakage tightness of containers. | | SPONSOR : Ministry of Social Affairs<br>ORGANIZATION : ECN<br>PROJECTLEADER : H.J. van Grol |
| INITIATED : June 1974 | LAST UPDATING : May 1978 | SCIENTISTS : J.W.H. van den Bergh<br>H. Pruimboom |
| STATUS : ending | COMPLETED : end 1978 | |

General aim

To assemble data on the measurements of the leak rates on containment vessels in order to advise the authorities on periodic supervision of leakage performances as compared to design leak rates.

2. Particular objectives

The program consists of:
- making an inventory of generally used methods of leakage rate measurements including an evaluation resulting into a general purpose method,
- specification of main and secondary variables when measuring leakage rates,
- performing an analysis on extrapolation modes in order to establish the safe lower limit of test overpressure in relation to design pressure, and
- evaluation of experiences obtained elsewhere with measuring containment leakage rates.

3. Experimental facilities: Not applicable

4. Project status

The activities done so far yielded the next results:
* Preliminary Regulation on the Leak-tightness of Reactor Containments" (Draft in Dutch). Report ECN 0.544.01 - GR 1: January 1977.
* "Leak Rates and Leak-tightness Predictions on Basis of Measurements Performed on Reactor Containments" (Draft in Dutch),
* "Theory and Practice on the Measuring of the Leak Rate of Reactor Containments" (Draft in Dutch), December 9, 1976,
* "Summary of Anwers to the Questionnaire on Reactor Containment Leak-tightness" February 24, 1977
* "Nearly completion of draft version of final report.

5. Next and final steps

Completion of a final report containing the documents mentioned under 4 and incorporation of comments of the sponsors on these documents and the draft version of the final report.

6. Relation with other projects: -

7. Reference documents: See under 4 and 5

8. Degree of availability

The reports will be translated in the Enlish language. They will be submitted to the Ministry of Social Affairs, Postbus 69, Voorburg.

9. Budget: -

10. Personnel: 0.6 manyear

# 8. INSTRUMENTATION, CONTROL AND COMPUTERISED PROTECTION

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. 77 - 31. 12. 77 | 8 | RS 240 |

| Vorhaben/Project Title | |
|---|---|
| Untersuchungen zur Funktionstüchtigkeit der Druckhalter-Sicherheitsventile und des Abblasetanks bei Abblasen von heißem Druckwasser<br><br>Investigations on the Functioning of the Pressurizer Safety Valves and the Relief Tank during Blowdown of Hot Pressurized Water | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik<br><br>R 523, Erlangen<br>R [illegible], Erlangen |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1. 12. 76 | 28. 2. 78 | Landgraf/Helf |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | 31. 12. 77 | 1'176.600,-- DM |

1. <u>General Aim</u>

Verification of ability of PWR pressurizer safety valves and relief tank to function during blow down with hot pressurized water in order to be able to control ATWS-malfunctions without additional second shut down system.

2. <u>Particular Objectives</u>

The aim of the experiments is primarily the evaluation of the performance and in particular the control of safety valves, having construction and arrangement features currently used with PWRs. The results are expected to show the structural changes to the main safety valves, the control valves and arrangements necessary in order to retain the response characteristics in the event of a rapid change of phase (steam-hot pressurized water) of the blowdown medium.

In parallel to the above tests, the pressure build-up in the surge line and the relief tank will be determined analytically. In addition, the condition of the relief tank after failure of the rupture discs will be noted.

3. <u>Research Program</u>

3.1 Safety valve tests

3.1.1 Pilot valve tests

3.1.2 Main and pilot valve tests

3.2 Analytical examination of the blowdown system

Compilation of conditions of loading of blowdown system

4. Experimental Facilities

Computer models have already been provided for the analytical work; these, however, have still to be modified for the new boundary conditions.

The major components plus test samples which are necessary for conducting the test are available and will be modified to meet the specific test requirements. The necessary quantities of steam and pressurized water are also available.

5. Progress to Date

Valves, tubes and molds as well as the nitrogen compressor were ordered. The necessary test nozzle on the valve test stand pressurizer was constructed and calculated.

The preliminary test documents required for the NW 80 end nozzle of the valve test stand pressurizer were submitted to TÜV München. After permission had been received, the pressurizer was drilled and nozzles welded.

The required valves were delivered by Messrs. Sempell. Despite repeated grinding of the (valve) seat, the specified leak-tightness could not be achieved with the safety control valve.

Therefore the safety control valve was shipped back to the manufacturer for grinding of the sealing surfaces. In Juni 18, it was brought to Erlangen again. Supporting devices were fabricated for main valve, safety control valve, $H_2O$ channel and nitrogen channel (measurement channel VI). The rerouting of the pulse power-, control- and off-gas lines as well as the heat up of the thermocycle loop was started. Further minor components for the test procedure were ordered.

Then the final test procedure was completed. A draft piping diagram was prepared. A change-over from steam to water was planned and ordered.

The test arrangement was preliminary tested by TÜV Bayern. Pressure test and commissioning were performed in cooperation with the TÜV Nürnberg.

The instrumentation was installed and measurement lines tested.

Tube lines and valves were insulated. Control device for test stop in the event of over-pressure in the pulse power line (220 bar) as well as shutdown control in the event that minimum water level is reached in the $H_2O$ tank were installed in the test facility.

Saturated steam was applied to the safety control valve to adjust it to the desired opening pressure of 170 bar. A check-up of component functions revealed several defects which were corrected.

Subsequently final commissioning of the entire test arrangement was started.

Beside actuation tests being performed to tune the test arrangement to the proper safety control valve actuation response, a demonstration test was also performed. Nominal/actual comparisons were made of the test scope, test costs and time schedule, considering the impositions and requirements which were established in the meantime. The new test program and the corresponding evaluation matrix were completed. Then another check test was made of the expenditures to be expected and also a revision of the time schedule.

For calculation of the pressure build-up and temperature increase present in the blow-down tank during steam blow-down or hot pressurized water, the computer program "ABDRUNA" (verification of blow-down container pressure) was developed.

A comparison performed between measured values (commissioning tests of NPP Unterweser) and the verification of these tests by the "ABDRUNA" computer program, showed relatively good agreement.

The computer program DOMFREI which calculates pressure increase in the blow-down tank dome during free-blowdown, was modified and further developed. This test modification allows for the calculation of pressure build-up in the dome during blow-down of water in any condition.

6. Results

Commissioning phase of the test arrangement was completed. The initial steam test showed that the whole test arrangement is functioning well and appears to be representative and suitable for the planned tests.

The additionally planned investigation of the conditions found in and at the pulse power- and control line will also require - besides other modifications - an extension of the test arrangement.

A comparison of the available main safety valve with that to be used in the future, i.e. starting with NPP Grafenrheinfeld (BAG = Bayernwerk AG), showed that representative results with the currently used valve in Karlstein cannot be planned to be performed in Erlangen, where primary attention will be paid to the pilot valve. Three alternatives have thus been developed in Karlstein, in order to find out about main valves which may yield representative results.

7. Next Steps

The test program will be continued using steam, hot water and transition to hot water.

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. 77 - 31. 12. 77 | 8 | RS 281 |
| Vorhaben/Project Title: Schutzbegrenzungs-System mit Kleinrechnern<br><br>Protection Limitation System with Mini-Computers | | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 1, Erlangen |
| Arbeitsbeginn/Initiated: 1. 7. 77 | Arbeitsende/Completed: 30. 6. 80 | Leiter des Vorhabens/Project Leader: W. Aleite |
| Stand der Arbeiten/Status: Continuing | Berichtsdatum/Last Updating: 31. 12. 77 | Bewilligte Mittel/Funds: 1'368.000,-- DM |

1. **General Aim**

Due to the scope and complexity of task, preferably decentralized mini/micro computer systems will be developed and applied to improve the setting of process variables.

Thereby current protection-, limitating-, monitoring-, and control functions will be extended and refined. It is the aim of the program to gain experiences on the development, construction and operation of such a protection limitation system used in a plant.

2. **Particular Objectives**

A 4-computer system is to be installed to provide protection limitation from DNB and excess power density.

Experiences gained during construction and observation of the system, are to be standardized to such a degree that they may also be used for construction of other systems. Hereby programming of diverse computers has to be considered.

In co-operation with LRA key points will be prepared which will represent a contribution to assuredness, clarity and ease of handling and possibly to standardization of the computer systems. Relationships between construction features and reliability data will be determined.

3. Research Program

3.1 Installation procedure of the 4-computer system
- startup of the soft ware using the ANDI Model
- computer-computer coupling
- computer self-checking, construction of the external monitoring level
- transfer of the program into the CP 550
- interface definition, cabling start-up
- observation

3.2 Development of portable programs (programming language)

3.3 Model for reliability verification

3.4 Optimization of systems by means of the reliability model.

4. Test Facilities

In co-operation with LRA and Halden (AEG)-KWU has developed a reactor protection system for nuclear power plant Brunsbüttel which is planned to be tested as a back-up operation.

LRA has gained extensive experience in automatic computer control, program analysis and reliability verification which will be used for the present program.

Countries such as the FRG, GB and France are developing redundant systems which will process the coolant channel outlet temperatures of LMFBR.

5. Progress to Date and 6. Results

Work was performed on the start-up of the CP 550 analog channel and the incorporation of the executive routines into the KWU-CP 550 assembler. The components of the executive routines were tested and shipped to the assembler supplier for installation into the KWU CP 550 assembler.

Maintenance and back-up work was performed on the coupling program (KOCP) which controls the data exchange between PR 320 and CP 550. It is the aim of the work to achieve a simple and therefore reliable program management.

Transfer of the DNB program into the CP 550 and test work was initiated.

7. <u>Next Steps</u>

Start-up of the measurement data acquisition - and processing program using the PR 320 in co-operation with the analog computer (ANDI-80-model, circuit model) including correction of the program system according to experiences gained.

Installation of the computer-computer coupling between PR 320 and CP 550. This coupling arrangement serves as start-up assistance for a Closed Loop-Computer which has to operate without a recording device.

Transfer of the program system into the CP 550, followed by 1) on-channel, 2) four[illegible]nel test- and verification calculations in the lab.

Definition of interfaces of the DNB-RELEB criteria covering the remaining monitoring-, controlling-, limiting- and protection functions in nuclear power plant Grafenrheinfeld.

8. <u>Relation with Other Projects</u>

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9. <u>References</u>

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

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| 143-1 -13/4112-04 | | 8 |
|---|---|---|
| Titre<br><br>Test en ligne des capteurs. | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA / DgCS |
| Titre (anglais)<br><br>On-line method for testing of instrumentation. | | Organisme exécuteur<br>CEA/SES-SAI<br><br>Responsable<br><br>SAI - Saclay |
| Date de démarrage 01/01/77 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 01/12/80 | Dernière mise à jour 12/77 | |

1 - Objectif général :

Assurer, durant la période d'utilisation et in situ, le contrôle du bon fonctionnement des capteurs de natures diverses utilisés dans les dispositifs de protection des réacteurs à eau pressurisée.

2 - Objectifs particuliers :

Accroître la disponibilité et la sécurité de fonctionnement des dispositifs de protection et assurer par conséquent une meilleure utilisation de l'installation globale.

4 - Etat de l'étude :

Au cours de l'année 1977 des moyens expérimentaux ont été développés pour mesurer le temps de réponse des capteurs de température et de débit des centrales à eau pressurisée.

Une boucle de test a été construite pour fonctionner dans les conditions de laboratoire (0° à 50°C) et pour des vitesses de l'eau variant entre 0 à 3 m/s.
Une étude comparative a été entreprise pour évaluer les performances des différentes techniques donnant le temps de réponse dans les conditions du laboratoire.

Une campagne d'essai a été menée pour déterminer l'influence de la vitesse du fluide sur le temps de réponse d'une boucle Rosemount 176 KF. On a mis en évidence le temps limite de réponse.

Une méthode de mesure du temps de réponse utilisant l'information de température fournie par un thermocouple rapide placé près de la sonde à tester a été mise au point : elle utilise les techniques de l'identification du modèle liant les signaux du thermocouple et de la sonde.

Une méthode de mesure du temps de réponse par les techniques d'analyse du bruit naturel a également été développée. Elle repose sur l'analyse de Fourier et sur la modélisation du bruit dans le domaine temporel. Cette technique a été utilisée en outre sur des capteurs de la centrale de Tihange ( température et débit).

<table>
<tr><td colspan="2">143-1 -02</td><td>8</td></tr>
<tr><td colspan="2">Titre<br>Qualification des appareillages de mesure et des matériels utilisés dans un réacteur dans des conditions de fonctionnement normales suivies en fin de vie d'un ADR.</td><td>Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/IPSN - EDF - FRA</td></tr>
<tr><td colspan="2">Titre (anglais)<br>Qualification of class 1 E equipment for nuclear power plants.</td><td>Organisme exécuteur<br>CEA - EDF - FRA<br><br>Responsable<br>CEA - EDF - FRAM</td></tr>
<tr><td>Date de démarrage 1/1/78</td><td>Etat actuel<br>en cours</td><td rowspan="2">Scientifiques<br>BOUTEILLER , BUISSON, LAIZIER (CEA)<br>BARBET, FROIDEFOND (EDF)<br>CHAUVIN (FRAMATOME)</td></tr>
<tr><td>Date prévue d'achèvement 1/1/82</td><td>Dernière mise à jour 1/1/78</td></tr>
</table>

1 - Objectif général :

Les études concernant la qualification des matériels intervenant directement dans la sûreté des réacteurs PWR ont été initiées, et menées, depuis environ deux ans, par deux structures indépendantes :

- D'une part EDF (SEPTEN) en liaison avec FRAMATOME et le CEA (LCRI) a mis sur pied une première structure chargée de définir les conditions d'essai de ces matériels (irradiation, température, pression, aspersion...).
- D'autre part, le CEA (SCAPR) en collaboration avec EDF (E,R) et plus de vingt industriels a mis sur pied une deuxième structure chargée de rassembler les données expérimentales qui permettront de déterminer les conditions d'essais et d'extrapolation des tests de qualification nucléaire.

Le DSN a décidé de participer aux deux structures. Une première réunion générale a eu lieu en Juillet 1977. Au cours de cette réunion, il a été décidé de créer un groupe de coordination générale chargé de définir les profils d'essais et d'être informé de la méthodologie mise en place. La coordination générale, sur le plan de la sûreté, est assurée par le DSN. La préparation, l'exécution et l'interprétation des essais reste du ressort des fournisseurs, des clients et des exécutants.

.../

2 - Etat des études :

De nombreux polymères (matériaux organiques de synthèse) entrent dans la réalisation de composants utilisés dans les réacteurs nucléaires (joints, câbles, moteurs, connecteurs, relais, électrovannes, peinture, etc...). Ces matériaux sont soumis à des agressions (température, pression, irradiation, attaque chimique, vibrations, feu, etc...) de façon permanente ou accidentelle.

Depuis quelques années, des études concernant les effets de ces agressions sont entreprises par un certain nombre de laboratoires tant en France qu'à l'étranger en vue de la définition et de la qualification de ces matériaux dans les installations nucléaires.

Au C.E.A. :

a) La Section de Chimie Appliquée des Polymères et des Rayonnements (CAPRI) de la Division de la Chimie a soumis à des essais de tenue à la température, à l'irradiation et à l'aspersion de nombreux matériaux pour le compte d'entreprises extérieures ou du C.E.A.

A la demande du DSN une étude sur la tenue à l'irradiation des élastomères au silicone a été menée en 1976 (rapport CAPRI N° 006 du 9/1/77), étude ayant mis en évidence la dégradation des propriétés mécaniques de ce matériau pour des doses intégrées relativement faibles.
Lancée en 1977 , une action intitulée " Essais de qualification nucléaire pour prévisions à long terme " se poursuit. Cette étude s'inspire au départ des profils d'essais proposés par la norme américaine IEEE 323, elle devrait aboutir à la définition des tests simultanés ou séquentiels auxquels devront être soumis tous les matériels utilisés à proximité du réacteur.

b) Le Laboratoire de Contrôle des Rayonnements Ionisants (LCRI) de la Division de la Chimie, en 1976 et 1977, dans le cadre d'une fiche d'action intitulée " Evaluation des appareillages utilisés dans les réacteurs nucléaires ", a déterminé les types de matériels à soumettre aux irradiations et défini les moyens de dosimétrie et neutrons à mettre en oeuvre.

Une fiche d'action intitulée " Qualification nucléaire des matériels utilisés dans les centrales nucléaires " a été établie pour 1978. Elle doit permettre au LCRI de participer à la préparation matérielle des essais d'irradiation et à leur exécution au niveau de la dosimétrie associée.

c) Les Services d'Electronique de Saclay auxquels ont été sous-traitées les mesures de l'évolution des caractéristiques électriques des câbles soumis à l'irradiation au CAPRI (rapport interne SES-SAI 77/185)
Dans le cadre de la fiche action établie pour 1978 et intitulée " Définition des essais de vieillissement sous irradiation " ils procèderont, à la demande du DSN, à la définition des conditions d'essais et aux mesures électriques sur les câbles bas niveaux. Ils assisteront, sur leur demande, CAPRI et LCRI dans leurs essais.

.../

3 - Relation avec d'autres études :

Cette action regroupe les actions suivantes :

143-1-03- Essais de qualification nucléaire pour prévisions à long terme (matériaux polymériques)

143-1-04- Définition des essais de vieillissement sous irradiation.

143-1-05- Appareillages de mesures utilisés dans les réacteurs nucléaires.

4 - Informations additionnelles :

voir tableau I

TABLEAU 1

| FONCTION | CEA - DSN | EDF - SEPTEN | FRAMATOME (fournisseur) | CEA LCRI-CAPRI |
|---|---|---|---|---|
| Coordination générale | x | x | x | |
| Définition des ambiances dans les réacteurs | x | x | x | |
| Interprétation et élaboration de normes | x | x | x | |
| Définition des profils d'essais | x | x | x | (x) |
| Préparation matérielle des essais | | x | x | x |
| Exécution des essais : | | | | |
| Température | | x | | x |
| Pression | | x | | x |
| Irradiation | | | | x |
| Attaque chimique | | x | | x |
| Vibrations-séismes | | x | | x |
| Feu | | x | | x |
| Mesures liées aux essais : mécaniques | | x | x | x |
| électriques | | x | | x |
| dosimétrie | | | | x |
| Rédaction des P.V. d'essais | | x | x | x |

2ème structure :

| | CEA-DSN | EdF-E.R. | FRAMATOME | CEA-CAPRI | CEA-SES | FABRICA |
|---|---|---|---|---|---|---|
| Coordination générale (ambiances-normes d'essais-etc...) | x | x | x | x | | x |
| Préparation des essais } Choix des matériels } groupes de travail | | x | | x | | x |
| Exécution des essais | | | | x | x | |
| Rédaction des P.V. d'essais | | | | x | | |

Groupes de travail :

1°/ moteurs et isolants
2°/ joints-pièces mécaniques-raccordements-connexions)
3°/ matériels d'automatisme industriels
4°/ câbles
5°/ peintures

x participant de plein droit
(x) membre invité - conseiller technique du DSN
x initiateur de l'opération
x sous traitance technique
x à titre d'observateur

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| 143-1 - 03 | | 8 |
|---|---|---|
| Titre<br>Essais de qualification nucléaire pour prévision à long terme (matériaux polymériques) | | Pays<br>FRANCE<br>Organisme directeur<br>CEA / DgCS |
| Titre (anglais)<br>Ageing prevision nuclear qualification tests for polymeric materials. | | Organisme exécuteur<br>CEA/SCAPR<br>Responsable<br>SCAPR-SACLAY |
| Date de démarrage 1/4/77 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 1/1/82 | Dernière mise à jour 12/77 | |

1 - Objectif général :

- Déterminer les conditions d'essais accélérés permettant d'évaluer la durée de vie d'un matériau plastique soumis au vieillissement sous irradiation et à l'épreuve du feu.

- Rassembler les informations relatives à divers matériaux plastiques soumis à de telles actions, matériaux d'usage courant dans les installations nucléaires.

2 - Objectifs particuliers :

- Déterminer l'évolution dans le temps de matériaux au cours du service normal (40 ans à $10^1$ ou $10^2$ rad. $h^{-1}$) suivi d'un ADR se produisant en fin de vie $10^6$ rad. $h^{-1}$).

- Les essais porteront sur les matériaux constituant les câbles, les joints et pièces mécaniques, les moteurs, les automatismes, les connecteurs.

.../

3 - Installations expérimentales et programme :

- Laboratoires de chimie macromoléculaire de physico-chimie et d'essai des plastiques
- Sources de cobalt de 20.000 et 200.000 Ci
- Accélérateurs de 0,5 à 3 Mev et de 300 hV

4 - Etat de l'études :

Un groupe d'étude a été mis en place comprenant :

- un comité de coordination
- 5 groupes de travail par famille de produits, à savoir :
  - moteurs et isolants.
  - joints-pièces mécaniques-matériel de raccordement et connexions.
  - matériels d'automatismes industriels.
  - câbles.
  - peintures.
- Un profil d'essai EDF, adopté à titre provisoire et modulé suivant la durée réelle d'utilisation a été établi par produit.
- Chaque groupe de travail a sélectionné les matériaux et matériels à soumettre aux essais.
- Des équipements ont été acquis en plus des irradiateurs au cobalt 60 (20.000 et 156 000 Ci) :
  fours
  analyseurs de gaz de radiolyse
  dynamomètres pour essais mécaniques
  -matériels pour les contrôles électriques

Les matériaux ont été rassemblés et conditionnés en vue des mises en irradiation

Les premiers essais sont prévus pour le début de 1978.

5 - Prochaines étapes :

- Réalisation des différentes séquences d'essais sur les matériaux.
- Dépouillement, analyse des résultats et conclusions.

6 - Relation avec d'autres études :

Cette action entre dans le cadre plus général de la fiche d'action : "Qualification des appareillages de mesure et des matériels utilisés dans un réacteur, dans des conditions de fonctionnement normales suivies en fin de vie d'un ADR."

.../

| 143-1 -04/4114-03 | | | 8 |
|---|---|---|---|
| Titre<br><br>Définition des essais de vieillissement sous irradiation des composants de circuits de sécurité et de contrôle. | | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA / DgCS |
| Titre (anglais)<br><br>Definition of ageing qualification tests. | | | Organisme exécuteur<br>CEA/SES- Saclay<br><br>Responsable<br>SES - Saclay |
| Date de démarrage 01/04/77 | Etat actuel en cours | | Scientifiques |
| Date prévue d'achèvement 01/01/82 | Dernière mise à jour 12/77 | | |

1 - Objectif général :

Le bon fonctionnement des ensembles de sécurité et de contrôle neutronique des réacteurs implique que certaines caractéristiques particulières des composants entrant dans la composition de ces ensembles de mesure soient respectées. Il s'agit par exemple du potentiel de microclaquage, de la porosité des isolants,etc. En général, les essais après que les composants aient été soumis aux contraintes ne comportent pas de mesures de la modification de ces caractéristiques. Il importe donc de développer les méthodes permettant de combler cette lacune.

2 - Objectifs particuliers :

Avant l'application aux composants des conditions d'environnement de l'accident de référence, il faut faire subir à ces derniers un vieillissement accéléré simulant le vieillissement normal en ambiance de centrale pendant 40 ans. La méthode de vieillissement accéléré pour être valable devra correspondre à une altération des caractéristiques spécifiques déterminées par rapport à la fonction que le composant doit assurer. Suivant l'utilisation qui en est faite, le ou les critères de vieillissement d'un élément peuvent être très variés : Par exemple, le critère déterminant la durée de vie d'un câble d'alimentation ne peut pas être le même que celui d'un câble véhiculant les impulsions issues d'un détecteur.
Dans cette action, on se propose de travailler en collaboration avec les équipes chargées des études de vieillissement accéléré et des irradiations pour examiner si les méthodes proposées seront valables pour les matériels entrant dans le contrôle neutronique et la sécurité des réacteurs. On établira des méthodes d'essais après vieillissement accéléré. En dehors des mesures classiques (tenue mécanique, rigidité

.../

diélectrique, etc.) effectuées sur certains composants soumis à l'ADR, il importe, en particulier pour les câbles, de faire des mesures particulières pour ceux entrant dans la composition des ensembles de sécurité. Ces mesures font appel à des techniques spéciales ignorées pour les applications classiques.
Il s'agit par exemple de mesurer dans les isolants l'apparition de microdécharges dont l'amplitude est de l'ordre de $10^{-13}$C et qui peuvent être prises en compte au même titre que les impulsions dues aux neutrons venant du détecteur. Le vieillissement des alliages entrant dans la composition du blindage des câbles doit également être contrôlé (mesure d'impédance de transfert) etc.
Des appareillages permettant d'effectuer les essais doivent être développés et les méthodes de mesure étudiées et mises au point.

4- Etat de l'étude :

- essais effectués sur des échantillons de câbles de puissance ayant subi les contraintes thermodynamiques et d'irradiation correspondant au vieillissement et à l'ADR.
  (Etude de l'évolution de la résistance d'isolement, de la rigidité diélectrique, de la tension de claquage)

- Essais en cours de tenue au feu de câbles suivant les normes NFC 32070.

5 - Prochaines étapes :

- poursuite de la mise au point des méthodes de mesure des caractéristiques électriques des matériaux et matériels ayant subi des contraintes thermodynamiques et d'irradiation.

- Participation à l'élaboration de normes de qualification électrique de matériaux et matériels ayant subi ces contraintes.

6 - Relation avec d'autres études :

Cette action entre dans le cadre plus général de la fiche d'action : Qualification des appareillages de mesures et des matériels utilisés dans un réacteur dans des conditions de fonctionnement normales suivies en fin de vie d'un ADR.

| 143-1 -05/4115-06 | | 8 |
|---|---|---|
| Titre<br>Qualification des appareillages de mesure utilisés dans les réacteurs nucléaires. Réalisation des essais . | | Pays<br>FRANCE<br>Organisme directeur<br>CEA |
| Titre (anglais)<br>Measure instrumentation in nuclear reactors. | | Organisme exécuteur<br>CEA/DRIS-LCRI<br>Responsable<br>LCRI - Saclay |
| Date de démarrage 01/01/75 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 31/12/ 82 | Dernière mise à jour 12/77 | |

1 - Objectif général :

Evaluer les matériels utilisés dans les réacteurs PWR sous ambiance de rayonnement gamma et neutrons, et en ambiance d'accident (LOCA-ECCS)

2 - Objectifs particuliers :

Les essais porteront sur des capteurs de pression et de température, des câbles, des vannes et des moteurs.

3 - Installations expérimentales et programme :

TRITON - CAPRI
Enceintes pour réaliser des ambiances d'accidents.

.../

4 - Etat de l'étude :

- Les caractéristiques d'ambiance et les méthodes de simulation de ces ambiances sont en cours de définition.

- Les méthodes dosimétriques sont choisies (film au triacetate de cellulose).

5 - Prochaines étapes :

- Participation à l'étude et à la réalisation des enceintes pour les essais thermodynamiques et d'irradiation.

- Choix des matériels à essayer, des conditions et des séquences d'essais.

- Participation à l'exécution des essais (dosimétrie).

6 - Relation avec d'autres études :

Cette action entre dans le cadre plus général de la fiche d'action :
"Qualification des appareillages de mesure et des matériels utilisés dans un réacteur dans des conditions de fonctionnement normales suivies en fin de vie d'un ADR."

7 - Documents de référence : - rapports internes non disponibles

| 143-1 -06/4114-10<br>153-1 -02/4114-10 | | 8 |
|---|---|---|
| Titre<br><br>Etude du comportement des câbles auto-extinguibles aux silicones lors d'un A.D.R. | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br><br>Study of method for aging qualification of silicons non flamme propagating cable. | | Organisme exécuteur<br>CEA/DCA-SCAPR<br>Saclay<br><br>Responsable<br><br>DSN - SETSSR - Fontenay |
| Date de démarrage 01/06/76 | Etat actuel<br>en cours | Scientifiques.<br><br>(DCA) |
| Date prévue d'achèvement 01/01/79 | Dernière mise à jour 1/78 | |

1 - Objectif général :

Détermination des cinétiques de vieillissement des câbles aux silicones sous irradiation.

- Objectifs particuliers :

Tenue des câbles auto-extinguibles aux silicones au feu et à l'A.D.R. dans un réacteur PWR et dans un réacteur à neutrons rapides.

4 - Etat de l'étude :

L'étude de l'évolution des consommations d'oxygène a été réalisée sur 4 échantillons :

1 élastomère methyl vinyl silicone.
1 élastomère methyl phenyl vinyl silicone.
1 gamme à base methyl phenyl vinyl silicone.
1 élastomère methyl phenyl silicone.

.../

Les débits de dose étaient de $10^4$, $10^5$ et $3.10^5$ rad. $h^{-1}$.
La dose intégrée était de 36 Mrad coupée par une phase de vieillissement thermique de 240 heures à 135°C.
La consommation d'oxygène semble varier peu en fonction du débit de dose et les essais de vieillissement pourraient être effectués au débit de dose maxima de $3.10^5$ rad. $h^{-1}$

5 - Prochaines étapes :

- Etude de l'évolution des caractéristiques mécaniques et électriques des matériaux soumis aux contraintes thermodynamiques et d'irradiations correspondant au vieillissement à l'A.D.R. (en association avec les S.E.S. pour les mesures électriques).

- Evaluation des qualités d'auto-extinguibilité.

6 - Relation avec d'autres études :

- Voir fiches précédentes

| 143-1 -07/4114-02<br>153-1 -03 | | 8 |
|---|---|---|
| Titre<br><br>Critères de sûreté pour l'utilisation des systèmes séquentiels dans le système de protection. | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA / DgCS |
| Titre (anglais)<br><br>Safety criteria for using microprocessor in the protection system. | | Organisme exécuteur<br>CEA/SES-(Saclay)<br><br>Responsable<br>SES-Saclay |
| Date de démarrage 01/01/77 | Etat actuel<br>à lancer | Scientifiques |
| Date prévue d'achèvement 01/02/79 | Dernière mise à jour 15/02/77 | |

1 - Objectif général :

Mise au point d'une procédure permettant de s'assurer de la fiabilité des logiciels utilisés dans les systèmes de protection par calculateurs numériques.

2 - Objectifs particuliers :

Mise au point de procédures de test pour la qualification du logiciel au cours de son élaboration. Mise au point de systèmes de détection de pannes sur le produit final de façon à obtenir un fonctionnement "FAIL SAFE ".

3 - Etat de l'étude :

Avancement à ce jour :
- Un système est en cours de développement au laboratoire pour étudier le problème du dialogue entre deux calculateurs au moyens d'une unité d'échange.
- l'essai de la maquette est prévue en septembre 78.

.../

4 - Prochaines étapes

Application d'une procédure de qualification sur la maquette .

143-1 -08/4114-01 8.

| | |
|---|---|
| Titre<br><br>Utilisation de calculateurs dans les systèmes de protection. | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA / DgCS |
| Titre (anglais)<br><br>Use of computers in protection system | Organisme exécuteur<br>CEA/LETI (GRENOBLE)<br><br>Responsable<br><br>LETI - Grenoble |
| Date de émarrage 1/1/75 \| Etat actuel en cours | Scientifiques |
| Date prévue d'achèvement 1/1/80 \| Dernière mise à jour 12/77 | |

1 - Objectif général :

Mise au point d'un système de surveillance des paramètres importants pour la sûreté à base de micro-processeurs.

2 - Objectifs particuliers :

Les calculateurs classiques n'étant pas fiables pour un système de protection, on est donc conduit à appliquer la technique de redondance en utilisant une configuration à plusieurs calculateurs qui conserve un fonctionnement correct même en cas de panne d'un ou plusieurs éléments.
La configuration retenue utilise 3 calculateurs associés à un organe de décision en 2/3.
Les calculateurs possèdent un système de contrôle en marche de son bon fonctionnement. Lorsqu'il détectera une défaillance susceptible de mettre en cause l'accomplissement d'une fonction de protection, ce système met le calculateur hors service, en panne sûre, et transmet les informations relatives à la défaillance et aux actions entreprises pour en limiter les conséquences.

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4 - Etat de l'étude :

a - matériel

Le couplage entre l'organe de décision en 2/3 et un calculateur a été réalisé courant 1976. La mise en oeuvre de cet ensemble et les essais qui ont suivi ont permis de dégager les conclusions suivantes :

- Il semble intéressant de pouvoir étendre l'organe de décision à des systèmes à 4 éléments et d'adopter une logique en 2/4.
- Il est souhaitable d'améliorer la fiabilité de l'organe de décision actuel. Deux aspects sont particulièrement considérés :
  1 - Tout en gardant la possibilité d'un test permanent de l'organe de décision, l'autotest devra être réalisé de manière à le rendre, si possible, d'une complexité et d'une fiabilité comparable à celle de l'organe de décision.

  2 - Une meilleure protection de cet organe de décision vis-à-vis des parasites extérieurs est envisagée par l'adjonction d'interfaces réalisant son isolement.

  Les études seront poursuivies dans ce sens courant 1978.

b - logiciel

Les activités ont été menées dans trois directions :

- Définition d'un ensemble de règles relatives à la structuration du logiciel et à son écriture. Ce travail a été concrétisé par la rédaction d'un document au cours du premier trimestre 1977.
- Analyse et définition de la structure globale du système informatique à réaliser dans le cadre du projet.

  L'organisation retenue est de type multitâches comportant une tâche de contrôle de l'application (traitements et élaboration de la décision), une tâche de test, des tâches d'échanges et le moniteur de contrôle du système (initialisation, alarme, gestion des périphériques, gestion des requêtes superviseur).
  Les liaisons entre les divers éléments ont été définies. La programmation du système sera entreprise dès le début 1978.
- Définition et réalisation d'un logiciel de transmission entre calculateurs.

  La réalisation de la maquette met en oeuvre 3 processeurs (1 Solar 40 - 2 Solars 05). Seul le premier processeur dispose des moyens matériels et logiciels d'interface avec l'opérateur. Dans une première phase, l'acquisition des mesures et états sera simulée et générée par SOLAR 40 qui offre une puissance de calcul supérieure. L'ensemble de ces informations sera transmise aux autres processeurs par téléchargement.

  Pour réaliser ces fonctions et pour faciliter l'utilisation des Solar 05, le couplage entre les calculateurs a été réalisé par liaison asynchrone. Divers modes de fonctionnements ont été testés. Le logiciel de gestion des échanges a été défini, sa programmation est en cours.

  L'ensemble de ce logiciel devrait être opérationnel dans un délai d'environ 2 mois. Il sera par la suite intégré au système implanté sur la maquette.

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5 - Prochaines étapes :

c) Etudes sur la fiabilité des modules et composants à utiliser et surtout sur la sécurité et disponibilité du système entier choisi.
Ces études montreront sans doute encore des points faibles dans le système qui doivent être éliminés par la suite et ils devront fournir la preuve de la capacité du système à remplir les exigences demandées.

d) Un prototype du système entier informatique doit être réalisé et entièrement testé à l'aide d'un dispositif de test, par exemple un calculateur hybride qui permettrait d'effectuer, en principe toutes les combinaisons des valeurs d'entrée, mais en fait, à cause du grand nombre des combinaisons ($10^{64}$), seulement une petite partie de ces possibilités.

e) Statistique des défaillances déjà apparues qui permettrait ensemble avec une étude des défaillances (arbres de défaillance, ambiguïté des défaillances, probabilité d'occurrence des défaillances particulières, etc...) d'améliorer encore le système entier.

7 - Documents de référence : - rapports internes non disponibles

| 143-1 -09/4130-01 | | 8 |
|---|---|---|
| Titre<br>Méthodes et appareillages d'essais pour le contrôle de l'instrumentation. | | Pays<br>FRANCE<br>Organisme directeur<br>CEA / DgCS |
| Titre (anglais)<br>Methods and test devices for instrumentation control. | | Organisme exécuteur<br>CEA/SES (Saclay)<br>Responsable<br>SES - Saclay |
| Date de démarrage 01/01/74 | Etat actuel en cours | Scientifiques |
| Date prévue d'achèvement 01/12/78 | Dernière mise à jour 12/77 | |

1 - Objectif général :

Contrôle de la susceptibilité aux perturbations électriques des installations de contrôle neutronique et de détection de rupture de gaine. Méthode de mesure de microphonie et microclaquage.

2 - Objectifs particuliers :

Etablissement de normes sur les niveaux de perturbations électriques acceptables pour l'instrumentation (capteur, câble) entrant dans le système de protection.

4 - Etat des études :

Les mesures effectuées en 1976 sur la tranche I de Fessenheim ont été poursuivies en 1977 sur la tranche II.
Elles ont permis de conclure que :
- les réponses obtenues sur le matériel équipant les 2 tranches étaient similaires.
- la protection des équipements vis à vis des perturbations électriques était satisfaisante.

.../

L'immunité aux parasites est à présent caractérisée par l'atténuation du courant gaine de préférence à l'impédance de transferts. Une protection suffisante correspond à une atténuation minima du courant gaine de 100 dB. A titre indicatif les valeurs mesurées sur Fessenheim pour les différentes chaînes vont de 112 à 116 dB.

La méthode adoptée pour effectuer les mesures sur Fessenheim II a été allégée par rapport à celle utilisée sur Fessenheim I, grace à l'emploi d'un analyseur de spectre. Le temps des essais se trouve considérablement raccourci et la précision des mesures augmentée.

5 - Prochaines étapes :

Etablissement de normes pour la sensibilité aux parasites des chaînes de mesure entrant dans le système de protection.

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

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| 143-1 -10/4114-10 | | 8 |
|---|---|---|
| Titre<br><br>Vérification des chaines de mesures neutroniques. | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br><br>Checking of nuclear instrumentation. | | Organisme exécuteur<br>CEA/SES (Saclay)<br><br>Responsable<br>DSN-SETSSR<br>Fontenay |
| Date de démarrage 01/01/76 | Etat actuel<br>en cours | Scientifiques<br><br>(SES) |
| Date prévue d'achèvement 31/12/78 | Dernière mise à jour 12/77 | |

1 - Objectif général :

- Déterminer, le temps de réponse des chaînes de mesures nucléaires pour différents accroissements de réactivité, afin de vérifier que le matériel est apte à remplir ses fonctions lors d'accidents sur un réacteur donné.
- Vérifier la precision et la linéarité des chaînes de mesures nucléaires.

2 - Objectifs particuliers :

Mesure des caractéristiques des diverses tensions de sortie d'une chaine lin-log pour courant continu type SPALIS B13 et d'une chaine périodemètre à impulsions et fluctuations type SPIFSB1 comprenant une voie SPISB5 et une voie SPFSB2.

.../

4 - Etat de l'étude :

Un retard a été pris et l'action devrait se terminer courant 78 ; le financement est dès à présent assuré puisqu'il a été déjà programmé en 1976 et 1977.

5 - Prochaines étapes :

Des essais seront effectués en vraie grandeur sur le réacteur source HARMONIE, spécialement conçu pour des mises au point et des étalonnages d'instruments nucléaires dans des spectres de neutrons rapides ou thermiques.

La chaine neutronique à tester sera située dans la salle expérimentateur ou dans le hall pile à température et hygrométrie ambiante, soit environ 20° C et humidité relative 70% , durée de fonctionnement préalable 30 mn, circuits de tarages et calibrages vérifiés avant l'expérience, position normale d'utilisation.
On s'assurera qu'aucune induction magnétique ou champ électromagnétique, ne viendra perturber les mesures.

7 - Documents de référence disponibles :

" Ensemble de mesure neutronique à grande dynamique - combinaison impulsion fluctuation ", Y. PLAIGE, VO MAJ-VUONG - Bulletin d'Informations Scientifiques et Techniques N° 195, septembre 1974.

| 143-1 -11/4114-10 | | 8 |
|---|---|---|
| Titre<br>Essais des compteurs de démarrage des centrales PWR. (vieilissement) | | Pays<br>FRANCE<br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br>Test of proportional counters used in PWR reactors. (ageing) | | Organisme exécuteur<br>CEA/SES (Saclay)<br>Responsable<br>DSN/SETSSR- Fontenay |
| Date de démarrage 01/06/76 | Etat actuel en cours | Scientifiques<br>(SES) |
| Date prévue d'achèvement 31/12/78 | Dernière mise à jour 12/77 | |

1 - Objectif général :

Essais de fiabilité des compteurs équipant les chaines de démarrage des PWR, dans les conditions spécifiques d'utilisation.

2 - Objectifs particuliers :

Comparaison des compteurs BF3 et des compteurs à dépôt de bore.

4 - Etat de l'étude :

L'irradiation dans le réacteur TRITON a montré :

- que les compteurs BF3, ont fonctionné correctement jusqu'à des flux intégrés de l'ordre de $2.10^{16}$ n/cm3. Ensuite des phénomènes de microclaquage sont apparus et les essais ont dû être interrompus.

.../

- Par contre les compteurs à dépôt de bore n'ont pas répondu aux aspérances. En effet des fuites dues à des phénomènes de corrosion sont apparues à des flux de l'ordre de $10^{15}$ n/cm3.
  Un rapport d'essai paraîtra prochainement et explicitera tous ces phénomènes.

5.- Prochaines étapes :

Il est nécessaire de revoir la conception des compteurs à dépôt de bore. De nouveaux essais de qualification d'une année dans TRITON devrait valider la nouvelle solution retenue.

| TITLE 1 (original language) Diagnostica e Sicurezza tramite Analisi di Rumore | Classification 8 - 14 |
|---|---|
| TITLE 2 (english) Reactor Safety Studies Via Noise Analysis | Country: ITALY<br>Sponsor: CNEN<br>Organisation: CNEN |
| Date initiated 1/1/1975<br>Date completed in progress<br>Last updating June 1976 | Project Leader<br>F. Norelli |

1. General aim. Correct performance of in-core and ex-core instrumentation to safety monitoring and early detection of abnormal operating conditions and/or malfunctions.

2. Particular objectives :

   2.1 Set-up of a general theory for multi-detector reactor noise analysis in ergodic conditions, non equilibrium conditions and during pulsed experiments.

   2.2 Experimental determination of nuclear reactor kinetic parameters and control of multiplying assemblies that must be kept subcritical.

   2.3 Design and realization of special instrumentation: Stochastic Indicator Meters.

   2.4 Set-up of calculational codes for analyzing data from various types of sensors.

3. Experimental facilities : four light-water reactors (RANA, ROSPO, RITMO, TRIGA) and a copper-reflected highly-enriched-uranium fast reactor (TAPIRO).

4. Project status :

   4.1 A unified theory of reactor neutron noise analysis techniques has been developed. It is shown to generate all the neutron noise analysis techniques (ca 20) developed in 30 years of nuclear reactor physics.

   4.2 A Stochastic Indicator Meter for digital signals has been developed and will be operated within the end of 1976.

| TITLE 1 (original language)<br>Diagnostica e Sicurezza tramite Analisi di Rumore | Classification<br>8 - 14 |
|---|---|

5. Next steps :

5.1 The general theory will soon be expanded to include problems related to heat transfer and to a preliminary analysis of multi-zone model reactor.

5.2 A Stochastic Indicator Meter for analog signals will be realized within the first semester of 1977.

6. Relation to other projects :

Terms of cooperation are going to be defined with AB Atom-energy (Sweden), Halden Project (Norway), CEA (France).

7. Reference documents :

N. PACILIO, V.M. JORIO, F.NORELLI, R.MOSIELLO, A.COLOMBINO Toward a unified theory of reactor neutron noise analysis techniques - Annals of Nuclear Energy (in print).

R.MOSIELLO - Due algoritmi per il calcolo della derivata n-esima di una funzione composta - CNEN Report RT/FI (75) 12.

R.MOSIELLO - DERN : un programma per il calcolo della derivata n-esima di una funzione composta - CNEN Report RT/FI (75) 13.

F.NORELLI, R.MOSIELLO - NORMOS : un programma per il metodo di massima somiglianza - CNEN Report RT/FI (76).

8. Degree of availability :

Documents are not classified material, they can be requested from one of the authors by the following address:
LTCR, CSN CASACCIA, CP 2400 ROME, ITALY.

| Title 1 (Original language) Diagnostica e Sicurezza tramite Analisi di Rumore (Parte A) | Classification [8] - 14 |
|---|---|
| Title 2 (English) Safety and Diagnostics Via Noise Analysis (Part A) | Country ITALY<br>Sponsor CNEN<br>Organisation CNEN |
| Date initiated 1/1/1975<br>Date completed in progress<br>Last updating June 1977 | Project Leader<br>N. Pacilio |

1. General aim

Correct performance of in-core and ex-core instrumentation for safety monitoring and early detection of abnormal operating conditions and/or malfunctions.

2. Particular objectives

2.1. Set-up of a general theory for multi-detector reactor noise analysis in ergodic conditions, non-equilibrium conditions and during pulsed experiments.

2.2. Set-up of calculational codes for analyzing data from various types of sensors.

3. Experimental facilities

Light-water reactors (RANA, RITMO, TRIGA) and a copper-reflected highly-enriched-uranium fast reactor (TAPIRO).

4. Project status

A unified theory of reactor neutron noise analysis techniques has been developed. It is shown to generate all the neutron noise analysis techniques (ca 20) developed in 30 years of nuclear reactor physics.

5. Next steps

The general theory will soon be expanded to include problems related to heat transfer and to a preliminary analysis of multizone model reactor.

6. Relation to other projects

Terms of cooperation are going to be defined with Caorso (AMN-ENEL), Halden Project (Norway), CEA (France).

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| Title 1 (Original language) | Classification |
| --- | --- |
| Diagnostica e Sicurezza tramite Analisi di Rumore (Parte A) | 8 - 14 |

7. Reference documents

N. PACILIO, V.M. JORIO, F. NORELLI, R. MOSIELLO, A. COLOMBINO. Toward a unified theory of reactor neutron noise analysis techniques - Annals of Nuclear Energy, 3, 239 (1976).

8. Degree of availability

Documents are not classified material, they can be requested from one of the authors by the following address: RIT, CSN-Casaccia, C.P. 2400, Rome, Italy.

| Title 1 (Original language)<br>Diagnostica e Sicurezza tramite Analisi di Rumore (Parte B) | Classification<br>8 - 14 |
|---|---|
| Title 2 (English)<br>Safety and Diagnostics Via Noise Analysis (Part B) | Country ITALY<br>Sponsor CNEN<br>Organisation CNEN |
| Date initiated 1/1/1976<br>Date completed in progress<br>Last updating June 1977 | Project Leader<br>A. Serra |

1. General aim

Correct performance of in-core and ex-core instrumentation for safety monitoring and early detection of abnormal operating conditions and/or malfunctions.

2. Particular objectives

2.1. Experimental determination of nuclear reactor kinetic parameters and control of multiplying assemblies that must be kept subcritical.

2.2. Design and realization of special instrumentation: Stochastic Indicator Meters.

3. Experimental facilities

Light-water reactors (RANA, RITMO, TRIGA) and a copper-reflected highly-enriched-uranium fast reactor (TAPIRO).

4. Project status

Stochastic Indicator Meters for digital signals have been developed and will be operated within the end of 1977.

5. Next steps

5.1. Instrumentation for (1) measuring time constants and dynamic characteristics of reactor systems; (2) assessing correct operation and diagnosing expected or unexpected malfunctions in measuring apparati.

5.2. Self-checking instrumentation for nuclear particle detection apparati.

5.3. A Stochastic Indicator Meter for analog signals will be realized within 1977.

9-4

| Title 1 (Original language) | Classification |
|---|---|
| Diagnostica e Sicurezza tramite Analisi di Rumore (Parte B) | 8 - 14 |

6. Relation to other projects

Terms of cooperation are going to be defined with Caorso (AMN-ENEL), Halden Project (Norway), CEA (France).

7. Degree of availability

Documents are not classified material, they can be requested from one of the authors by the following address: RIT, CSN-Casaccia, C.P. 2400, Rome, Italy.

| Title 1 (Original language)<br>Sistema di controllo di reattori con barre bifasi | Classification<br>8 |
|---|---|
| Title 2 (English)<br>Reactor control system by two-phase rods | Country ITALY<br>Sponsor CNEN<br>Organisation CISE |
| Date initiated 1965<br>Date completed 1976<br>Last updating April 1977 | Project Leader<br>UTM (CISE) |

1. General aim: development of a new control system for pressure tube reactors. Reactivity control is obtained by density variation of a two-phase mixture (borated water and oxygen) flowing in vertical U-tubes through the reactor core.

3. Experimental facilities and programme

3.1. Experimental facilities

- BB2: out of pile facility simulating one control rod.
- BB3: facility having two control rods operating in RB 3 reactor (Montecuccolino - Bologna, CNEN)

3.2. Programme

- Tests for determining steady state features of two-phase rod system.
- Dynamic behaviour of the two-phase control system in open loop operation.
- Overall tests of the control system in an actual reactor (at zero power).
- Codes development for predicting purposes.

4. Project status

The overall programme has been completed. The feasibility of the new control system has been shown and the design features have been enhanced.

6. Reference documents


1) M. Luminari, G. Masini, F.A. Tacconi " Sviluppo del sistema a fluido bifase per la regolazione rapida della potenza neutronica del reattore prototipo CIRENE: impianto fuori pila BB-2" CISE R-300, 1970.

2) C.A. Marchiondelli, G. Masini, M. Perego, F.A. Tacconi "Sistema delle Barre Bifasi: impianto prototipo BB-3 per le prove di controllo di un reattore a potenza zero" CISE R-337, 1973.

3) F.A. Tacconi "Il programma BUANA '74 per la simulazione del sistema Barre Bifasi" CISE R-350, 1974

4) R. Granzini, G. Masini, A. Vanossi, A. Venturi, G. Zappellini "Prove di regolazione del reattore RB 3 col sistema delle barre Bifasi. Analisi dei risultati della seconda campagna sperimentale" CISE R-385, Dic. 1976.

946

| Title 1 (Original language) | Classification |
| --- | --- |
| Sistema di controllo di reattori con barre bifasi | 8 |

7. Degree of availability: to a limited extent

947

| | |
|---|---|
| PROJECT TITLE : Reactor Safety Studies via Noise Analysis | CLASSIFICATION 14 - 8 |
| SPONSORING COUNTRY : Italy | ORGANISATION : C.N.E.N. |
| DATE INITIATED : 1.1.1975<br>DATE COMPLETED : ........ (in progress) | PROJECT LEADER<br>N. Pacilio, LFCR |

9-4

| Title 1 (Original language) | Classification |
|---|---|
| Diagnostica e Sicurezza tramite Analisi di Rumore (Parte B) | 8 - 14 |

6. Relation to other projects

Terms of cooperation are going to be defined with Caorso (AMN-ENEL), Halden Project (Norway), CEA (France).

7. Degree of availability

Documents are not classified material, they can be requested from one of the authors by the following address: RIT, CSN-Casaccia, C.P. 2400, Rome, Italy.

| Title 1 (Original language) | Classification |
|---|---|
| Dynamic studies for safety analysis | 4 - 8 |
| Title 2 (English) | Country ITALY<br>Sponsor / Organisation CNEN |
| Date initiated 1962<br>Date completed in progress<br>Last updating April 1977 | Project Leader<br>M. Di Bartolomeo |

| Title 1 (Original language) | Classification |
|---|---|
| Statistical analysis of randome signals | $\boxed{\text{I}}$ - 3 - 4 - 8<br>IO - I4 |
| **Title 2 (English)** | Country ITALY<br>Sponsor / Organisation CNEN |
| Date initiated I966<br>Date completed in progress<br>Last updating April I977 | Project Leader<br>A. Federico |

| Energieonderzoek Centrum Nederland | CLASSIFICATION : 8.10.4 |
|---|---|
| **TITLE :** Ruisanalyse in vermogensreactoren met het oog op reactorbewakingsmogelijkheden. | COUNTRY: THE NETHERLANDS |
| | SPONSOR : ECN |
| | ORGANIZATION : ECN |
| **TITLE (ENGLISH LANGUAGE):** Noise analysis in power reactors for malfunction detection | PROJECTLEADER : E. Türkcan |
| INITIATED : 1974 / LAST UPDATING : June 1978 | SCIENTISTS : - |
| STATUS : in progress / COMPLETED : 1981 | |

General aim

Noise measurements and analysis in power reactors, mostly in PWR.

Particular objectives

1. Establish noise signatures of the Borssele PWR (in particular ex-core and in-core neutronic noise and primary circuit pressure noise).
2. Interpret noise spectra in terms of e.g. reactivity effects and core support barrel vibrations, using cross-correlation techniques.
3. Automatic separation of the noise into components due to different physical origins.
4. Automatic surveillance of the spectra of the different noise components.

Experimental facilities

- Borssele reactor: 18 ex-core and 6 in-core neutron detectors, and 10 primary circuit pressure transducers with suitable electronics.
- Extensive computer based measuring and analyzing equipment for multi-detector noise analysis.

Project status

Objective 1&2: Several experiments per core since December 1974. In-core results since December 1977.

Objective 3 : Off-line separation in operation for noise above 5 Hz.

Next steps

Objective 1: To be continued for present and next cores.

Objective 2: Continuous improvement of understanding, especially of pressure noise.

Objective 3: Extension to lower frequencies and for effects hither to not considered; on-line analysis.

Objective 4: Pattern recognition type of techniques are envisaged for real time application.

Relation to other projects: -

Reference documents

|1| Türkcan, E., Measurements and Analysis of Ex-core Neutron Detector Noise of Borssele Reactor (PWR) at Full Power, Reaktortagung, Düsseldorf (1976), p. 577.

|2| Türkcan, E. and J.B. Dragt. Noise Applications in Pressurized Water Reactors; Implementation at the Borssele Reactor. Enlarged Programme Group Meeting on Process Supervision and Control in Nuclear Plants, Frederikstad, 6-9 June, 1977.

|3| Dragt, J.B. and E. Türkcan. Borssele PWR Noise: Analysis and Interpretation. Progress in Nuclear Energy, Vol. 1, nr. 2-4 (1977), p. 293-307.

|4| Türkcan, E. and J.B. Dragt. New Results of the Noise Applications in the Borssele Pressurized Water Reactor. Enlarged Programme Group Meeting on Water Reactor Fuel Performance and Applications of Process Computers in Reactor Operation, Loen, 5-9 June, 1978.

Degree of availability

Reports are available through ECN, Postbus 1, 1755 ZG Petten, The Netherlands.

Budget: -

Personnel: -

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. 77 - 31. 12. 77 | 9 | RS 108 |

| Vorhaben/Project Title | |
|---|---|
| Berstsicherheit für Primärkreislauf<br><br>Fracture Safety for the Primary Circuit | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 2, Erlangen |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1. 8. 73 | 30. 6. 77 | Dr. Dorner |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Completed | 31. 12. 77 | 4'586.000,-- DM |

1. General Aim and 2. Particular Objectives

A concept of a fracture safety device for the primary circuit had to be developed, which offered chances for a better protection after vessel failure. This was relieved by components with simple geometry and small surfaces. Under this project investigations were carried out to change the present design in order to protect the containment after a pressure vessel rupture without reducing access from the outside for inservice inspection and without reducing reliability.

3. Research Program

The investigation program encompassed the following work:

1. Design of a fracture safety device system for the pressure vessel
2. Design of a fracture safety device system for steam generator
3. Design of a fracture safety device for the pressurizer
4. Design of a fracture safety device for the main coolant pumps
5. Design of a fracture safety device for the primary circuit (piping)

4. Experimental Facilities

No experimental facilities were necessary.

5. Progress to Date

The work has been completed.

6. Results and 7. Next Steps

The results are evaluated. The final report has been written.

8. Relation with Other Projects

RS 104 Investigation Program for the Testing of a Fracture Safety Device Protection System for Reactor Components

9. References

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10. Degree of Availability

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## 9. OTHER SAFEGUARDS

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. 77 - 31. 12. 77 | 9 | RS 234 |
| Vorhaben/Project Title<br>Auswertung des Vorversuchs V und Abschluß des Förderungsvorhabens RS 104<br><br>Evaluation of the Preliminary Investigation V and Completion of R+D-Task RS 104 | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>KRAFTWERK UNION AG<br>Reaktortechnik<br>R 2, Erlangen |
| Arbeitsbeginn/Initiated<br>1. 10. 77 | Arbeitsende/Completed<br>30. 4. 78 | Leiter des Vorhabens/Project Leader<br>Dr. Dorner |
| Stand der Arbeiten/Status<br>Continuing | Berichtsdatum/Last Updating<br>31. 12. 77 | Bewilligte Mittel/Funds<br>173.436,-- DM |

1. General Aim

Presentation of the state of knowledge based upon recent experiences. This work will complete task RS 104.

2. Particular Objectives

Evaluation of the already performed preliminary test V with respect to:

- verification of results obtained from current preliminary investigations especially from pre-test III
- investigation of burst protection as affected by burst phenomena
- investigation of extreme thermohydraulic processes within and outside of the test pipe during pipe failure.
- comparison of the test results with currently performed pre-tests
- presentation of the recent experiences gained by this program, documentation of present status of knowledge
- completion of task.

3. Research Program

3.1 Burst Process, Pressure-Relief- and Blowdown Process

a) verification of preliminary test III results with respect to failure modes, fracture mechanical and thermohydraulic processes in the pipe section. Detailed information on crack propagation, deformation and thermohydraulics within and outside of the pipe will be required for the development of model concepts.

b) checking of current results with respect to burst protection loads, which will facilitate finding of a sizing base for burst protection design.

c) detailed information on burst protection as affected by burst processes.

3.2 Preliminary Investigation Carried Out

To verify current results and in view of the planned continuation of the program, a pre-test V was performed in Dec. 76.

In this test, a pipe NW 350 was installed in a burst protection test section (repetition of pre-test III). At the predetermined breaking point of the test pipe an axial longitudinal crack appeared as it was the case with pre-test III.

Purpose of the test was:

- to check the predetermined breaking point
- to verify extreme thermohydraulic processes (pressure gradients, thermodynamic unbalance) within and outside of the test pipe and possible to obtain more details on these processes
- to determine more exactly, if possible, the position of the critical cross-section in the blowdown process
- to investigate burst equipment sizing.

4. Test Facilities

Neither test equipment nor computer programs will be required since evaluation refers to an already performed test.

5. Progress to Date

Investigation and checking of pressure transducers
delivery of pipe
write-up of all measurement data
first evaluation of measurement data.

6. Results

No special results exist up to now.

7. Next Steps

Dimensioning of test pipe
Evaluation of measurement results
Write-up of test- and final report.

8. Relation with Other Programs

RS 104, RS 108

9. References

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10. Degree of Availability

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|---|---|
| N.V. KEMA | CLASSIFICATION: 9.1.9.2.9.3 |
| TITLE: Rekenmodel voor de simulatie van transiënten in kokendwaterreactoren | COUNTRY: THE NETHERLANDS |
| | SPONSOR: KEMA<br>ORGANIZATION: KEMA |
| TITLE (ENGLISH LANGUAGE): Computercode for the simulation of transients of boiling water reactors | PROJECTLEADER: R.M. van Kuijk |
| INITIATED: – LAST UPDATING: 1978 | SCIENTISTS: P. Kloeg |
| STATUS: – COMPLETED: 1977 | |

General aim

Calculations of the dynamic response of BWR's in the case of transients and small accidents.

Particular objectives

Study of the control systems, capacity of safety valves, influence of setpoints, heat transfer in the core, fuel temperatures, scram action.

Experimental facilities and programme

Computer code REBOR.

Project status

Operational for Dodewaard BWR.

Next steps

Not applicable.

Relation with other projects

Not applicable.

Reference documents

Internal KEMA reports.

Degree of availability

Free on basis of exchange with other programmes.

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX

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| N.V. KEMA | CLASSIFICATION: 9.1.9.2.9.3 |
|---|---|
| **TITLE:** Rekenmodel voor de simulatie van transiënten in kokendwaterreactoren | COUNTRY: THE NETHERLANDS |
| | SPONSOR: KEMA |
| | ORGANIZATION: KEMA |
| **TITLE (ENGLISH LANGUAGE):** Computercode for the simulation of transients of boiling water reactors | PROJECTLEADER: R.M. van Kuijk |
| INITIATED: - / LAST UPDATING: 1978 | SCIENTISTS: P. Kloeg |
| STATUS: - / COMPLETED: 1977 | |

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| N.V. KEMA | CLASSIFICATION: 9.1.9.2.9.3 |
|---|---|
| **TITLE:** Rekenmodel voor de simulatie van transiënten in kokendwaterreactoren | **COUNTRY:** THE NETHERLANDS |
| | **SPONSOR:** KEMA |
| | **ORGANIZATION:** KEMA |
| **TITLE (ENGLISH LANGUAGE):** Computercode for the simulation of transients of boiling water reactors | **PROJECTLEADER:** R.M. van Kuijk |
| **INITIATED:** – / **LAST UPDATING:** 1978 | **SCIENTISTS:** P. Kloeg |
| **STATUS:** – / **COMPLETED:** 1977 | |

# 10. CORE AND PRIMARY CIRCUIT IN STEADY STATE CONDITIONS

142-1 -03/4113-01
152-1 -03

3.1

* 10

| | |
|---|---|
| **Titre**<br><br>Tenue de structures - types sous excitation sismique. Essais sur table vibrante. | **Pays**<br>FRANCE<br><br>**Organisme directeur**<br>CEA /DgCS |
| **Titre (anglais)**<br><br>Behaviour of typical structures under seismic excitation.<br>Shake table tests. | **Organisme exécuteur**<br>CEA/DEMT - Saclay<br><br>**Responsable**<br>(DEMT-Sacla |

| | | |
|---|---|---|
| Date de démarrage 1/75 | Etat actuel en cours | Scientifiques |
| Date prévue d'achèvement 12/78 | Dernière mise à jour 12/77 | |

| PROJECT TITLE : Statistical analysis of random signals for safety problems | CLASSIFICATION 10 - 14 - 8 |
|---|---|
| SPONSORING COUNTRY : Italy | ORGANISATION : CNEN |
| DATE INITIATED : 1966<br>DATE COMPLETED : in progress | PROJECT LEADER : A. Federico |

<u>Description</u> : Concern the development of methods for acquisition and elaboration of data coming from nuclear reactors and experimental loops.
These methods give the possibilities to study reactor physics, thermohydraulic and mechanical effects in normal and accidental conditions. (Possible application: <u>early failure detection</u>).

<u>Facilities</u>

- Transducers, amplifiers, filters, magnetic recorders for data acquisition.
- 1 hybrid computer EAI 8945
  1 hybrid computer EAI PACER 700
- Software for statistical elaboration running on the hybrid computer mentioned.

<u>Reference documents</u>

1) L.Cimorelli - A.Federico
Applications of spectra analysis tecniques to examine natural and superimposed neutronic flux fluctuations in a nuclear power reactor
Rapp, CNEN - IN(69)3 - Marzo 1969

2) A.Federico - S.Taglienti
Frequency and time-domain systems for statistical signals elaboration developed in CNEN laboratories-IAEA specialist meeting on Analysis of Measurements to Diagnose Potential Failures.
Roma, Aprile 10-11, 1972

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

975

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.1977 - 28.2.1977 | 10.1 | RS 68 A |

| Vorhaben/Project Title | |
|---|---|
| Application of Statistical Analysis Methods in Power Reactors under the Safety Oriented Aspects of Early Fault Detection<br><br>Anwendung statistischer Analysenverfahren in Leistungsreaktoren mit dem sicherheitstechnischen Ziel der Früherkennung von Schäden. | Land/Country<br>FRG<br><br>Fördernde Institution/Sponsor<br>BMFT<br><br>Auftragnehmer/Contractor<br>Technische Universität Hannover<br><br>Institut fuer Kerntechnik |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.3.1975 | 28.2.1977 | Prof.Dr.-Ing.Stegemann |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Completed | 20.2.1978 | 177.000,-- DM |

## . General Aim

Theoretical and experimental investigations of the dynamical behaviour of the neutron flux in the core of a power reactor due to thermohydraulic and mechanical feedback.

Analysis of fluctuating in-core detector signals in order to establish the characteristic data of normal operating conditions and to get an indication of any anomalies in the case of certain significant deviations from these characteristics.

## 2. Particular Objectives

Noise measurements in both Boiling- and Pressurized Water Reactors during full power operation in order to gain extensive data about space-dependent reactivity effects caused by pressure oscillations, ibrations or coolant density feedback.

Development of a new and reliable in-core measuring system, using "self-powered" neutron detectors.

Application of these continuously improved devices for neutron flux distribution measurements with high spatial resolution as well as for reactor power noise investigations without disturbance of the plant operation.

## 3. Research Program

3.1 Completing the computer analysis of the tape-recorded detector signals.

3.2 Evaluation of the resulting data with respect to their information content about thermohydraulic parameters in the reactor core.

3.3 Comparison of experimental and theoretical results.
3.4 Collecting and representing of all the data in the final research report.

## 4. Experimental Facilities, Computer Codes

–

## 5. Progress to Date

To 3.1 The signals of the in-core self-powered detectors, which had been measured and tape-recorded in an extensive experimental campaign at the Lingen power plant in December 1976, were processed in a detailed frequency and correlation analysis, using a special computer program system which was developed for this purpose at the institute.

To 3.2 Evaluation of the analysis was performed under the following special aspects:
determination of the axial velocity profiles of steam bubbles in the core with high spatial resolution,
investigation of the space-dependent auto power spectral densities (APSD) of the reactor noise,
influence to neutron flux signals due to vibrating structures,
determination of the space-dependent noise amplitudes, e.g. the normalized root mean square values (NRMS) of the noise.

To 3.3 From theoretical calculations the significant thermohydraulic parameters like steam bubble velocity, slip ratio, boiling boundary and void content and their axial shapes could be determined. Comparison of these data with the experimental results supplied further information about local operating conditions like bundle power, power density or mass flow through the fuel element.

To 3.4 The results were represented in the final research report.

## 6. Results

The computer analysis and the corresponding evaluations were completed. With respect to thermohydraulic parameters in a boiling water reactor, e.g. steam bubble velocities, steam content, boiling boundary, the measurements supplied results of excellent accuracy and spatial resolution. In connection with the theoretical investigations a detailed description of the thermohydraulic behaviour of the reactor core can be given.

## 7. Next steps

The research program was finished on Febr. 28, 1977 and completed by the presentation of the final report.

## 8. Relation with other Projects

The experiences and results of these investigations will be the basis of the research program RS 232.

## 9. References

P.Gebureck, W.Jaschik, D.Stegemann: Final Report RS 68/68 A

## 10. Degree of Availability of the Reports

Institut für Kerntechnik, T.U. Hannover, Elbestr. 38A,
3000 Hannover 21.

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.1977 - 31.12.1977 | 10.1 | RS 232 |
| Vorhaben/Project Title<br>In-Core Data Acquisition and Processing with the Aim of Early Fault Detection in Light Water Reactors<br><br>In-Kern-Meßdatenerfassung und -verarbeitung mit dem sicherheitstechnischen Ziel der Früherkennung von Schäden in Leichtwasserreaktoren | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>T.U. Hannover<br><br>Institut für Kerntechnik |
| Arbeitsbeginn/Initiated<br>1.12.1976 | Arbeitsende/Completed<br>31.12.1979 | Leiter des Vorhabens/Project Leader<br>Prof.Dr.-Ing.D.Stegemann |
| Stand der Arbeiten/Status<br>Continuing | Berichtsdatum/Last Updating<br>Dec. 1977 | Bewilligte Mittel/Funds<br>570.200,-- |

## 1. General aim

Development of a complex system for acquisition, processing and presentation of measured in-core data with the safety oriented aim of early fault detection in light water reactors.

## 2. Particular Objectives

Basing upon the experience in application of statistical analysis methods in power reactors (RS 68A), an on-line surveillance system will be developed, allowing the early detection of certain anomalies in reactor operation and supporting the operators to avoid dangerous operating conditions. The additional informations consist of a number of nuclear and thermohydraulic data. The system design includes any future tasks of combined analysis of neutron noise and vibration measurements.

## 3. Research Program

3.1. Construction of a highly flexible electronic measuring system providing acquisition and preparation of selected safety relevant process signals in a power reactor.

3.2. On-line processing of the data in time and frequency domain using Fast Fourier analysis. Computation of characteristic time and space dependent functions, e.g. noise amplitudes, spectra, phase relations, correlation functions.

3.3. Presentation of all characteristic values and functions on color video display and plotter on request.

3.4. Evaluation of the reduced data with respect to their information content about significant thermohydraulic and mechanical parameters (steam content, steam bubble velocities in BWR, excessive vibrations of structures etc.)

4. Experimental Facilities, Computer Codes

5./6. Progress to Date and Results

(3.1) Development of a differential, self-compensating low-noise amplifier suitable for the usual in-core fision chamber instrumentation as well as for self-powered neutron detectors. The amplifier design involves separate treatment of the stationary and the fluctuating signal components.
Tests under full power conditions at the power plant Brunsbüttel and further improvements have been completed.
Production of a total number of 32 amplifiers, the signal distributor and multiplexer is proceeding as planned. Other required hardware is available.

(3.2) Already existing computer programs have been checked for applicability.
Development of micro-programs for fast Fourier transforms of time series and processing of the transformed data blocks completed.
Adaption of the micro-program for Fourier analysis with respect to the real-time conditions.

(3.3) Driver programs for the color video display unit and the analog magnetic tape recorder are available now.
The developed programs provide presentation of auto and cross power spectral densities, phase and coherence relations, auto and cross correlation functions, normalized noise amplitudes and local steam bubble velocity values on color video display.

(3.4) Signals of the in-core instrumentation at the BWR Brunsbüttel have been measured and recorded on magnetic tape during full power operation.
After extension of the system to on-line analysis and evaluation of the signals of one instrumentation tube with four detectors, simulation of the on-line operation was performed by

means of the recorded in-core signals. The results were compared with those ones achieved from an off-line investigation using a different program system for signal analysis.

## 7. Next steps

Development of a driver program for the multiplexer and tests.
Program achievement for arranging and storing the data of interest on magnetic disc.
Extension of the simultaneous on-line analysis from 4 channels to date up to the desired number of 32.
Theoretical investigations using a program system for calculations of nuclear and thermohydraulic core data. One-dimensional computation of neutron spectra (50 groups) as function of spatial coordinate in the subsections of a fuel element.

## 8. Relation with other Projects

## 9. References

M.Zeller: Program description and user directory for the RTE-II-Drivers:

DVR77 (colored video display system)
DVR76 (colored video display console)
DVR75 (analog magnetic tape recorder)

IKH-Report 77/77.

M.Zeller: Processing of noise signals at the HP2100 computer using micro-programs. Survey and program descriptions. IKH-Report 90/77.

## 10. Degree of Availability of the Reports

Institut für Kerntechnik, Technische Universität Hannover, Elbestraße 38A, 3000 Hannover 21.

Classification: 10.2

| Title: Statisk Reaktorfysik | Country: DENMARK |
|---|---|
| | Sponsor: Risø National Laboratory |
| Title: Development of calculation methods for static and quasistatic reactor physics in light water reactors | Organization: Risø National Laboratory |
| Initiated date: 1968 Completed date: <br> Status: progressing | Scientists: C.F. Højerup, G.K. Kristiansen, P. Lauridsen, L. Mortensen, H. Neltrup, T. Petersen, B. Schougaard |

1. General aim

Provision of static and quasistatic reactor physics information relevant for safety assessment of light water reactors.

2. Particular objective

Reliable methods for calculating -
Power shape as well within fuel boxes as overall for the reactor as function of burn up.
Reactivity as function of burnup.
Reactivity control and reactivity budgets.
Reactivity coefficients.
Stability.

3. Experimental facilities and programmes

4. Project status

1. Progress to date

a) Multi-group cross section generation from UKNDL completed
b) Multi-group G $\leq$ 76, collision probability methods for fuel rods completed and coupled to multi-group diffusion theory calculations completed.

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c) Few group 2D and 3D diffusion theory calculation codes, using finite difference techniques, finite element, flux synthesis and nodal theory for the reactor core completed.

d) 3D flux synthesis or 3D nodal theory coupled to hydraulic channel calculation for BWR including burn up completed.

## 2. Essential Results

Reactor physics code system verified in calculation on Yankee Rowe, Connecticut Yankee and Dresden.

## 5. Next Steps

1. Test the system further against BWR measurements.
2. Further refinement of methods concerning fuel element calculation and development of fast 3D methods.

## 6. Relation with other projects

Provision of cross-sections and parameters for dynamics projects and fuel management studies.

## 7. Reference Documents


1. A.M. Hvidtfeldt Larsen, H. Larsen and T. Petersen. Calculation on a Boiling Water Reactor as a Test of the Risø Reactor Code Complex. Risø Report No. 268, 1972.
2. Torben Petersen. Aspects of Prediction of the Performance of a Boiling Water Reactor. Risø Report No. 289, 1973.
3. H. Neltrup and Per B. Suhr. Survey Calculation on the Hadam Neck (Connecticut Yankee) Power Plant as a Test of the Risø Reactor Physics Code System. Risø Report No. 298, 1973.
4. C.F. Højerup, The Cluster Burn up Programme CCC and Comparison of its Results with NPD Experiments, Risø-M-1898 (1976) 6 pp.

8. Degree of availability

Partly available, partly available on exchange basis.

| Berichtszeitraum/Period<br>1.01.-31.12.1977 | Klassifikation/Classification<br>10,3 | Kennzeichen/Project Number<br>RS 81 |
|---|---|---|
| Vorhaben/Project Title<br>Mischungseffekte bei parallel durchströmten Kanälen<br><br>Mixing effects in parallel channels with water two-phase flow | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Euratom CCR<br>Ispra / Italy |
| Arbeitsbeginn/Initiated<br>01.01.1975 | Arbeitsende/Completed<br>31.12.1977 | Leiter des Vorhabens/Project Leader<br>Dr.Herkenrath/Dr.Hufschmidt |
| Stand der Arbeiten/Status<br>Continuing | Berichtsdatum/Last Updating<br>December 1977 | Bewilligte Mittel/Funds<br>DM 1.303.590,- |

GENERAL AIM

The investigation of the interaction and mixing effects between adjacent subchannels of a fuel rod bundle cooled by boiling water is aimed to providing experimental data on the local distribution of mass-flow, enthalpy and voidfraction so as to achieve a better thermohydraulic description and more precise DNB margin for a LWR core. Furthermore, studies in steady-state conditions are a necessary preliminary to studies of the mixing process in transient conditions such as those occurring during power transients or blowdown. Therefore, the first phase of the programme will be devoted to investigations under steady state conditions.

The actual programme provides the studies of steady-state measurents of the subchannel interactions in two-phase flow conditions with 16 rod cluster test-sections in BWR-geometry (pressure 70 bars), and in PWR-geometry (pressure 160 bars) with appropriate splitting devices at the outlet and the studies of transient mixing measurements for partial depressurization (to about 80% of the initial pressure) with the above mentioned testsections. For not damaging the loop components- especially the pumps - a complete blowdown of the loops is not possible.

For a more fundamental understanding of the mixing process investigations with two-channel test-sections of different shapes (firstly an eight-shaped channel with different heat-inputs) are foreseen also in both, steady-state and transient conditions.

## 2. PARTICULAR OBJECTIVES

The mixing programme was started in the frame of collaboration contracts between CNEN (Rome) and BMFT (Bonn) on one hand and Euratom on the other. The contracts involved the studies of the boiling mixing phenomena using two different test sections:

- A two-channel test section for fundamental studies (pressure up to 150 bars; high pressure water loop PRIL) and
- a 16-rod cluster test section with BWR-geometry (pressure 70 bars; high pressure water loop BOWAL).

Later, the BOWAL studies should be extended to a pressure of 160 bars using PWR geometries.

After first operational tests with the two-channel test section in the PRIL loop at the beginning of 1974, this part of the programme was interrupted and effort was concentrated on the preparation and execution of the 16-rod bundle tests in BOWAL.

## 3. RESEARCH PROGRAM

Experimental investigation of subchannel interactions in two-phase flow conditions (mixing):

- Steady state measurements of the mixing effect with a 16-rod cluster in BWR geometry (70 bars)
- Mixing studies in transient conditions with 16-rod clusters in BWR- (70 bars) and PWR geometries (160 bars).

## 4. EXPERIMENTAL FACILITIES

The high-pressure water loop BOWAL (3.6 MW power input) to be used for the boiling mixing experiments with the 16-rod cluster test-sections in BWR- and PWR-geometry and the high-pressure water loop PRIL (0.6 MW power input) to be used for the fundamental mixing studies and for calibration tests with void-meters have been described in detail in / 1 / and the BOWAL loop is schematically shown in Fig. 1. The main data of the two loops are the following:

### PRIL LOOP

This loop consists of:

- controllable forced circulation
- partial blowdown ( 20%) facility for the transient tests

- pressurizer, condenser, subcooler.

The characteristics of PRIL are:

- Maximum power: 600 kW
- Maximum pressure: 200 bar
- Maximum temperature: 365°C
- Special instrumentation: different void meters.

BOWAL LOOP

This loop consists of:

- controllable forced circulation
- partial blowdown (20%) facility for the transient tests
- pressurizer, condenser, subcooler and preheater (500 kW).

The characteristics of BOWAL are:

- Maximum power: 3,6 MW
- Maximum pressure: 250 bar
- Maximum temperature: 365°C
- Special instrumentation: distribution of temperature and power along a number of the rod, pressure distribution along the test section. Outlet enthalpy distribution among the 6 basic subchannels of the cluster.

The 16-rod cluster test section PELCO-S for the BWR investigations (70 bars) is described in / 2 /. The PWR test section EUROP (16rods, 160 bars) is actually in the final state of construction.

The X-ray voidmeter devices to be used for the transient measurements has been developed and will be described in a special report after testing under real operation conditions.

5. PROGRESS TO DATE

6. RESULTS

The first results of the mixing experiments with the 16-rod cluster test section have shown that the measuring device is not sufficient for an exact interpretation of the mixing process. The temperature, pressure and mass flow must be measured for the five characteristic subchannels of the rod bundle. The missing two measuring devices and condensers have been taken from the PRIL loop to avoid unacceptable time delays. The new concept of the BOWAL loop is shown in Figure 1.

The sampled subchannels of the 16-rod cluster sections (PELCO-S, 70 bars and EUROP, 160 bars) are seen in Fig. 2. Each sampled subchannel (index s) has its reference channel (index r) with appropriate pressure tapes at the beginning of the splitting device (Fig. 3) for the isokinetic measurement technique.
The effectiveness of the modification of BOWAL loop (simultaneous analysis of five subchannels instead of the originally installed combination of three channels) can be shown in Fig. 4. The outlet quality of the sampled subchannels ($x_{UK}$) is plotted against the main outlet quality $x_o$ in the total cross section of the test section. The quality $x_{UK}$ is calculated by means of the outlet enthalpy measured with the calorimetric condensers / 1 / and the quality $x_o$ results from the heat balance with the total electrical power input and the total mass flow.

In the upper part of Fig. 4 the results of the measurements with the three subchannel arrangement indicate that the $x_{UK}$-values are below the 45°-line. This means that the total heat input calculated by means of these subchannel values is much smaller than the electric power to the test section.

The $x_{UK}$-values gained from the five subchannel arrangement (lower part of Fig. 4) show that the outlet qualities in the subchannels number 3, 4 and 5 (Fig. 2) are slitely higher than the 45°-line, whilest the qualities in channels 1 and 2 are significantly smaller. This is valied especially for the corner subchannel (no. 2). With the new arrangement the heat balance is in the order of about 8%. The measured subchannel enthalpies are multiplied by the respective number of equal subchannels and then summed up. A much better heat balance is not probable because of the tolerances in the fabrication of the test section (especially in the upper part with the splitting and sampling device). The predicted measurement errors / 1 / for the isokinetic technique based on an ideal configuration yielded better heat balances.

X-ray voidmeters

The X-ray voidmeters composed of three high voltage generators KRISTALLOFLEX 800 and six tubes AGW 61 T (60 kV, 3000 W) have been delivered and first tests are made. Different modifications, especially with respect to the detector device (ionisation chambers) have been proved and the final conception is in fabrication.

PWR 16-rod cluster EUROP

After special pretests for the sealing of the heater rods and after the delivery of the spacers, original KWU-type, the PWR 16-rod cluster test section EUROP is nearly ready for assembling at CISE, Milan. The delivery is foreseen for February 1978.

## 7. NEXT STEPS

- Continuation of the mixing experiments with test section PELCO-S (BWR-geometry, 70 bars) under steady state conditions in the following range:

| | |
|---|---|
| pressure: | 70 bars |
| mass flow density: | 1000 ÷ 2000 kg/m²s |
| inlet quality: | -3 ÷ -27% |
| heat flux: | up to 2.6 MW |
| outlet quality: | -3 ÷ + 45% |

- Comparison of the experimental results with existing mixing codes in collaboration with NUCLITAL, Genova.
- Installation of a supplementary subcooler (4 MW), necessary for the tests with low outlet qualities.
- Calibration of the X-ray voidmeters under real operation conditions (up to 160 bars) in PRIL loop.

Because of the execution of the foreseen experimental programme cannot be achieved in the planned time a prolongation of the research contract for at least two years has been asked.

## 8. RELATION WITH OTHER PROJECTS

- RS 109 Experimental investigation of the influence of PWR loops on blowdown

- RS 163 Theoretical and experimental investigations on thermo- and fluiddynamic behaviour of the reactor core during the first blowdown phase.

## 9. REFERENCES

/ 1 / Herkenrath, H., Hufschmidt, W.
"The pressurized and boiling water loops BOWAL and PRIL for boiling mixing studies of the Heat Transfer and Fluid Mechanics Division of the JRC Ispra"
EUR report (in press)

/ 2 / Gaspari, G.P., Germani, G.F., Lucchini, F., Marelli, A.
"PELCO-S: A BWR 16-rod test section for subchannel experimental analysis" CISE-Doc. Service, PELCO No. 4, Max 1975

## 10. DEGREE OF AVAILABILITY OF THE REPORTS

Reports mentioned in 9, except ref. 2, are not classified and are on sale at the office for Official Publications of the European Communities, 37, rue Glesener, Luxembourg.

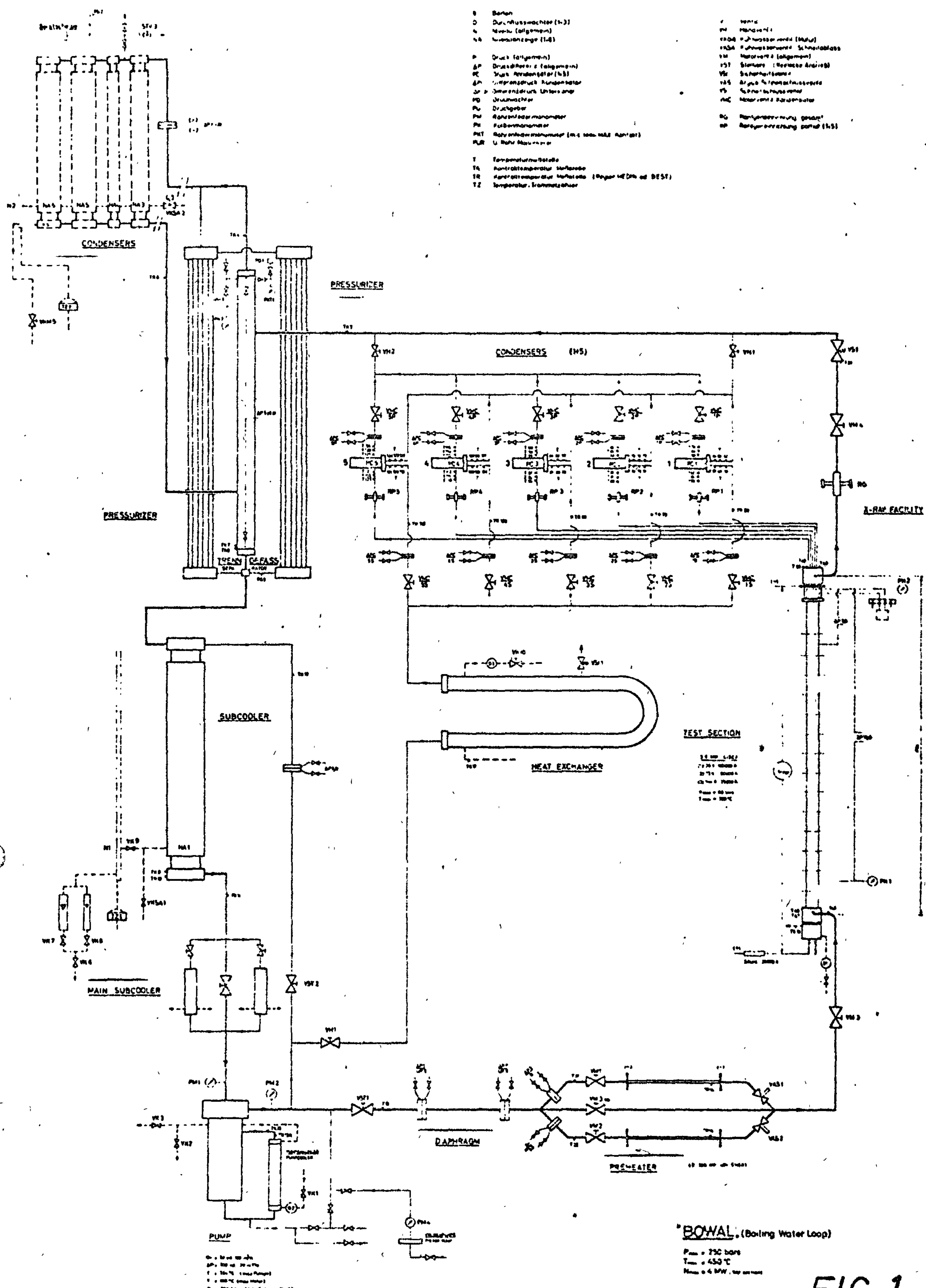
CONDENSERS
PRESSURIZER
PRESSURIZER
CONDENSERS (HS)
X-RAY FACILITY
TEST SECTION
HEAT EXCHANGER
SUBCOOLER
MAIN SUBCOOLER
PUMP
DAPHRAGM
PREHEATER
BOWAL (Boiling Water Loop)

FIG. 1

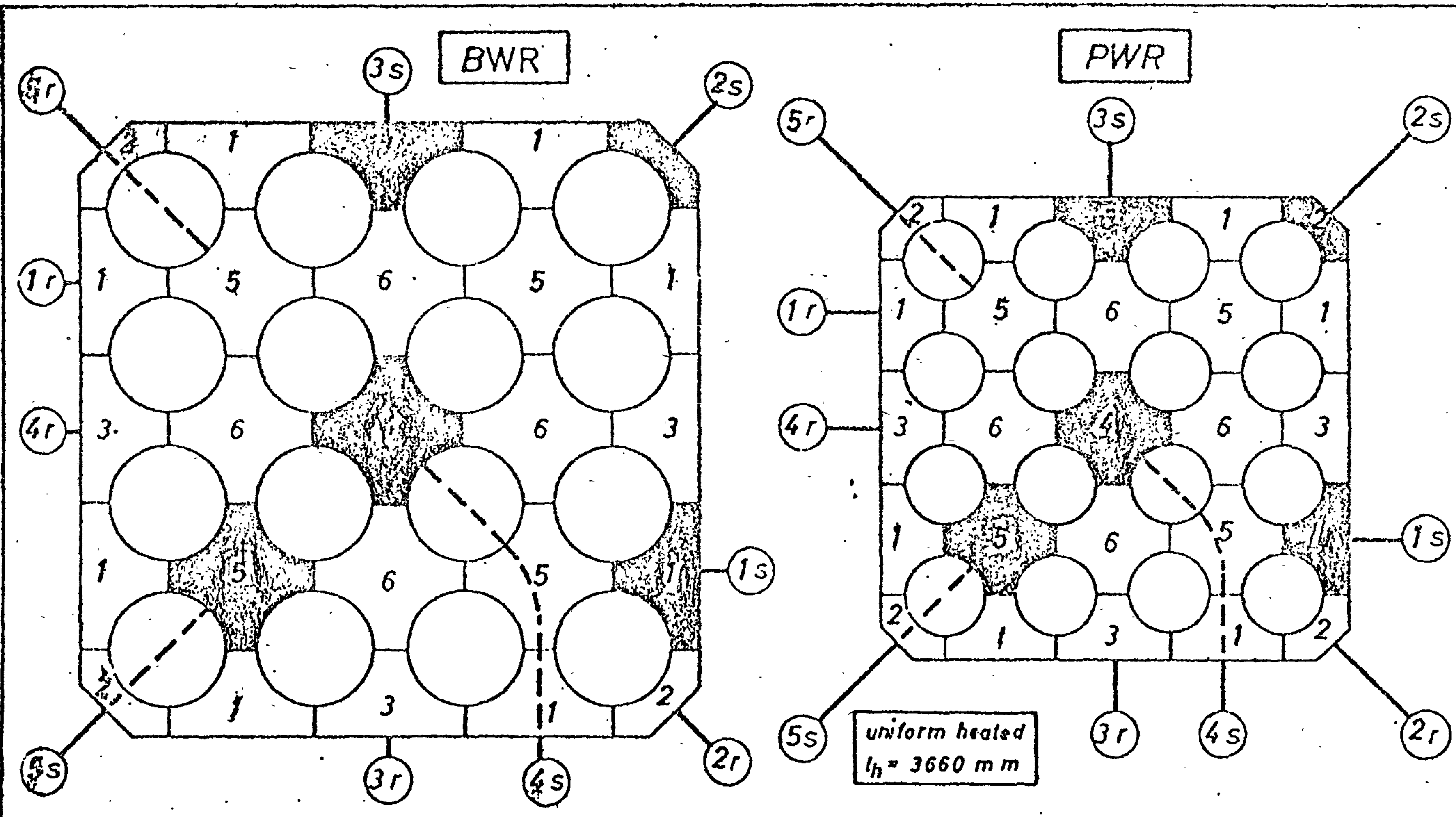


FIG. 2 Cross-section of the test-section outlet with splitters between the sub-channels and appropiate taps for static pressures (isokinetic measurement)
s = sampled subchannel r = reference subchannel

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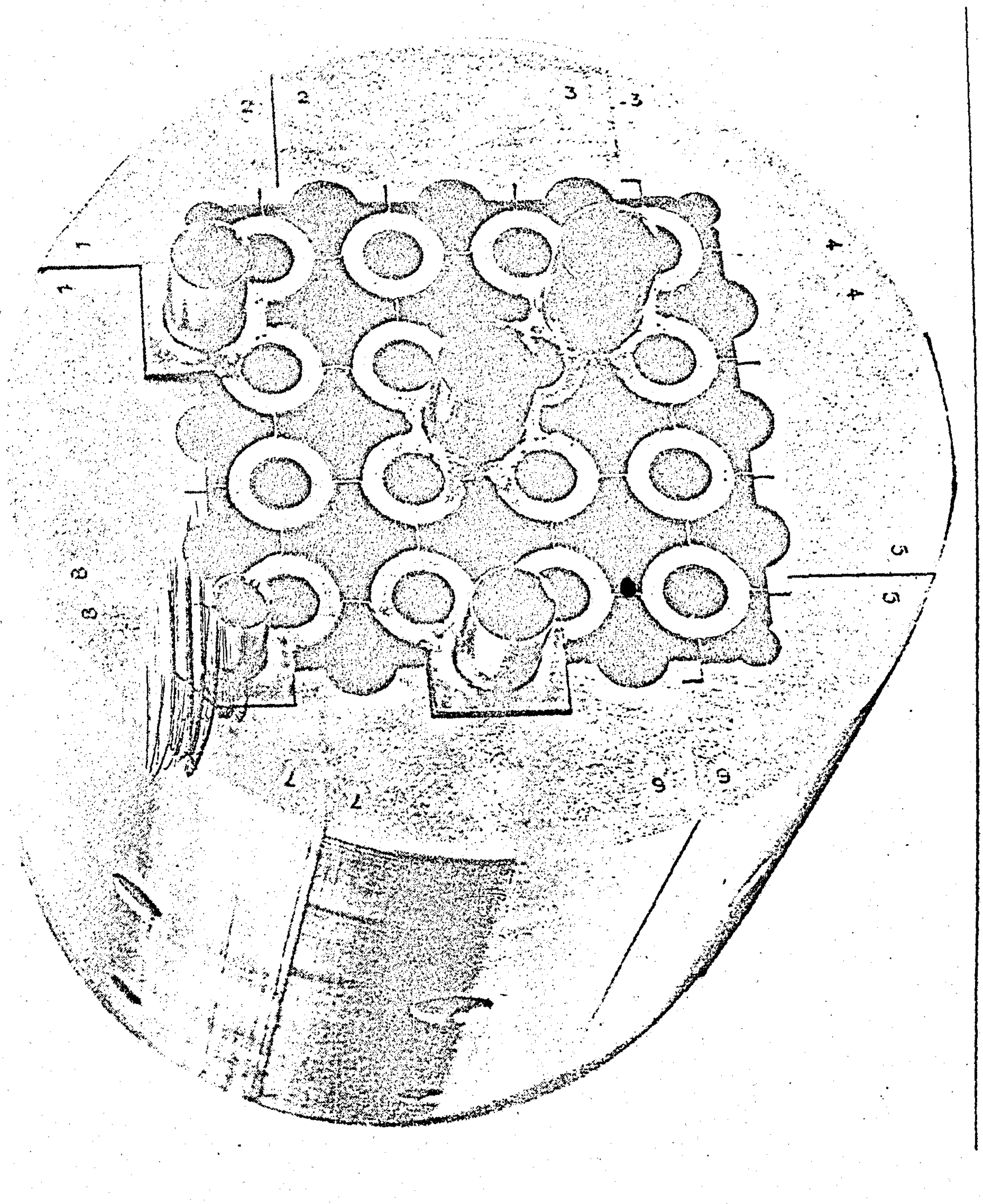

FIG. 3 : Upper part of 16-rod cluster test-section PELCO S with splitters and sampling pipes. (after ref. [1] )

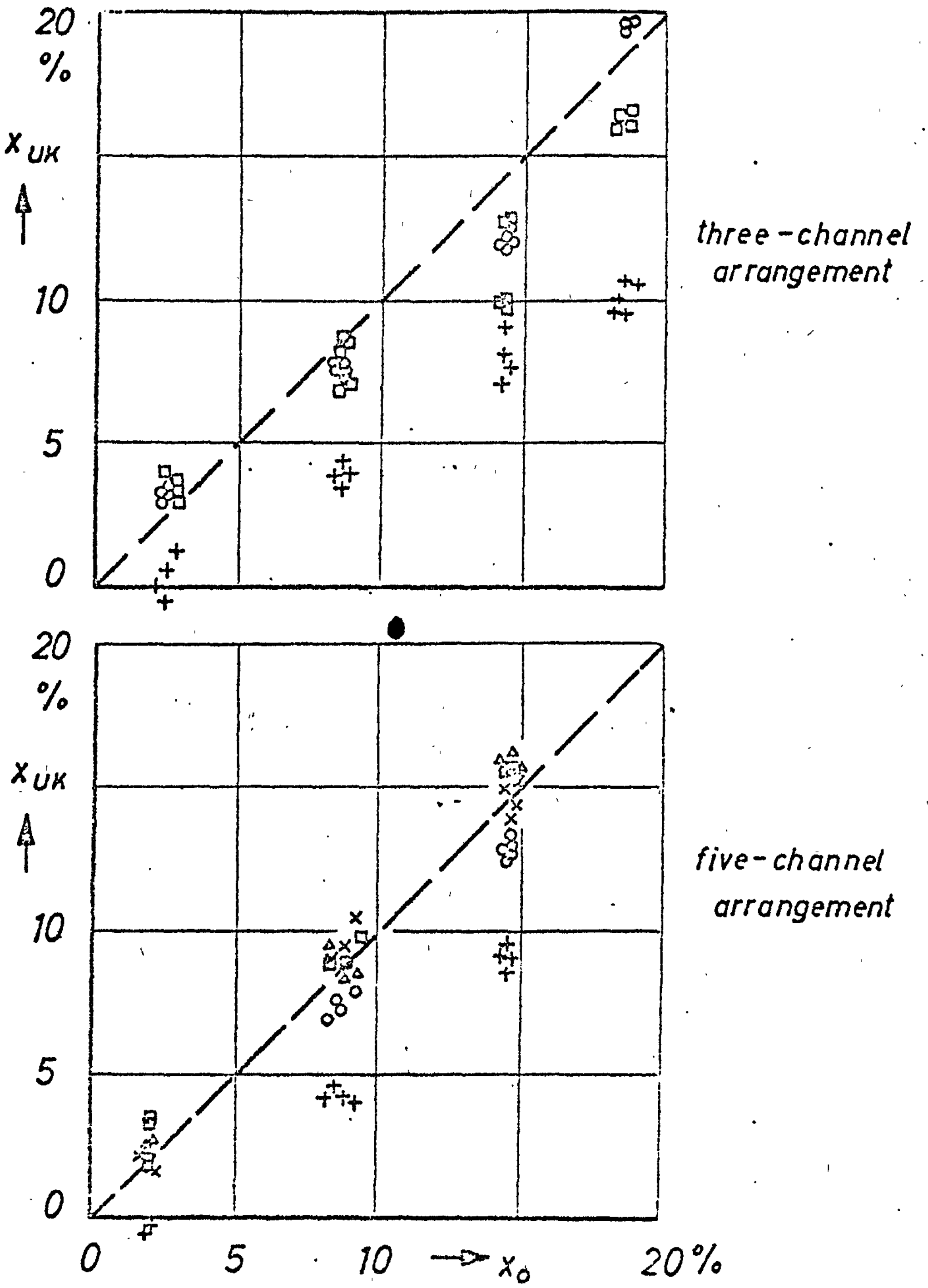


Fig. 4 : Measured outlet qualities from sampled sub-channels $x_{UK}$ vs. mean outlet quality $x_o$ for PELCO S at 70 bars and a mass flow density of 1000 kg/m²s (channel ns. after fig. 2 : 1 ○ ; 2 + ; 3 △ ; 4 × ; 5 □ )

Classification: 10.3

| | |
|---|---|
| Title: I | Country: DENMARK |
| | Sponsor: Risø National Laboratory |
| Title: II: SDS, a thermohydraulic subchannel programme for steady-state analysis of water reactors | Organization: Risø National Laboratory |
| Initiated date: May, 1970    Completed date: Autumn, 1974<br>Status: Code made, Verification finished    Last updating: Autumn, 1974 | Scientists: A. Olsen, F. Cortzen, V.S. Pejtersen |

1. General aim

To provide a general subchannel code, for the analysis of thermal-hydraulic, steady-state conditions in water reactors.

2. Particular objectives

The objective is to make a calculation model (code) which can predict, based on sound physical models, local void, mass flow, enthalpy and burn-out margin for safety evaluation and design.

3. Experimental facilities and programme

The experimental programme has been executed concurrently with the code development to supply data on local phenomena in fuel elements with respect to thermo-hydraulics.

Experiments were made on a 7-rod and a 9-rod BWR geometry, as well as on (concentric and excentric) annular geometries.

4. Project status

1. Progress to date

A FORTRAN 4 code exists, capable of solving, on a subchannel basis, steady-state thermal-hydraulic problems, for BWR's and PWR's. Specifically predicting void- and mass-flow-distributions on a fuel-box or core basis. (No limit on number of subchannels, so far up to 400 have been tested).

2. Essential results

On the modelling side a substantial improvement has been made with respect to the formulation of the subchannel equations, physically, mathematically and codewise. This in turn has made it possible to simulate, on a sound basis, hitherto not approachable experiments, such as flow-blockages.

5. Next step

Physical models and correlations must be verified in respect to new and further experimental data.

6. Relation with other projects

The basic subchannel project was a joint venture of three Scandinavian organizations: Danish AEC, AB Atomenergi, Sweden, and Institutt for Atomenergi, Norway. The nordic co-operative project was completed 1973, and the Danish project was an extension of the work. The experience gained with this project has been utilized for the work with the transient subchannel code, TINA, developed within the framework of the NORHAV project.

7. Reference documents

Z. Rouhani: Review of Momentum balance in Subchannel Geometry Eur. Two-phase flow meeting (ETPFM), Brussels 1973.

J. Bosio, O. Imset: Two-phase flow investigations in a 7-rod bundle. ANS European Meeting, Karlsruhe, Oct. 1973.

8. Degree of availability

Not available.

| Title 1 (Original language) | Classification |
|---|---|
| Esperienze di frazione di vuoto in sezioni di prova 8x8 tipo BWR/6 | 10.3 |
| **Title 2 (English)** | **Country** ITALY<br>**Sponsor** NUCLITAL<br>**Organisation** NUCLITAL/CISE |
| Void fraction experiments in 8x8 BWR/6 test sections | |
| Date initiated January 1977<br>Date completed June 1979<br>Last updating May 1977 | **Project Leader**<br>F. LUCCHINI (NUCLITAL) |

1. General aim

Verification and development of design void fraction correlations applicable to steady state and transient analysis.

2. Particular objectives

Obtaining experimental data by a test section simulating as close as possible the actual 8x8 fuel element as far as geometry, spacers, local and axial peaking pattern are concerned.

3. Experimental facilities and programme

The experiments will be carried out in the CISE IETI-4 loop using the quick-closing valve technique.
The following test conditions will be investigated:

pressure 40 - 100 bar, specific mass flowrate 50 - 2000 $Kg/m^2s$, outlet quality from 5% up to critical quality. Two local peaking patterns will be tested.

4. Project status

At present the 8x8 test section is under construction.

5. Next steps

The experiments are planned in the 1978 second half. The analysis completion is scheduled in June 1979.

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| Title 1 (Original language) | Classification |
|---|---|
| Esperienze di frazione di vuoto in sezioni di prova 8x8 tipo BWR/6 | 10.3 |

6. Relation with other projects

These experiments follow a previous void fraction programme carried out by a 4x4 rod bundle .

Results will be strictly related to a following programme concerning operational and LOCA transient conditions which is now under definition.

Dryout experiments will be carried out in the same test section.

7. Reference documents

None

8. Degree of availability

Proprietary

| Title 1 (Original language) | Classification |
|---|---|
| Comportamento termoidraulico di un canale di potenza del reattore CIRENE nella fase di avviamento | 10.3 |
| **Title 2 (English)** | Country : ITALY<br>Sponsor : CNEN<br>Organisation : CISE |
| Thermohydraulic behaviour of a CIRENE power channel during reactor start-up | |
| Date initiated June 1975<br>Date completed December 1976<br>Last updating 1977 | Project Leader<br><br>UTM (CISE) |

1. General aim: Thermohydraulic behaviour of a boiling power channel at positive inlet quality in the low total flowrate range both in steady state and transient conditions.

3. Experimental facilities and programme

3.1. Experimental facility:

- CIRCE: large scale facility simulating in a closed circuit 2 full-scale power channels; water flowrate: 22 kg/s, steam flowrate 3 kg/s; test section maximum power 12,5 MW (d.c.).

3.2. Programme

3.2.1. Steady state tests with positive quality at inlet to determine two-phase flow patterns and to measure critical power.

3.2.2. Transient tests by reducing inlet steam flowrate from the steady state value to zero: measurements of flowrate, pressure, pressure drops, mass transients in the circuit.

4. Project status

4.1. Progress to date: the experimental programme has been completed.

4.2. Essential results: Defined information has been obtained as far as the minimum flowrate is allowed for a stable operation and the procedure to be followed during steam reduction phase.

7. Degree of availability: to a limited extent.

| Title 1 (Original language) | Classification |
|---|---|
| Esperienze di dryout con sezioni di prova 8x8 tipo BWR/6 | 10.3 |
| **Title 2 (English)** | **Country** ITALY |
| Dryout (CHF) experiments in 8x8 BWR/6 test sections | **Sponsor** NUCLITAL<br>**Organisation** NUCLITAL/CISE |
| **Date initiated** January 1977<br>**Date completed** June 1979<br>**Last updating** May 1977 | **Project Leader**<br>G.P. Gaspari (NUCLITAL) |

1. General aim

Verification and development of project dryout (CHF) correlations applicable to steady state and transient analysis.

2. Particular objectives

Obtaining CHF experimental data by a test section simulating as close as possible the actual 8x8 fuel element as far as geometry, spacers, local and axial peaking pattern are concerned.

3. Experimental facilities and programme

The experiments will be carried out on the CISE IETI-4 loop in the following ranges of conditions: pressure 40 - 100 bar, specific mass flowrate 50 - 2000 $Kg/m^2s$.
Two local peaking patterns will be tested.

4. Project status

At present the 8x8 test section is under construction.

5. Next steps

The experiments are planned in the 1978 second half. The analysis completion is scheduled in June 1979.

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| Title 1 (Original language) | Classification |
|---|---|
| Esperienze di dryout con sezioni di prova 8x8 tipo BWR/6 | 10.3 |

6. Relation with other projects

These experiments follow a previous dryout programme carried out by a 4x4 rod bundle.

Results will be strictly related to a following programme, concerning operational and LOCA transient conditions, which is now under definition.

Void fraction measurements will be carried out in the same test section.

7. Reference documents

None

8. Degree of availability

Proprietary

1005

Classification
10.3

| Title 1 | Country |
|---|---|
| Burnout experiments in BWR round tubes | Italy |
| | Sponsor |
| | CNEN |
| Title 2 | Organisation |
| | CNEN-CISE |

| Initiated 1.1.75 | Completed 30/9/75 | Project leaders |
|---|---|---|
| Status planning | Last updating Dec. 74 | V.Marinelli |

1. General aim

   In order to optimize the burnout correlation for a rod bundle it is necessary to have accurate reference to rounde tube data. Afterwords the rod bundle and mixing effects can be taken into account.

2. Particular objectives

   Accurate round tube CHF correlation.

3. Experimental facility and program

   A set of 300 point data will be obtained in the CISE "Ieti-I" plant on two tubular sections 600 cm long and having diameters of 1.26 and 1.7 cm at pressures of 70,50,30 bars and specific mass flow rate between 12 and 200 $gr/cm^2 sec$.

4. Project status

   1) Progress to date The installation of the test sections has been initiated and test data matrix has been decided.

   2) Essential results none

5. Next steps

   First preliminary experiments.

6. Relation with other projects

   The experimental results and the correlation will serve as basis for the burnout experiments in BWR fuel elements.

7. Reference documents

   None

8. Budget

   35 M/lit. / Personnel: 40MM

| Title 1 (Original language) | Classification |
|---|---|
| Esperienze di burnout su elementi di combustibile per BWR | 10.3 |
| **Title 2 (English)** | Country : ITALY<br>Sponsor : CNEN<br>Organisation : CNEN-CISE |
| Burn-out experiments in BWR fuel elements | |
| Date initiated 1972<br>Date completed August 1977<br>Last updating April 1977 | **Project Leader**<br>G. Basso (CNEN)<br>F. Lucchini, G.P. Gaspari (CISE) |

1. General aim

Development of burnout correlations based on the critical quality-boiling length concept, utilizing both the bundle parameters and the subchannel parameters.

2. Particular objectives

Measurements of CHF data on BWR 16 rod test sections, 12 ft long, having different heat flux distributions.

3. Experimental facilities and programme

The experiments have been done in the CISE IETI-3 and IETI-4 loops of 8 MW, in the following range of parameters: pressure 70 ata, specific flowrate G = 12-200 g/cm$^2$ sec.

4. Project status

4.1. Progress to date

The measurements have been completed for radial power distribution: data for 6 different test sections were obtained. Measurements on a bundle having a cosine flux distribution are in progress and will be completed in mid 1977.

4.2. Essential results

Critical quality-boiling length parameter correlates well the data on a bundle parameters basis and a new calculation procedure, named ACHAB method, has been developed to predict the CHF margin in terms of power ratio.
The method, based on rod centered subchannel analysis predicts quite satisfactorily the influence of radial power distribution.

5. Next steps

It has been recognized the necessity of obtaining new round CHF data in order to optimize the rod-bundle CHF correlation. These CHF tests on round tubes, having the same heated diameters as the rod bundle subchannels, are now under way on the CISE IETI-1 facility.

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| Title 1 (Original language) | Classification |
|---|---|
| Esperienze di burnout su elementi di combustibile per BWR | 10.3 |

6. Relation with other projects

CHF round tube experiments will provide data to be used in the theoretical developments of this programme.

7. Reference documents

1) G.P. Gaspari et al. " Experiments on the influence of Radial Heat Flux Distribution on Critical Heat Flux in BWR Bundle Geometry" Reactor Heat Transfer Meeting of Karlsruhe - October 9-11, 1973.

2) B. Marinelli et al. " Dryout Experiments in a 16 rod BWR Geometry with Six Different Radial Test Flux Distribution", 1975 Heat Transfer Conferenc S. Francisco, Cal. August 11-13, 1975.

8. Degree of availability

Open.

| PROJECT TITLE : Basic studies of two phase mixing in fuel cluster geometries. | LWR 10.3 |
|---|---|
| SPONSORING COUNTRY : Commission of the European Communities | ORGANISATION : JRC - Ispra |
| DATE INITIATED : 1973<br>DATE COMPLETED : 1977 | PROJECT LEADER : H. Herkenrath |

Description :

1. General aim

Basic studies of thermohydraulic mixing between the subchannels of a boiling fuel cluster.

2. Particular objectives

To examine boiling mixing (steady state and transient condition in a 16 rod cluster (BOWAL loop). Validation of cluster thermohydraulics codes. Calibration of different void meters for steady state and transient conditions (PRIL loop).

3. Experimental facilities and programme

3.1 PRIL loop

This loop consists of :

- controllable forced circulation
- partial blowdown (~20 %) facility for the transient tests
- pressurizer, condensor, subcooler

The characteristics of PRIL are :

- Maximum power : 600 KW
- Maximum pressure : 200 bar
- Maximum temperature : 365°C
- Special Instrumentation : different void meters.

3.2 BOWAL loop

This loop consists of :

- controllable forced circulation
- partial blowdown (20 %) facility for the transient tests

1010

- pressurizer, condensor, subcooler and preheater (500 KW)

The characteristics of BOWAL are :

| | | |
|---|---|---|
| - Maximum power | : | 3,6 MW |
| - Maximum pressure | : | 250 bar |
| - Maximum temperature | : | 365°C |
| - Special Instrumentation | : | distribution of temperature and power along a number of the rod pressure distribution along the test section. Outlet enthalphy distribution among the 6 basic subchannels of the cluster. |

## 4. Project status

### 4.1 Progress to date

The calibration tests showed the necessity of several modifications, mainly in the regulation and measuring device. The experimental studies under steady state conditions with the BWR-test section (70 bars) have been started in January 1977.

## 5. Next steps

Completion of the steady state measurements with the BWR test section.
Calibration and mounting of 6 x-ray void meters for studies under transient conditions.
Construction of a PWR test section (160 bars).

## 6. Reference documents


- JRC Safety programme progress report 1975 and 1976.
- Herkenrath, H. and W. Hufschmidt
"The pressurized and boiling water loops BOWAL and PRIL for boiling mixing studies of the Heat Transfer Division of JRC, Ispra".
EUR-Report (to be published).
- Herzberger, P., W. Hufschmidt and F. Wind,
"Die Verwendung von Gamma - oder Röntgenstrahlen zur Messung des Dampfgehaltes in Zweiphasenströmungen".
EUR 5415. d (1975).

| ENERGIEONDERZOEK CENTRUM NEDERLAND (ECN) | | CLASSIFICATION : 10.3 |
|---|---|---|
| **TITLE :** Burnout onderzoek aan 9-staafs bundel | | COUNTRY: THE NETHERLANDS |
| | | SPONSOR : Ministry of Social Affairs<br>ORGANIZATION : ECN |
| **TITLE (ENGLISH LANGUAGE):** Burnout experiments on 9-rod bundles | | PROJECTLEADER : S.B. van der Molen |
| INITIATED : 1972 | LAST UPDATING : May 1978 | SCIENTISTS : H. Hoogland<br>D.W. Middleton<br>D. Werner |
| STATUS : nearly finished | COMPLETED : End 1978 | |

General aim

Study of the burnout phenomena in bundles

Particular objectives

Experimental study of the influence of the position of the supporting grids on the value of the burnout-flux and the place where the burnout occurs.

Experimental facilities and program

A testloop for pressures up to 100 bar is available and the power supply of 700kW gives the opportunity of the high heat fluxes needed for the burnout investigations. As burnout detection method, the Wheatstone bridge principle is used.

Project status

The burnout flux of the central rod of a 9-rod bundle is determined at a pressure of 70 bar as a function of a massflow rate, in the range of 700 - 1800 kg/m$^2$ sec, the heatflux from the outer rods and as function of the position of (58, 126 and 240 mm) of the supporting grids with respect to the outlet side of the bundle. The project concerns experiments performed in a bundle with an axial uniform heat flux distribution, whereas the radial heat flux distribution can be varied.

Next steps

Calculations with Cobra III c and comparison with experimental results. Completion of the study through a report, scheduled end of 1978.

Relation to other projects: no

Reference documents

R.C.N. Technical memo 1063-020 (in Dutch)

Degree of availability:

Through ECN library channel

Budget: -

Personnel: -

| | |
|---|---|
| ENERGIEONDERZOEK CENTRUM NEDERLAND | CLASSIFICATION : 10.3 |
| **TITLE :** Ontwikkeling rekenprogramma VITESSE Stationaire temperatuur en snelheidsverdelingen in onsamendrukbare fluïda | COUNTRY: THE NETHERLANDS |
| | SPONSOR : ECN |
| | ORGANIZATION : ECN |
| **TITLE (ENGLISH LANGUAGE):** Development computer code VITESSE Stationary temperature and velocity distribution in incompressible fluids | PROJECTLEADER : Slagter, W. |
| INITIATED: February 1975 / LAST UPDATING : June 1978 | SCIENTISTS : Roodbergen, H. Vonka, V. Spreeuw, E. |
| STATUS : progressing / COMPLETED : 1979 | |

General aim

The development of a computer program that enables the calculation of local temperatures and velocities in a bundle geometry.

Particular objectives

The computer code is a three-dimensional program and makes use of the finite element method.

Experimental facilities and program

Experiments based on the Laser Doppler method will be considered.

Project status

The computer code has been verified on basis of experiments performed by others and accepted calculations with regard to well defined configuration, known from the literature. The physical model for fully developed two-dimensional turbulent flows in a bundle geometry has been completed. Two turbulence-models were investigated a) Prandtl's mixing length model

b) Kinetic energy of turbulence model.

Next steps

Three dimensional turbulent flows will be incorporated.

Relation with other projects: -

Reference documents

Hoekstra, E.K. (Compiler) Fast Reactor program Fourth Quarter 1975. Progress report. RCN-244; Petten, March 1976.
Slagter, W., Roodbergen, H.A., TRIP: A finite element computer program for the solution of convective heat transfer problems. RCN-243; Petten, January 1976.
Slagter, W., Finite element analysis for turbulent flows of incompressible fluids in fuel rod bundles; published in Nuclear Science and Engineering, vol. 66.
Slagter, W., Roodbergen, H.A. and Dekker, N.H., Prediction of fully developed turbulent flow in noncircular channels by the finite element method; presented in Instanbul, July 1978.

Degree of availability

Upon mutual agreement at ECN-Petten

Budget: -

Personnel: -

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| ENERGIEONDERZOEK CENTRUM NEDERLAND (ECN) | CLASSIFICATION : 10.3 |
|---|---|
| **TITLE :** Experimenteel onderzoek naar mixing en de invloed van rooster op mixing in een 36-staafs bundel | COUNTRY: THE NETHERLANDS<br>SPONSOR : ECN<br>ORGANIZATION : ECN |
| **TITLE (ENGLISH LANGUAGE):** Mixing experiments in 36-rods bundle and the influence of supporting grids on mixing | PROJECTLEADER : S.B. van der Molen |
| INITIATED : 1973 — LAST UPDATING : May 1978<br>STATUS : finished — COMPLETED : July 1978 | SCIENTISTS : H. Hoogland<br>A. Warmenhoven |

General aim

Experimental investigation of the mixing phenomena in a bundle geometry.

Particular objectives

Experimental study of the heat and mass exchange between the subchannels in a bundle in which a nonhomogeneous heat distribution is generated.

Experimental facilities and program

Testloop for low-pressure experiments.

Project status

A 6x6 bundle is provided with a length of 6 heating rods in special positions. At this moment at two axial positions in the bundle the temperature of the flowing liquid is measured by a number of thermocouples over the cross-section of the bundle in order to investigate the heat exchange between the subchannels in case of the nonhomogeneous heatflux distribution due to the heat production by only 6 rods of the total of 36. Measurements have been carried out with and without a supporting grid positioned between the two temperature measuring planes.

Next steps

The obtained data show that for low mass flows mixing does not increase when a supporting grid is installed. A report of the measurements-results is planned in July 1977.

Relation to other projects: No.

Reference documents: No.

Degree of availability: Through ECN library.

Budget: -

Personnel: -

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX

1017

Classification

10.3

| | |
|---|---|
| Title 1<br><br>DRYOUT IN BOILING CLUSTERS | COUNTRY<br>UNITED KINGDOM |
| | SPONSOR UKAEA |
| | ORGANIZATION<br>AEE WINFRITH |
| Title 2 | Project Leader<br><br>J OBERTELLI |
| Initiated 1966     Completed :<br>Status :     Last updating | Scientists: |

Description:

1. General Aim

To control the conditions for a boiling-type reactor so that dryout is sufficiently unlikely to occur; this implies a knowledge of the consequences of dryout.

2. Particular Objectives

To observe dryout in an out-of-pile rig under conditions representative of the reactor.

3. Experimental Facilities

The 9 MW electrically heated rig capable of taking a full-sized bundle.

4. Project Status

Dryout has been studied as a function of the bundle geometry, flow, etc.

5. Next Steps

Further measurements to study the effect of power distribution and bundle geometry, and the effect of a continuously-running spray.

Reference Documents

Internal documents.

Obertelli and Owen AEEW-M935 (Unclassified).

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 10.4 | RS 97 A |
| Vorhaben/Project Title: Körperschallmessungen an Reaktordruckbehältern und am Primärkreislauf von Kernkraftwerken<br><br>Solid-Borne Sound Measuring on Reactor Pressure Vessels and on the Primary Circuit of the Cooling System | | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Allianz-Zentrum für Technik<br><br>8045 Ismaning |
| Arbeitsbeginn/Initiated: 15.4.1975 | Arbeitsende/Completed: 14.4.1977 | Leiter des Vorhabens/Project Leader: Dipl.-Ing. Raible |
| Stand der Arbeiten/Status: Completed | Berichtsdatum/Last Updating: December 77 | Bewilligte Mittel/Funds: 747.861,-- DM |

## 1. General aim

The object of Project RS 97 A is the detection of impending damage to the core components and in the primary circuit of the cooling system at an early stage. An essential feature of this measuring system is that the transducers are located outside of the reactor pressure vessel and the remaining cooling system, and that information can thus be obtained on the behaviour of the core components and the primary circuit during operating, with simultaneous monitoring thereof.

## 2. Particular objectives

The particular objectives and main emphasis of Project RS 97 A are:

2.1 Study of parameters on an experimental loop for a Leak Detection and Monitoring System based on acoustic emission.

2.2 Development of an enlarged Loose Parts Monitoring System (LPMS), with which loose and loosened parts can be detected (on-line) and located (off-line).

## 3. Research program

3.1 The research program for the study of parameters for the Leak Monitoring System embraced a total of about eighty experiments: Variation of the following parameters was carried out:

| | |
|---|---|
| Four Types of leaks: | leaking flanges, leaking valves, cracks, nozzle throat. |
| Two leakages rates: | 24o l/h to 24 l/h |
| Three enthalpies: | steam, water/steam, water |
| Background noise: | yes/no |

Investigations were carried out to determine what is the best type of transducer, in which optimum frequency range leaks are detectable and in which manner different types of leaks are discernible in solid-borne sound by means of noise analysis.

3.2 LPMS: In continuation of Project RS 97, it is planned to design and produce a prototype of a Loose Parts Monitoring System. This system will be tested with the aid of signals from loose parts in a reactor pressure vessel which were recorded on magnetic tapes.

## 4. Experimental facilities

4.1 For the study of parameters, the Kraftwerk Union Erlangen has made available an experimental loop, which made it possible to simulate leaks under specific reactor conditions.

4.2 For the production of printed circuits and a prototype, the requisite facilities are available at the Allianz-Zentrum.

4.3 Computer codes: For analysing the noise of leakages, the power spectra are calculated, which are issued via display or X-Y-recorder and are also stored on discs for subsequent recall.

## 5. Progress to date

5.1 In cooperation with the Kraftwerk Union Erlangen and the Battelle-Institut Frankfurt, the research program on leaks was implemented. The noises of different types of leakages were recorded on magnetic tapes and were then analysed in off-line procedure. During the course of the tests, television recordings were also made on magnetic video tapes for optical documentation of the water or steam jet. Some tapes were handed over to the Reactor Safety Association Garching for special moise analysis.

5.2 Circuits for a data logger of a Loose Parts Monitoring System were designed and prepared. The data logger is mounted into a 19" rack-cabinet.

This system works in accordance with the following operating modes:

- automatic data logging: if typical solid-borne sound signals occur in the form of burst signals and these exceed the threshoed of the level discriminator, the solid-borne sound signal indicator is set. The first incoming measuring point is stored and displayed. The magnetic tape recorder is started and then records the signals for a certain time.

- operating by hand: the 4 channels of the magnetic tape recorder can be allotted to each measuring point. The recording is started by hand.

- replay from the tape recorded signals is possible. They can be recorded on the recorder strip chart as a hard-copy.

- an audio module permits the hearing of signals from selected accelerometers. It acts either as a remote stethoscope listening to the operation of the selected components or as equipment for receiving the recorded signals from the tape.

- an early warning of impending malfunction can be obtained by comparing known audio or tape records with operating signals.

5.3 The Loose Parts Monitoring System was enlarged by a computer code, which computes the ranges of the sources of the burst-signals after input of the delay times between the different measuring points. Another code outlines the cylindrical shell of the reactor pressure vessel and plots the location of the source of the burst signals with the aid of the hyperbolic cross-method.

## 6. Essential results

6.1 The evaluations of the pre-experiments for leak monitoring are completed. The essential results are:

- spectrum analyses and evaluations of the RMS values of the narrow and wide bands as well signals show that the the noise level depends on the leak cross section as well as on the leakage rate.

- good leak detection efficiency was demonstrated against background noises existing at the time.

- it was possible to detect all investigated leakages rates ($\geq$ 25 kg/h), even with the most remote transducer (6 m from the leak).

- the limit of detection for the nozzle throat, leaking flanges, leaking valves and cracks - determined by means of linear regression - is in the proximity of 1o kg/h and thus essentially lower then the smallest simulated leakage.

## 7. Next steps

The pre-experiments on a test loop with simulated leakages, clearly showed that

- local leak monitoring using transducers near the component is easily possible.

- it is still not possible to determine which of the optimum frequency ranges and which of the qualified transducers suitable for a global Leak Monitoring System is the best, using only a few transducers over a wide range of the cooling system. Without knowledge of the operating noises of the various types of reactors final conclusions cannot be drawn. Noise measurements are therefore to be carried out at various power plants on different components during operation.

## 8. Relation with other projects

Battelle Institut Frankfurt:
RS 193: "Pre-experiments for Leak Monitoring by means of Acoustic Emission Analysis".

## 9. Reference

P. Jax, Battelle Institut Frankfurt
"Vorversuche zur Leckageüberwachung mit Hilfe der Schallemissionsanalyse".
BF-R-62.944-1.

B. Raible, Allianz-Zentrum für Technik Ismaning
"Körperschallmessungen an Reaktordruckbehältern und am Primärkreislauf von Kernkraftwerken".
Teil 1: Vorversuche für ein Lecküberwachungssystem.
Bericht zum Forschungsvorhaben RS 97 A.

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

All reports specified under 9 are available at the "Gesellschaft für Reaktorsicherheit", Glockengasse 2, 5000 Köln.

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.08.1977 - 31.12.1977 | 10.4 | RS 290 |

| Vorhaben/Project Title | |
|---|---|
| Geräuschmessungen an Kernreaktoren im Frequenzbereich von 1 kHz bis 1 MHz<br><br>Noise Measuring on the Primary Circuit of the Cooling System in Nuclear Power Stations | Land/Country: FRG<br>Födernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Allianz-Zentrum für Technik, 8045 Ismaning<br><br>Elektrotechnik und Elektronik (ELT) |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.8.1977 | 30.09.1978 | Dipl.-Ing. Raible |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | December 1977 | 354.000,-- DM |

## 1. General aim

The object of this project is the development of a Leak Detection and Monitoring System based on acoustic emission, with the aid of which leakages can be detected and located on pressurized components.

As opposed to the hitherto applied methods of leak detection the significant features of an acoustic emission system are:

- very short response time
- higher sensitivity
- precise leak location capability.

## 2. Particular objectives

In continuation of the development of a Leak Detection and Monitoring System for the reactor coolant loop based on acoustic emission, the particular object of this project is the measuring and analysing of noises, which are emitted during operation of the PWR and BWR and a test circuit in the frequency range from 1 kHz to 1 MHz. By the installation of various types of sensors, data is obtained on which type of sensor is best suited for the most favourable frequency range.

The so-called "operating noises" depend on the operating conditions, e.g. on the parameters of the medium in the cooling system.

Typical "operating noise patterns" are measured in the above-mentioned frequency range and compared with the "leakage-noises" of simulated leaks.

Furthermore, the propagation of sound in the actual structures of various components is investigated with the aid of an artificial noise source.

When the noise pattern dependent on the various locations in a reactor cooling system and on the working conditions is known, an "atlas of noises" can be compiled, which then represents the basis for an acoustic monitoring system.

## 3. Research program

The points of main emphasis of this project work are measurements of the operating noise in a wide frequency band during the phase of initial start-up (zero power phase) and during power output. The project embraces the following 5 points:

- extensive operating noise measurements using 10 measuring points on the reactor cooling system of a Pressure Water Reactor (PWR) during the non-nuclear phase of the initial start-up (first hot-check-operation) and at the same time, measurements of the sound propagation, too.

- comparative measurements in a PWR at 4 measuring points on the cooling system during power output (nuclear phase).

- comparative measurements in a BWR at 4 measuring points on the reactor cooling system during power output.

- operating noise measurements at a test plant to compare the noises in a reactor plant and a test plant using 4 measuring points.

- the development of an artificial noise source for sound propagation and attenuation measurements (Battelle-Frankfurt).

In cooperation with the Kraftwerk Union Erlangen, the Battelle-Institut Frankfurt and the Gesellschaft für Reaktorsicherheit Garching, the research program was elaborated and will be carried out in joint partnership.

## 4. Measuring facilities

The measurements are carried out on two types of reactor power plants, the BWR and PWR and on a test plant of the KWU Erlangen.

Measurement facilitation depends on timing and on the moment when a PWR is in the start-up phase or a PWR or a BWR is in the refuelling phase.

## 5. Progress to date

In 1976, the noise of simulated leakages on a test loop was measured using accelerometers for the low frequency range and ultra-sonic transducers for the high frequency range. The spectrum of the leakage noises and the measurements of the narrow-band RMS values show that the noise level depends on the leak cross-section as well as the leakage rate, in both the low frequency and the high frequency range. From the background noise existing of the time, it was shown that there was good sensibility for leak detection.

On the other hand, before deciding on the optimum frequency range and on the best type of sensor for a leak monitoring system, the actual operating noise in the above-mentioned frequency range, under various working conditions and on various components, must be known.

For this project RS 290 in 1977, the first step carried out was noise measurements on actual reactor components of the Power Plant Biblis A after refuelling. The most important analyses of these measurements are to hand.

## 6. Essential results

The essential results are:

- the noise level in the reactor components is higher than on the test-loop, where measurements were conducted in 1976.

- in a power plant during power output, the noise level differs in the various components. It increases in the following sequence:
  - main-steam pipe
  - main-cooling pipe hot (reactor vessel outlet)
  - feedwater pipe
  - main-cooling pipe cold (reactor vessel inlet).

- the noise-level fluctuates as time progresses, this applying to both the narrow-band and wide-band measured values.

This behaviour is still under discussion; an explanation for this is not yet to hand.

## 7. Next steps

As the next step, it is planned to conduct measurements at the Power Plant Gösgen during the non-nuclear first and second phase of the initial start-up (1st and 2nd hot-check-operations).

## 8. Relation with other projects

RS 97 A, Allianz-Zentrum für Technik
"Solid-Borne Sound Measuring on Reactor Pressure Vessels and on the Primary Circuit of the Cooling System.
1st part, Pre-Experiments for a Leak Monitoring System."

RS 193, Battelle-Institut Frankfurt
"Pre-Experiments for Leak Monitoring by means of Acoustic Emission Analysis."

RS 289-RS 292. The project RS 290 must be seen in close connexion with the projects RS 289 to RS 292.

## 9. Reference

P. Jax, Battelle-Institut Frankfurt
"Vorversuche zur Leckageüberwachung mit Hilfe der Schallemissionsanalyse"
BF-R-62.944-1

B. Raible, Allianz-Zentrum für Technik Ismaning
"Körperschallmessungen am Reaktordruckbehälter und am Primärkreislauf von Kernkraftwerken."
Teil 1: Vorversuche für ein Leckageüberwachungssystem.
Bericht zum Forschungsvorhaben RS 97 A.

## 10. Degree of availability

All reports specified under 9. are available at the "Gesellschaft für Reaktorsicherheit, Glockengasse 2, 5000 Köln.

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.8.77-31.12.77 | 10.4 | RS 289 |

| Vorhaben/Project Title | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor |
|---|---|
| Geräuschmessungen an Kernreaktoren im Frequenzbereich 1 kHz bis 1 MHz<br><br>Noise Measurements on Nuclear Reactors in the Frequency Range Between 1 kHz and 1 MHz | Battelle-Institut e.V. Frankfurt<br>Abt. Betriebsverhalten von Werkstoffen |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| August 1, 1977 | September 30, 1978 | Dr. P. Jax |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | December 31, 1977 | DM 582.452,61 |

1. General Aim

Development of a surveillance system for leaks on an acoustic basis for the primary system of a nuclear power plant. Compared with the methods previously used for leak detection, this surveillance system is to show fast reponse and higher sensitivity at a substantially improved accuracy of locating leaks.

2. Particular Objectives

The structure-borne sound emitted by nuclear reactors and experimental plants during operation in the frequency range from about 1 kHz to about 1 MHz is to be recorded and analyzed. Use of various sensor types - the institutions participating in this project, i.e. Allianz-Zentrum für Technik, Ismaning; Gesellschaft für Reaktorsicherheit, Garching; and KWU, Erlangen, will make measurements in the low-frequency range up to 80 kHz, while Battelle will perform measurements in the ultrasonic range - is to provide information about the optimum frequency range for a surveillance system for leaks and the most suitable sensor type. For this purpose the noise patterns typical of PWR and BWR systems will be examined and compared with the "leakage noise" measured on experimental plants and artificial leaks.

In addition, the acoustic wave propagation in the real structures is to be investigated by using an artificial noise source.

## 3. Research Program

3.1 Extensive measurements of the operating noise at the cooling system of a PWR while putting it into operation (hot trial runs) at ten measuring points, and measurements of acoustic wave propagation

3.2 Reference measurements on a PWR during power operation, with four measuring points at the reactor cooling system

3.3 Reference measurements on a BWR during power operation, with four measuring points at the reactor cooling system

3.4 Measurements of operating noise on an experimental loop of the KWU in Erlangen, with four measuring points, for comparison with the operating noise measured on nuclear reactors, to be used as a basis for future investigations

3.5 Development of an artificial noise source for acoustic wave attenuation measurements

## 4. Experimental Facilities, Computer Codes

Ad 3.1 to 3.4 The measuring line consists of a sensor, a wide-band preamplifier (amplification 40 dB) with a band-pass filter adjusted to the frequency range and a main amplifier. The signals are temporarily stored on a magnetic tape unit with a high upper frequency limit of 1.6 MHz. For evaluation during the experiment and afterwards from the tape, the average signal levels are recorded by means of an r.m.s. voltmeter, and the frequency spectra are evaluated by means of a Fourier analyzer consisting of transient recorder, Fourier processor and digital magnetic tape storage unit.

Ad 3.5 The noise source consists of a mobile compressor with reducing valve for exact pressure adjustment, one long flexible tube, and the head that can be attached to the structure by means of a magnetic holder; in this head the noise is generated by the air passing through a nozzle.

## 5. Progress to Date

Ad 3.1 Installation of 24 sensors at 11 measuring points in the primary system of the Gösgen-Däniken nuclear power station. At each measuring point there is a wide-band high-temperature sensor (sensitive in the range between 0.1 and 1 MHz), which is directly attached to the structure by means of a heat-resistant silicone paste, and a resonant 70-kHz sensor attached through waveguides. Depending on the local conditions, the holder is either screwed in or attached by a clamp connection.

Ad 3.2 Installation of ten sensors at four measuring points in the Biblis A nuclear power station. Sensors same as above (Gösgen-Däniken nuclear power station); fixed by clamping devices.

Recording of the operating noise on magnetic tape during selected periods of time. Measurement of the r.m.s. values. Start of the evaluation of the magnetic tape recording.

Ad 3.5 Design and construction of a noise source for the generation of continuous noise by escaping air.

Investigation of the effect of various parameters (nozzle-to-object spacing, nozzle shape, nozzle diameter, air pressure, and surface roughness of the object to be examined), using a large steel plate and a steel block, on r.m.s. value and frequency spectrum in the case of wide-band recording of the noise.

The work under item 3.5 has largely been completed.

## 6. Results

Ad 3.5 The objective of this phase of research (noise source reproducibly generating wide-band noise in the frequency range between 10 kHz and 1 MHz) was achieved with the following parameters: air pressure 10 bar, Laval nozzle with 1 mm diameter and 1 mm or 10 mm distance from the object.

A comparison with first results obtained in measurements at Biblis showed that the AE intensity of the source at a nozzle distance of 10 mm is of the same order of magnitude as that of the operational noise at Biblis.

The bandwidth of the emitted frequency spectrum was found to increase with decreasing nozzle diameter and distance and with increasing pressure. In addition, these systematic investigations provided information which is of interest in connection with the generation and propagation of leakage noises.

## 7. Next Steps

Ad 3.1 Measurements during hot trial runs

Ad 3.2 Evaluation of the measurements

Ad 3.3 Installation of sensors
+ 3.4

## 8. Relation with Other Projects

The research is being conducted in close cooperation with the Allianz-Zentrum für Technik, Ismaning; with KWU, Erlangen; and with the Gesellschaft für Reaktorsicherheit, Garching (projects RS 290 to RS 292). It is based on results of the research project RS 193.

## 9. References

./.

## 10. Degree of Availability of the Reports

./.

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. 77 - 31. 12. 77 | 10.4 | RS 291 |

| Vorhaben/Project Title | |
|---|---|
| Geräuschmessungen an Kernreaktoren im Frequenzbereich ca. 1 kHz bis ca. 1 MHz<br><br>Noise Measurements on Nuclear Power Plants Within a Frequency Range of approx. 1 kHz up to approx. 1 MHz | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik<br>R 214, Erlangen |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1. 8. 77 | 30. 9. 78 | Dr. Fischer |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | 31. 12. 77 | 430.240,-- DM |

1. General Aim:

To develop a leakage monitoring system on an acoustic basis for pressure boundaries of safety related reactor systems.

Compared with current leakage detection methods, this type of monitoring system is expected to achieve quick response, enhanced sensitivity and a considerably improved localization of leakages.

2. Particular Objectives

Subsequent to the development of an acoustic leakage monitoring system for reactor cooling systems, it is the aim of the program to evaluate and analyse the sonic emission in the structure of operating reactor and test plants within a frequency range of approx. 1 kHz up to approx. 1 MHz. By the use of different types of transducers, the optimum frequency range for a leakage monitoring system, and the most suitable transducers will be determined.

This sound emission, here called "operating noise" depends - among other things - on the operating condition of a plant, i.e. on the parameter of the cooling system medium. Thus typical "operating noise" sound patterns will be established for both PWR and BWR plants within the above frequency ranges which then will be compared with "leakage sounds" observed in test plants and simulated leakages.

Furthermore the sound emission caused by an artificial noise source in real structures will be investigated.

3. Research Program

Central point of the experimental part of the program will be broad-band measurements of operating noises during start-up and during power operation.

The test matrix is as follows:

3.1 Comprehensive measurements of the operating noises observed on a PWR during start-up operation (hot functional test) using 10 measurement points for the reactor cooling system and sound propagation and attenuation measurements.

3.2 Reference measurements on a PWR during power operation using 4 measurements points in the reactor cooling system.

3.3 Reference measurements performed in a BWR during power operation using 4 measurement poonts in the reactor cooling system.

3.4 Measurements of the operating noises observed in a test plant as compared to those obtained from reactors giving a basis for further investigations with 4 measurement points.

3.5 Development of an artificial noise source for attenuation measurements.

4. Experimental Facilities

Plant measurements performed both during commissioning phase and in operation as well as preparation for such measurements including necessary installation and assemblies, access to hard- and software required as well as close co-operation with commissioning and/or operating personnel.

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

To 3.1

In Nov. 77 measurements in Biblis A were performed at:

- main coolant line hot
- main coolant line cold
- feed water line
- steam line

Measurement procedures for first measurements in Gösgen-Däniken (first hot operational test) were prepared and checked.

To 3.5

The noise source developed at the Battelle-Institute Frankfurt (BF) was tested in Dec. 77 using a BF test plate. The noise spectrum emitted was recorded in the low-frequency area as a function of various parameters. The results obtained provide the basis for the use of the noise source in Gösgen-Däniken.

6. <u>Results</u>

To 3.1

The signal amplitudes - given in g-RMS - of the operating noises obtained were recorded at the above measurement points at various times and analysed with respect to height, local distribution and operation-related variation.

During operation almost periodical intensity fluctuations were observed on the main steam line which may be caused - according to present state of knowledge - by a minor, almost periodically opening "internal leakage". According to the operator, a correlation exists to the main steam quantity indication.

To 3.5

It was found that the distance of the compressed-air nozzle from the test plate is a decisive parameter for the low-frequency portions of the noise spectrum. While very high frequency portions ( 500 kHz) result from the 1 mm distance, at 10 mm distance higher intensities will be excited in the low-frequency part of the spectrum ( 100 kHz). From the intensities measured the use on flow-through components appears to be promising.

7. Next Steps

To 3.1

Further detailed evaluation.
Performance of measurements and evaluation.

To 3.3

Organisational and technical preparations in order to perform the working program during nuclear operating test in NPP Isar.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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1037

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.10. - 31.12.1977 | 10.4 | RS 292 |

| Vorhaben/Project Title | |
|---|---|
| Noise Measurements in Nuclear Reactors, Frequency Range from ca. 1kHz to ca. 1 MHz<br>Geräuschmessungen an Kernreaktoren im Frequenzbereich von ca. 1 kHz bis ca. 1 Mhz | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: GRS Garching |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.8.1977 | 30.9.1978 | Dr. D. Wach |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | 31.12.1977 | 125..5o4,-- DM |

## 1. General Aim

At present, within various research projects concerning the quality assurance in nuclear power plants new on-line surveillance methods and systems are being developed, which permit an early detection and localisation of mechanical faults within the primary circuit of power reactors (e.g. vibration surveillance, loose particle detection). Part of this problem area is the development of a new leakage surveillance system using acoustic noise. It is the aim of this project, to investigate the possibilities of such a system, which as compared to systems used up to now, has a higher sensitivity, a faster response and improved localisation possibilities.

## 2. Particular Objectives

Allianz Ismaning, Battelle Frankfurt and KWU Erlangen have proved the principle applicability of accelerometers and ultrasonic gauges in the course of previous investigations in a test loop with simulated leaks. As a decision on the applicability in reactor plants, the optimal frequency range and the sensors which are best suited for leakage detection can be achieved only when knowing the noise generated during normal reactor operation, the present investigations shall be performed in the primary systems of nuclear power plants. The two objectives of this work are

- the investigation of the influence of the structure upon the sound propagation by means of correlation analysis
- signal management by means of a uniform system in order to enable a comparison of the results gained from signals meas-

ured by the various participants in the project during selected operational conditions.

Tests with an artifical noise source are of special importance. Comparing the measuring results with and without noise source permit to investigate the filter behaviour of the reactor structure, the actual sound propagation and finally, to what extent the results of the test loop can be transferred to an actually operated reactor coolant loop.

## 3. Research Program

3.1 Planning and performing of the measurements of the operational noise in the framework of the committee responsible for the projects RS 289 to RS 292. (Tests shall be performed during a first (test) operation of a PWR and during power operation of a PWR and a BWR.)

3.2 Digitalisation and investigation of signals measured by different sensor types during selected test phases, by means of a stochastic signal analysis. Comparison of results with reference to the different measuring methods.

3.3 Analysis of selected signals by means of correlation methods in order to investigate the influences of the reactor structure upon the sound propagation and considering the background noise.

## 4. Experimental Equipments, Computer Programs

The computer codes DIGI, DIGCHN, TUMX92, TUMX68 will be used. Further on special storage and analysis equipment will be needed.

## 5. Progress to Date

Planning of the measurements in Biblis A and Gösgen was performed in cooperation with the other participants in the project. First analyses were performed with the Biblis A data.

## 6. Results

The investigations were started with the reference measurements in Biblis A because of operational reasons. First results are to be expected in connection with the analyses of the signals to be measured at the Gösgen plant.

7. Next Steps

In the beginning of 1978 measurements of the operational noise and the artifical noise source in the Gösgen plant will be performed. The BWR-measurements will be done during the second quarter.

8. Relations With Other Projects

The project has to be seen in tight connection with RS 289 (Battelle Frankfurt), RS 290 (AZT Ismaning) and RS 291 (KWU Erlangen). It is based upon the results gained from the former projects RS 97A and RS 193.

1041

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 01.01.1977-31.12.1977 | 10.4 | RS 0066A |

| Vorhaben/Project Title | |
|---|---|
| Modellmäßige Behandlung der Schwingungen von Kerneinbauten in Leichtwasserreaktoren<br><br>Model calculation of vibrating core components in light water reactors | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Lehrstuhl f. Apparatetechnik und Anlagenbau<br>Universität Erlangen |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 01.11.1974 | 31.03.1978 | Prof.Dr.-Ing.E.Klapp |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | December 1977 | 486.315,-- DM |

1. General aim.

Apparaturs and plants using large quantities of fluids are often damaged by flow-incluced vibrations. It is not possible by theory to gain information concerning the best way of laying out a mechanical structure to reduce or eliminate vibrations. Under real conditions the flow pattern itself and the geometry defined by the mechanical structure are very complicated. It therefore remains for the experimental engineer to explain the relationship between the movement of the mechanical structure and the excitation caused by the flow. For a systematic investigation in the planning stage experiments making use of models can also be carried out complementary to measurements taken at the plant itself. Model technology is justified theoretically in mechanics of similarity.

2. Particular objectives

Under investigation are the flow-induced vibrations of the core components in the reactor pressure vessel of PWR caused by the flow of the main coolant. The measurements are taken of pressure vessel models (scale ~ 1:10) of normal construction (4-loop-plant).
The main goal is to obtain criteria for a safe design of core components and to test if model experiments in the project stage of new plants are suited to predict damages caused by flow-induced vibrations. To solve this problem it is necessary to know the response of the structure according to the dynamical load caused by the main coolant flow. For a safe design of a mechanical structure it is important to be sure that there is no dominant feedback-mechanism (fluid-resonant, fluid-dynamic, fluid-elastic) in the flow field; the effect of feedback is always a selective amplification of the load energy about

dominant frequency bands associated with an increase of spatial correlation. Without feedbackmechanism the load energy in a turbulent flow, which causes the structure to vibrate, has a statistical distribution being unable to excite the structure systematically

3. Research programm

3.1 Derivation of the characteristic numbers relevant for this problem, from the mechanics of similitude.

3.2 Experimental research on a model similar to a hydrodynamic setup with air as flow medium (uncoupling the flow field from the influence of the structural movements)

3.3 Experimental research on a model similar to a hydro-elastic setup with water as flow medium (viewing of the flow system coupled with the structural movements)

3.4 Possibilities and limits of the theoretical treatment of vibrations induced by the flow of the main coolant in the core components of PWR; development of a semiempirical mathematical model

4. Experimental facilities; computer codes

See this report for 1976

5. Progress to date

3.2 Completion of the core assembly and experiments with the hydrodynamic similar model

3.3 Experimental setup and instrumentation of the hydro-elastic similar model

3.4 Theoretical work in the field of flow-induced structural vibrations.

6. Results

3.2 It was shown by acoustic excitation that singular peaks in the spectrum of the pressure pulsations, dynamically straining the core components, are due to resonance phenomena in the model's zone of flow. Superposed to these peaks are flow induced influences. Since both phenomena have different physical causes, there can be no common scale factor in the test data's transfer from model experiments with air as medium of flow to a similar large scale setup. This is true for both, the amplitude range-as well as for the frequency range of the pressure pulsations:

a.) Fluid-Resonance Effects

Aside the geometry the velocity of sound in the flow field is the essential influence for the dimensionless frequency scale. For the dimensionless amplitude scale the interferance energy ist next important to the damping in the flow field.

b.) Fluid-Dynamic Effects

For the dimensionless frequency scale is the geometry next to the velocity of flow of essential influence; in the amplitude range the flow's degree of turbulence aside from a characteristic pressure head is especially suited for a dimensionless representation of the pressure pulsations.

Consequently both influences must be recognized and separated for a meaningful transfer of this experimental program's test data to a large scale setup

3.2 During the period of this report it was possible to complete the construction of the model and the final setup of the experimental plant. The general view (figure 1), shows the hydro-elastic model of the reactor pressure vessel with core components. So far it was possible to implement the concept of simultaniously measurement of movement - and load factors at the same positon of the mechanical structure. E.g. there were adapters built for the test positions at the core barrel, each adapter holding 2 pressure transducers and 3 acceleration transducers. The first test runs were used to check the experimental plant in actual use and to evaluate the pump's and loop's influence on the pressure vessel.

3.3 In a theoretical study the fluid's influence on the behavior of vibration of the adjoining mechanical structure was analyzed. This was based on the common model of hydro-elasticity (figure 2). Through simplified assumtions approximative solutions were to be found, which describe the experimental results adequately exact without demanding excessive calculations. Naturally the geometrical conditions are of substantial influence. Through the use of energetic methods estimations could be found on the virtual mass and additional damping (due to the fluid in the ring gap) between pressure vessel and core barrel.

## 7. Next steps

3.2 Conclusion of the experiments with air as the flow medium

3.3 Measurements with the hydro-elastic model; experimental determination of the mechanical properties of the hydro-elastic model

3.4 Development of semi-empirical models of flow induced vibrations of the core components of PWR

## 8. Relation with other projects

See this report for 1974

## 9. References

Annual report for the GRS/BMFT (German)

1045

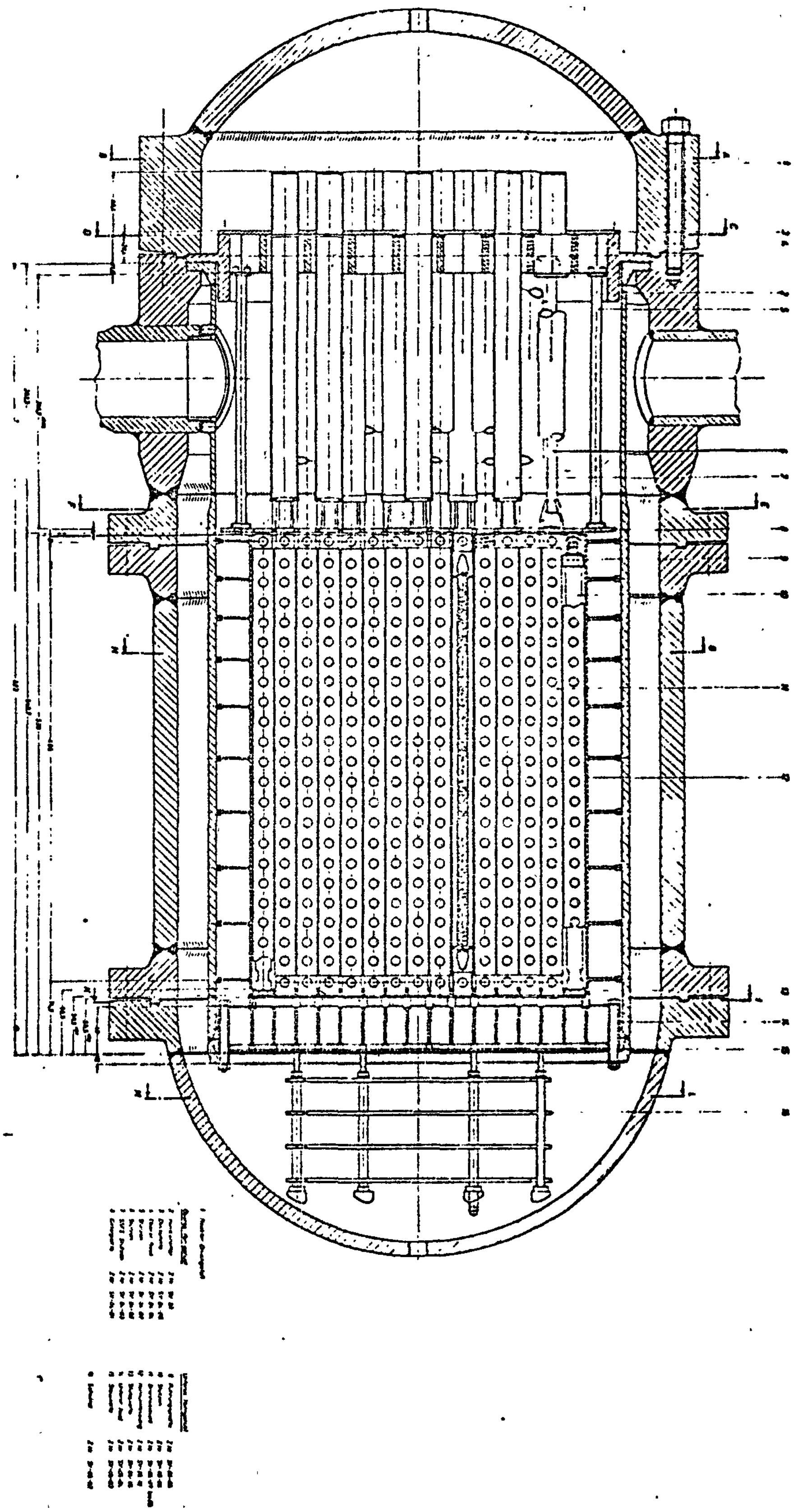
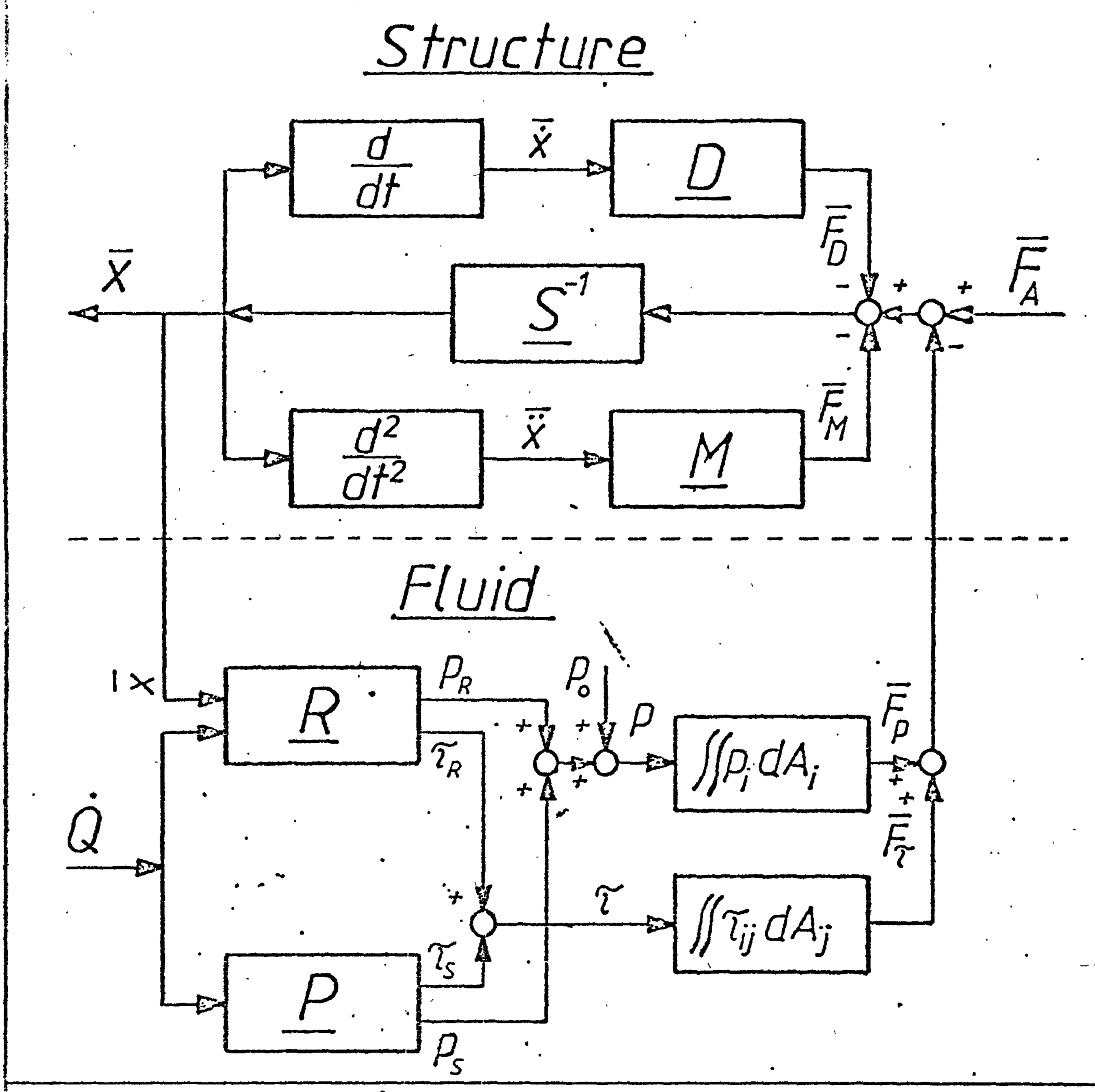


Fig.: 1




Legend:

D : Damping operator
S : Stiftness operator
M : Mass operator
R : Feed-back operator
P : Fluid-dynamic operator
t : Time

p : Pressure
τ : Shear stress
A : Area
F : Force vector
X : Displacement vector

| | | |
|---|---|---|
| UEN AA | Hydro-elastic System | Fig.: 2 |

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. 77 - 31. 12. 77 | 10.4 | RS 218 |
| **Vorhaben/Project Title** | | **Land/Country** FRG |
| Experimentelle Untersuchungen zur Absicherung von Druckbehältereinbauten gegen strömungsinduzierte Schwingungen beim Siedewasserreaktor, Modell-Vorversuche<br><br>Experimental Investigation of Protection of Pressure Vessel Internals against Flow-Induced Vibrations in BWR's, scaled pre-tests | | **Fördernde Institution/Sponsor** BMFT<br>**Auftragnehmer/Contractor** KRAFTWERK UNION AG Reaktortechnik<br><br>R 52, Karlstein |
| **Arbeitsbeginn/Initiated** | **Arbeitsende/Completed** | **Leiter des Vorhabens/Project Leader** |
| 1. 9. 1976 | 31. 12. 1977 | Dr. Simon |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Completed | 31. 12. 1977 | 176.080,-- DM |

1. General Aim

The project consists of investigations of the effect of higher fluid velocities at the pump outlet on the vibration behaviour of RPV internals in the BWR core inlet plenum, in which the advanced internal BWR coolant recirculation pump has been installed.

2. Particular Objectives

Protection by experimental means of the currently constructed control rod guide tubes and in-core instrumentation tubes considering their vibrational behaviour.

3. Research Program

3.1 Overall Program
3.2 Pre-tests on a 1 : 5 scale model
3.3 Preparation for tests
3.4 Vibration measurements at the Karlstein test stand
3.5 Evaluation of tests
3.6 Measurements in a 1300 MW standard plant

4. Experimental Facilities

Pre-tests on a BWR flow model (1 : 5 scale) will be performed in order to make sure that representative flow fields are present in the Karlstein test stand. It is planned to investigate in Karlstein 11 original control rod guide tubes including their supports and with dummy control rods, and 4 in-core instrumentation tubes, in the most adverse position with respect to vibrational effects, which are the 1st and 2nd control rod guide tube rows.

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

Pre-tests have been run to assure that the deviation of the maximum velocity measured in the full-scale model will be less than 10 % compared with the 1 : 5 scale model.

The velocity distribution was measured with parallel operation of all pumps. With two prepared control rod guide tubes the pressure fluctuation was measured around the control guide tubes.

After modification the flow model, simulating a 36 $^{o}$ pump section, flow distributions were measured.

6. <u>Results</u>

The tests showed that the by-pass line, which was used for modification of the flow distribution to those of the full-scale model, had no influence on the flow distribution behind the core barrel. Therefore the heat-wire measurements were carried out with opened by-pass.

The evaluation of the flow distribution and pressure fluctuation measurements showed that in the region of the 1st control rod line no difference was found between full-model and segmented model. For the 2nd and 3rd line the results were not comparable as the lower part was changed in the segmented model.

7. <u>Next Steps</u>

The work has been completed, the final report will be written.

8. <u>Relation with Other Projects</u>

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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. 11. - 31. 12. 1977 | 10.4 | RS 297 |

| Vorhaben/Project Title | |
|---|---|
| Schallsignale in Kreiselpumpen bei 2-Phasenströmung<br><br>Sound Signals from Centrifugal Pumps Working in Different Flow-Regimes | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Technische Universität Berlin<br>Institut für Kerntechnik |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1. November 1977 | 31. Oktober 1980 | Prof.Dr.-Ing.U.Wesser |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | Dec. 31, 1977 | DM 337.940,-- |

## 1. General Aim

The purpose of this project is the development and test of a flow-regime detection method for centrifugal pumps. The investigations are part of experiments in the field of loss-of-coolant accidents in nuclear reactor power plants.

## 2. Particular Objectives

Primary investigations are carried out at Institut für Kerntechnik (IKT), TU Berlin, to test if characteristic quantities of sound will be significant influenced by different flow-regimes in centrifugal pumps. LOBI-pump-tests as WCL-Hamilton-Canada yield sound signal records corresponding to the later blown-down data of project RS 109 and for developing the measuring system. Sound signals detected from centrifugal pumps during test runs of project RS 109 at Ispra are analysed to identify different flow-regimes.

## 3. Research Program

There are three research activities:

3.1 Primary investigations with pump test facility at IKT

3.2 Data collection during LOBI-pump-tests at WCL-Hamilton-Canada and development of measuring system

3.3 Analysis of blow-down-test records of project RS 109 at Ispra.

## 4. Experimental Facilities, Computer Codes

The blow-down facility at IKT for primary investigations contents mainly pressure vessel and blow-down pipe with centrifugal pump. Test fluid is destillied water degassed by boiling. There are two pump operation modes, operation in forward and in reverse direction.

The fluid signals in the audible and in the ultrasonic range of frequencies are detected with impact sound and with air transducers located at different points of the pump.

5. Progress to Date

5. 1 Primary investigations with pump test facility at IKT. Data of twelf pump test runs at IKT are recorded and analysed. Experimental parameters - fluid temperature and pressure and the flow-regime - are compared with Power-Density-Spectra.

5.2 ---

5.3 ---

6. Results

The signals of each sound transducer, in the audible and in the ultrasonic range of frequencies, depend on the experimental period of blow-down test run and therefore on the prevailing conditions of flow-regime. Test run periods with high void fraction gave rise to great magnitudes of Power-Spectral-Density. Also increasing fluid temperature and pressure tended to augment Power-Spectral-Density for all flow-regimes, see Fig. 1. The evaluation time for each experimental point corresponded with the time necessary to analyse blow-down experiments.

7. Next Step

The research program will be continued with the measurement of flow-regime sound signal during LOBI-pump-tests at WCL-Hamilton-Canada.

8. Relations with Other Projects

Cooperation exist with project RS 284 "Kavitationssignale". Primary investigations are carried out in the blow-down facility of RS 135. Final aim is the analysis of data at the blow-down loop of Euratom Ispra, project RS 109.

9. References

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10. Degree of Availability of the Reports

-

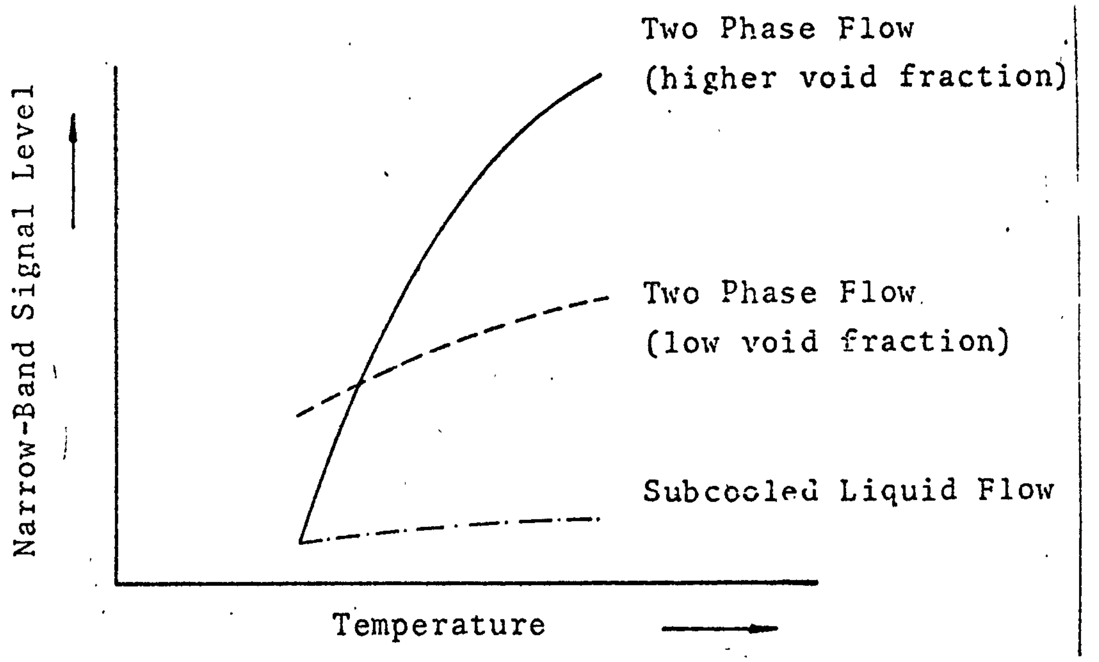


Figure 1: Narrow-Band Signal Level as Function of Fluid Temperature

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.11.1977 - 31.12.77 | 10.4 | RS 284 |
| **Vorhaben/Project Title** Statistische Analyse von kavitationsspezifischen Schallsignalen aus Notkühlpumpen<br><br>Statitical analysis of cavitation sound from emergency cooling pumps | | **Land/Country** FRG<br>**Fördernde Institution/Sponsor** BMFT<br>**Auftragnehmer/Contractor** Technische Universität Berlin<br>Institut für Kerntechnik |
| **Arbeitsbeginn/Initiated** 1.November 1977 | **Arbeitsende/Completed** 31.Oktober 1980 | **Leiter des Vorhabens/Project Leader** Prof.Dr.-Ing.U.Wesser |
| **Stand der Arbeiten/Status** continuing | **Berichtsdatum/Last Updating** Dec. 31, 1977 | **Bewilligte Mittel/Funds** 439.180.-- DM |

## 1. General Aim

The purpose of this projects is to determine the variation of statistical signal quantities with the onset and the development of cavitation in emergency cooling pumps. High disturbing noise level and the special conditions of nuclear reactors are operating conditions for the measurement system to be applied.

## 2. Particular Objectives

Theoretical work will be done to analyse the phenomena of sound generation from cavitation zones in pumps. Sound transducers already used in cavitation tests will be optimised with regard to the different measurement applications of the reserarch project. Computing routines for the statistical analysis are adjusted to the characteristic features of cavitation signals. Experimental work leads to measuring devices appropriate to detect cavitation sound signals from emergency cooling pumps. Cavitation tests in experimental pump facilities yield records suitable to select and determine characteristics quantities of cavitation signals. Experiments with original-sized emergency cooling pumps working in different operation modes are projected to test and to develope the methods of signal detection and evaluation.

## 3. Research Program

There are five main research activities:

3.1 Measurements of cavitation sound signals from original-sized pumps (especially emergency cooling pumps)

3.2 Measurements of cavitation sound signals from laboratory test facilities

7056

3.3 Development of sound detector systems

3.4 Development of signal evaluation methods

3.5 Work on basic problems of cavitation sound phenomena

## 4. Experimental Facilities, Computer Codes

Cavitation measurements are carried out in experimental facilities at the Institut für Kerntechnik (IKT), TU Berlin, and - to investigate large pumps and original-sized emergency cooling pumps - in test facilities of industry. The cavitation test equipment at IKT include a radial pump. The speed of the pump can be varyied. The fluid is destilled water. By means of a controllable heater and a cooler nearly stationary working conditions at different temperature levels can be maintained.

## 5. Progress to Date

Discussions with industry were carried out to prepare pump investigations in cavitation test facilities. Impact sound transducers were constructed, builded up and tested.

## 6. Results

Test of various types of sound transducers also under conditions of primary circulating pumps showed that some types are especially qualified to separate cavitation sound signals from the disturbing noise level and to reduce the electronic equipment.

## 7. Next Steps

The experimental work will be continued with cavitation sound measurements from emergency cooling pumps under simulated accident conditions.

## 8. Relations with Other Projects

Cooperations exist with project RS 259 "NaK-Testloop" and project RS 297 "Sound Signals from Centrifugal Pumps Working in Different Flow-Regimes".

## 9. References

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## 10. Degree of Availability of the Reports

-

1057

[illegible]-1- 06 / 159-1- 04 /4165-10

5.I
X 10.4

| | |
|---|---|
| Titre :<br>TRANSFERT DE LA CONTAMINATION DANS LES REACTEURS EN SERVICE | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br>TRANSFERRING OF CONTAMINATION IN OPERATING REACTOR | Organisme exécuteur<br>CEA/DSN/SESTR/CADARACHE<br><br>Responsable<br>DSN/SESTR |

| | | |
|---|---|---|
| Date de démarrage 01/09/73 | Etat actuel EN COURS | Scientifiques |
| Date prévue d'achèvement 1981 | Dernière mise à jour 30/11/77 | |

| TITLE 1 (original language)<br>Comportamento vibrazionale di componenti del nocciolo | Classification<br>10.4 |
|---|---|
| TITLE 2 (english)<br>Vibration Behaviour of Core Components | Country: ITALY<br>Sponsor: CNR - CNEN<br>Organisation: University of Pisa |
| Date initiated 1973<br>Date completed 1976<br>Last updating June 1976 | Project Leader<br>E. Manfredi |

Description :

Research program :

To study, either by experiments and through a theoretical approach, the vibration behaviour of core components when subjected to fluid flow. The fuel element vibration behaviour has been the first objective of this research.

Facilities :

- Water test loop which is able to carry out experiment with test sections up to 3 meters lenght and 200 mm outer diameter with flow velocities up to 10 m/sec
- vibration transducers and flow measurement instruments
- signal_processing instrumentation.
- computers IBM 370 of CNUCE and IBM 1130 for the analysis of test data and for the development of theoretical studies.

Reference documents :

1. C. Carmignani, E. Manfredi, S. Reale
Influence of assembly configurations on the flow induced vibrations of a BWR fuel element.
3rd Smirt Conference - London 1975.

| Title 1 (Original language) | Classification |
|---|---|
| Comportamento vibrazionale della colonna di combustibile Cirene - Fretting corrosion. | 10.4. HWR |
| **Title 2 (English)** | Country: ITALY |
| Dynamics of fuel strings in axial flow - Fretting corrosion. (HWR - Calandria tubes) | Sponsor: CNEN<br>Organisation: CNEN-CISE |
| Date initiated: 1974<br>Date completed: In progress<br>Last updating: March 1977 | Project Leader<br>G. LELLI |

1. General aim

Optimization of fuel bundles for HWRs in order to minimize fretting corrosion damages.

2. Particular objectives

Reduction of the vibrational amplitude of fuel strings under coolant flow.

3. Experimental facilities and programme

The experimental facility consists of a full scale adiabatic loop. The experimental programme consists in the analysis of the vibrational behaviour of fuel strings and of the fretting corrosion marks both on the bundles and on the pressure tubes. The experimental results will be useful also to qualify mathematical models developed in order to simulate the fuel strings behaviour.

4. Project status

A few preliminary results show a good vibrational behaviour of the fuel string and few minor fretting-corrosion marks. The above mentioned encouraging goal is expecially due to the improvements obtained in the assembling of the fuel bundles.

5. Next steps

One more test in the adiabatic loop will be carried out.
In pile experiments are planned.

6. Relation to other projects

Optimization of mechanical design of fuel bundles for HWR.

7. Reference documents

- G. Lelli "Computer modelling of the dynamics of a string of fuel bundles in axial flow. Part B: response to a generalized forcing function and further developments". July 1975. CRNL-1339.

| Title 1 (Original language) | Classification |
|---|---|
| Comportamento vibrazionale della colonna di combustibile Cirene - Fretting corrosion | 10.4. HWR |

- G. Lelli "A comparison of experimental data, on the natural frequencies and mode shapes for Split-Spacer and LS-3 bundles, with the calculated data from an analytical model". Jan. 1975. Internal report.
- J. Fedoruk, G. Lelli "A comparison of Experimental Mobility data for Split Spacer and LS-3 Fuel bundles with calculated values from an analytical model". August 1975. Internal report.
- Z. Arcangeli, G. Colombo, G. Lelli, M. Vignolini "Misure comparate di deformazione a parallelogramma di fasci di combustibile Cirene". Aprile 1976. Internal report.
- Z. Arcangeli, G. Lelli "Studio del regime vibrazionale della colonna di combustibile Cirene. Frequenze proprie. Effetto della rigidezza". Giugno 1976. Internal report.
- G. Lelli, M. Vignolini "Verifica di un valore massimo della spinta esercitata dalla molla di compressione della colonna di combustibile Cirene. Considerazioni relative alla fretting corrosion ed allo stato di sollecitazione delle piastre di estremità dei fasci". Febbraio 1977. Internal report.

Degree of availability

To a limited extent.

| N.V. KEMA | CLASSIFICATION : 10.4 |
|---|---|
| TITLE : Ruisanalyse van de reactor en het primaire circuit | COUNTRY: THE NETHERLANDS |
| TITLE (ENGLISH LANGUAGE): Reactor and primary circuit noise analysis | SPONSOR : KEMA<br>ORGANIZATION : KEMA<br>PROJECTLEADER : |
| INITIATED : – / LAST UPDATING : 1978<br>STATUS : – / COMPLETED : 1977 | SCIENTISTS : J. Hoekstra, J. v.d. Veer, E. Türkcan |

General aim

On load surveillance of vital systems.

Particular objectives

Dynamic behaviour of a BWR.

Experimental facilities

Dodewaard nuclear power plant.

Project status

Still in progress.

Next steps

Not applicable.

Relation to other projects

Same investigation at Borssele nuclear power plant by E. Türkcan.

Reference documents

None.

Degree of availability

Through the organizations KEMA and ECN.

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX

1065

| Energieonderzoek Centrum Nederland | CLASSIFICATION : 8.10.4 |
|---|---|
| TITLE : Ruisanalyse in vermogensreactoren met het oog op reactorbewakingsmogelijkheden. | COUNTRY: THE NETHERLANDS |
| | SPONSOR : ECN |
| | ORGANIZATION : ECN |
| TITLE (ENGLISH LANGUAGE): Noise analysis in power reactors for malfunction detection | PROJECTLEADER : E. Türkcan |
| INITIATED : 1974 — LAST UPDATING : June 1978 | SCIENTISTS : - |
| STATUS : in progress — COMPLETED : 1981 | |

# 11. MATERIALS AND MECHANICAL PROBLEMS IN NORMAL AND ACCIDENT CONDITIONS

1067

| | |
|---|---|
| | Classification 11.1 |
| Title 1<br><br>Fuel Element Behaviour under Irradiation | Country : Belgium<br><br>Sponsor : BELGONUCLEAIRE<br><br>Organisation : BELGONUCLEAIRE - CEN/SCK |
| Title 2 | |
| Initiated : 1972 Completed : -<br>Status : in progress Last updating : June 1978 | Project leader :<br>Mr. H. Bairiot |

1. General Aim

Assessment of fuel element behaviour under irradiation (including mixed oxide).

2. Particular Objectives

(a) Fuel densification study and evaluation of the qualitative and quantitative effects (e.g. neutronic point-of-view, physical mechanisms, attempt to deduce empirical correlations for predetermination calculations) ;

(b) Fuel behaviour under accidental conditions ;

(c) Power ramp capabilities ;

(d) Surveillance of statistical behaviour of Pu fuel in BWR and PWR power plants.

3. Experimental Facilities and Programme

- In-pile (BR-2 reactor) and out-of-pile tests for fuels and fuel elements. Neutron radiography. Post-irradiation examination. Pool side inspection.
- VENUS critical facility for study of local power peaking effects with simulated axial gaps.
- Fuel assemblies irradiated in BR-3, SENA, DODEWAARD, GARIGLIANO, etc ...

4. Project Status

Progress to date : VENUS tests, irradiations in BR-2, BR-3 and SENA. Routine operation of codes COMETHE (steady state) for fuel behaviour, SPARTAN and THEATRE 3 for transient evaluation of fissile material inhomogeneity.

Essential results : Power peaking effects, fuel-clad interaction, fission products.
COMETHE was selected as "the best code" in a comparative evaluation performed by EPRI (March 1977).

5. Next Steps

- COMETHE transient under development.
- Experimental data base continuously expanded.

6. Relation with other Projects

Some actions in the frame of the "Pu recycle programme" sponsored by the ECC.

7. Reference Documents

8. Degree of Availability : proprietary

- Disclosure on a bilateral basis to be agreed.

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1. - 31.12.1977 | 11.1 | PNS 4235.3 |

**Vorhaben/Project Title**

Untersuchungen zum Einfluss des Oxidbrennstoffes und von Spaltprodukten auf die mechanischen Eigenschaften von Zry-Hüllrohren bei Störfalltransienten

Investigations of the Influence of Oxide Fuel and Fission Products on the Mechanical Properties of Zry-cladding Tubes under Transient Conditions

| | |
|---|---|
| Land/Country | FRG |
| Fördernde Institution/Sponsor | BMFT |
| Auftragnehmer/Contractor | Kernforschungszentrum Karlsruhe (KfK)<br>Projekt Nukleare Sicherheit IMF I |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| Ende 1974 | 1980/81 | Dr. P. Hofmann |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | December 1977 | |

## 1. General Aim

Evaluation of the extent of inner corrosion and of the influence of chemical interactions between oxide fuel and fission products, respectively, on the one hand, and the Zry cladding material, on the other hand, on the mechanical properties of Zry cladding tubes at high temperatures.

## 2. Particular Objectives

- Investigation of the influence exerted by the oxygen potential of the $UO_2$ on the deformation and rupture behavior of Zry-4 tubing in loss-of-coolant accidents.
- Stress corrosion cracking behavior of Zry cladding tubes with respect to volatile fission products (Cs, I, Te, Cd).
- $UO_2$/Zry reaction experiments under Power Cooling Mismatch (PCM) conditions; determination of the reaction kinetics between $UO_2$ and Zry.

## 3. Research Program

3.1 Isothermal, isobaric as well as temperature and pressure transient experiments on short $UO_2$ or Argon containing Zry-claddings tube specimens under inert gas conditions.

3.2 Creep rupture as well as temperature and pressure transient tests under inert gas on short Zry tubular specimens containing little amounts of iodine or other volatile fission products. Influence of $UO_2$ and thin oxide layers on the stress corrosion behavior of as-received Zry cladding tubes. Examination of the internal tube surfaces of iodine containing burst specimens, using a scanning electron microscope (SEM).

3.3 Isothermal annealing tests performed under inert gas conditions on short Zry-tube specimens filled with $UO_2$ pellets at temperatures up to 1500°C and $UO_2$/Zry contact pressures up to 80 bar. X-ray and electron microprobe investigations of the reaction products formed during the chemical interactions between $UO_2$ and Zry.

## 4. Experimental Facilities

ad 3.1 The burst tests are performed in the new tube bursting apparatus TUBA. In addition to the temperature of the cladding tube and the inner gas pressure in the tube specimen, the cladding tube diametral expansion is determined continuously using a light optical method (high-speed camera).

ad 3.2 Tests on stress corrosion cracking (SCC) are mainly carried out in the SCC apparatus, part of them also in the TUBA facility.

ad 3.3 The $UO_2$/Zry reaction experiments are performed in a high-pressure gas autoclave system under inert gas conditions.

## 5. Progress to Date

- Investigations of the influence exerted by the oxygen potential of oxide fuel on the deformation and rupture behavior of short Zry-4 cladding tubes.
- Performance of temperature and pressure transient and isothermal, isobaric experiments with as-received Zry cladding tubes in order to study the stress corrosion cracking behavior with respect to elemental iodine.
- Preliminary investigations into the influence exerted by the $UO_2$ oxygen potential and by thin oxide layers present on the cladding tube inner surface on the SCC behavior of Zry-4 with respect to iodine.
- Performance of $UO_2$/Zry reaction experiments under PCM conditions. Determination of the $UO_2$/Zry reaction kinetics. Influence of simulated fission products and thin $ZrO_2$ layers on the internal cladding tube surface on the $UO_2$/Zry chemical interactions and on reaction kinetics. Experiments for determination of the initial $UO_2$/Zry phase boundary after reaction annealings.
- Investigations of $UO_2$ and Zry based on analytical chemistry. Quantitative determination of oxygen distribution across the Zry-4 cladding tube section.

## 6. Results

- Under LOCA typical test conditions the $UO_2$ oxygen potential exerts practically no influence on the burst temperature and the burst pressure; there exists only a small influence on the deformation behavior and burst strain of the Zry-4 cladding tubes. The circumferential burst strains of the Zry-tubes vary between 20 and 130%. In accordance with the literature, a minimum in the burst strain exists at about 930°C.

- The first burst experiments with Zry tube specimens containing iodine under temperature and pressure transient test conditions clearly show that the cladding material can fail as a result of stress corrosion cracking. At burst temperature ≤850°C the maximum circumferential strains and rupture stress of iodine containing specimens are distinctly smaller than that of Ar-reference specimens. Above 850°C the influence exerted by iodine on the deformation behavior of Zry-4 decreases markedly.
- The creep rupture tests on the stress corrosion cracking behavior of Zry cladding tubes with respect to iodine show that for all temperatures tested (700 - 1000°C) and endurance times (<10 min) the time until burst is markedly shorter for specimens containing iodine as compared with specimens containing no iodine. However, with the endurance times getting longer (>10 min) and inner pressures and cladding tube tangential stresses, respectively, decreasing accordingly, the influence of iodine on the time until rupture becomes clearly smaller. As to burst strains, the results already obtained from transient experiments are confirmed; below 900°C the burst strains of tubular specimens containing iodine are markedly lower than that of comparison specimens without iodine.
- First experiments with Zry-4 tubular specimens either subjected to preliminary inner oxidation (≤25 µm) or filled with $UO_{2+x}$ powder show, especially at temperatures ≥900°C, a distingtly higher strength due to the uptake of oxygen, as compared with reference specimens. This means that, compared with the tubular specimens only filled with argon and not subjected to preliminary oxydation, the endurance times of the said tubular specimens are substantially longer. However, in the presence of iodine, the influence exerted by oxygen strongly decreases, i.e. the endurance time of cladding tubes are again greatly reduced by stress corrosion cracking.
- The examination of the fracture surfaces by scanning electron microscopy show that the fracture has nucleated at the surface exposed to the corrosive environment. There are two well defined types of fracture features, namely intergranular cracking, and ductile shear. As cracks propagate through the tube wall the principle mode of failure changes from intergranular cleavage to ductile dimpling. This change of fracture mode occures because the thickness of the remaining tube wall decreases but still has to be sustain the load. In the absence of iodine the fractography of tubes showes an entirely ductile mode of failure.

- The $UO_2$/Zry-reaction experiments performed under PCM-conditions show that under good $UO_2$/Zry-contact conditions $UO_2$ is reduced above 900°C by Zry. X-ray diffractometer studies of the $UO_2$/Zry interaction zones yield the formation of α-Zr(O), α-uranium, and a (U,Zr) alloy, part of which still containing some $UO_2$ finally dispersed. The oxygen content in the different phases was evaluated quantitatively by means of Auger-Electron-Spectroscopy (AES) and electron microprobe analysis.
- Obviously, simulated fission products (Cs, I, Te) do not exert an influence on the chemical interactions between $UO_2$ and Zry. Part of the Te is detected together with Cs at the reaction front in Zry. Thin $ZrO_2$ layers on the inner surface of the cladding tube are able to prevent $UO_2$/Zry reactions up to 1000°C.
- In the $UO_2$/Zry chemical interactions under PCM conditions the individual reaction zones grow between 1000 and 1400°C according to a parabolic time law. Oxygen diffusion into Zry is the rate determining step of the $UO_2$/Zry-reaction.
- Experiments with tungsten markers (sheets, wires) provided between $UO_2$ and Zry for the determination of the initial $UO_2$/Zry phase boundary after reaction annealings have shown that the α-Zr(O) phase adjacent to $UO_2$ penetrates into $UO_2$ as a restult of Zry reacting with $UO_2$. The initial $UO_2$-Zry phase boundary lies within the range of the (U,Zr) alloy following the α-Zr(O) phase.
- Investigations based on analytical chemistry (AES) of the inner cladding surface of $UO_2$ containing Zry tube specimens show that much oxygen is absorbed by the cladding tube surface (up to about 25 wt.%). Besides on the O/U ratio, the temperature, and the time until burst, the maximum oxygen content depends heavily on the fuel surface.

## 7. Next Steps

- Continuance of isothermal and transient tests on stress corrosion cracking of Zry cladding tubes due to iodine and other volatile fission products. Determination of the critical iodine concentrations. Influence of thin oxide layers on the inner cladding tube surface on the critical iodine concentration.
- Stress corrosion cracking experiments on pre-cracked Zry cladding tubes (sharp edges, crack due to fatigue). Examinations of tubular specimens by the scanning electron microscope prior to and after the tests relative to stress corrosion cracking.

- SEM investigations of ruptured Zry-4 tube specimens; determination of the mechanisms leading to cladding tube failure.
- Post-irradiation examination of irradiated fuel rods belonging to PNS task 4237.
- $UO_2$/Zry reaction experiments under PCM conditions. Influence of simulated fission products on the $UO_2$/Zry-reaction kinetics.
- Investigations of $UO_2$ and Zry based on analytical chemistry.

## 8. Relation with Other Projects

PNS 4235.1, 4235.2, 4235.4 and 4237.

## 9. Literature

/1/ PNS 2.Semi-annual Report 1976, KFK 2435 (1977), p. 306

/2/ PNS 1.Semi-annual Report 1977, KFK 2500 (1978)

/3/ P.Hofmann, C.Politis
Investigations of the influence of oxide fuel and fission products on the mechanical properties of Zry-4 cladding tubes under transient conditions.
Thermal Reactor Safety Meeting of the ANS, August 1977.

/4/ P.Hofmann
Über die mechanische Beanspruchung von Zr, Zry und anderen Werkstoffen durch die Bildung von Oxidschichten (literature review)
KFK Ext. 6/77-2 (1977).

## 10. Degree of Availability

Unrestricted distribution.

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 01.01.-31.12.1977 | 11.1 | PNS 4235.4 |

| Vorhaben/Project Title | |
|---|---|
| Berstversuche an Zircaloy-Hüllrohren unter kombinierter mechanisch-chemischer Beanspruchung<br><br>Burst-tests of Zircaloy Cladding Tubes under Mechanical and Chemical Load | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS) IMF |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1977 | 1980 | Dr. S.Leistikow/L. Schmidt |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | December 1977 | |

## 1. General Aim

Cladding Material Behaviour under Accident Conditions.

## 2. Particular Objectives

Investigation of the High Temperature Properties of Zircaloy 4 under combined mechanical and chemical load (inner pressure, oxide fuel, fission products, steam)

## 3. Research Program

Burst-tests of indirectly heated Zircaloy Tubes under different accident conditions.

## 4. Experimental Facilities

Facility for Burst Testing under Fuel Interaction, Steam Oxidation, and LOCA Conditions (FABIOLA).

## 5. Progress to Date

Besides single effect experiments burst test are going to be prepared for a more realistic description of Zircaloy cladding behaviour under LOCA-typical temperature- and pressure-transient conditions and fuel and coolant interaction. An experimental test facility is designed in which pressurized fuel rod simulators will be electrically indirectly heated in steam. The main material properties to be measured as function of temperature and time will be the maximum circumferential elongation, the time-to-rupture, and the extent of external and internal oxidation. At present the concept for its layout is completed and the main components are ordered.

## 6. Results

Pretests for the measurement of the Zircaloy cladding tube surface temperature using special pyrometers, were performed in the temperature range between 500 and 800°C. These pretests proved that under the given conditions the accuracy of the pyrometers is equal to those of thermocouples in case that an uniform stable surface oxide layer was preformed in steam. Thus, as an important item of testing procedure, the measured maximum circum-ferential elongation and time-to-rupture can be correlated to the effective and continiously recorded temperature and pressure transients. Equally, the results of post-test evaluation on external and internal oxidation will be related to the chosen operating conditions.

## 7. Next steps

Supply of all necessary components and start of facility construction. Pretests of prototype heaters. Specification and order of an X-ray cinematographic device for continious measurement tube expansion.

## 8. Relations to other programs

All others of KfK/PNS, KWU, and USNRC.

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. 77 - 31. 12. 77 | 11.1 | RS 203 |
| Vorhaben/Project Title<br>KFA/KWU-Leistungsrampen Testbrennstabbestrahlungen 1976/1977<br><br>KFA/KWU-Power Ramps Fuel Rod Irradiation Tests 1976/1977 | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>KRAFTWERK UNION AG Reaktortechnik<br>RB 31, Erlangen |
| Arbeitsbeginn/Initiated<br>1. 4. 76 | Arbeitsende/Completed<br>31. 12. 80 | Leiter des Vorhabens/Project Leader<br>H. Knaab |
| Stand der Arbeiten/Status<br>Continuing | Berichtsdatum/Last Updating<br>31. 12. 77 | Bewilligte Mittel/Funds<br>1'554.000,-- DM |

1. General Aim

Power ramp experiments in HFR Petten are being carried out with PWR and BWR standard fuel rods of different production parameters, in order to investigate the reason for fuel rod defects, especially the fuel-clad interaction.

2. Particular Objectives

The mechanism causing fuel rod defects by alternating power and positive power ramps is not yet well known. In practice, fuel rod defects have been discovered after local power increases in power reactors and in power ramp tests in Halden, Studsvik and Risø.

In order to study this problem, the irradiation and load change behaviour of PWR and BWR fuel rods with oxide fuel ($UO_2$) is being investigated in HFR Petten. The fuel rods, assembled to segmented rods, have been or are still being preirradiated in the Obrigheim and Würgassen power stations up to 5000 - 24000 MWd/t(U) burn-up.

3. Research Program

3.1 Unloading from reactor and sectioning of segmented PWR and BWR fuel rods, intermediate inspection of the fuel rod segments/test fuel rods.

3.2 Neutronradiographic investigations on the test fuel rods.

3.3 Installation of the test fuel rods into the irradiation capsules and performance of power ramp experiments in HFR Petten.

3.4 Post-irradiation examination of the test fuel rods.

3.5 Intermediate evaluation of the power ramp experiments and report of the results.

4. Experimental Facilities

The test fuel rods have been or are still being preirradiated in the Obrigheim and Würgassen power stations for 1-3 reactor cycles. After preirradiation the test fuel rods are inserted into test capsules and are ramped under representative LWR conditions in the pool-side-facility of HFR Petten. The fuel rods are ramped to linear heat generation rates of $q' \leq 650$ W/cm with ramp rates of $\dot{q}' \leq 100$ W/cm min and heat rate increases of $\Delta q' \leq 350$ W/cm. It is planned to equip the test capsules with inductive elongation detectors in order to measure changes of the fuel rod length during ramping. The tests are performed as a joint project of KFA Jülich and Kraftwerk Union.

The following parameters will be varied:

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|---|---|
| fuel: | $UO_2$ of different structures, density, pellet geometry |
| cladding tube: | production parameters (e.g. graphite on inner surfaces) |
| fuel rod: | pressurization, burn-up |

5. Progress to Date

Altogether, 33 in-situ and start-up ramp experiments with PWR fuel rods have been carried out in HFR Petten until the end of 1977. The fuel rods had been preirradiated in the Obrigheim nuclear power station at approx. 210 W/cm linear heat generation rate (LHGR) to burn-ups of up to

12000 MWd/t(U) - one reactor cycle - and
22000 MWd/t(U) - two reactor cycles - respectively.

With the in-situ ramp experiments, the LHGR of the fuel rods have been increased to maximum power levels of 420 W/cm to 650 W/cm with a ramp rate of approx. 100 W/cm min after a preconditioning irradiation of 400 h at about 300 W/cm. With the start-up experiments the fuel rods have been ramped to a maximum power level of approx. 540 W/cm with ramp rates of 0,25 W/cm min to 100 W/cm min without any preconditioning.

6. Results

The first results of the in-situ ramp experiments indicate a high failure probability of one-cycle fuel rods at ramp LHGRs in the range of 580 W/cm. However the failure probability decreases both at higher and lower ramp powers. At ramp LHGRs falling below 470 W/cm, no defective one-cycle rods have been observed. Start-up experiments have shown, that ramp rates of 10 W/cm min and above do not seem to influence considerably the defect behaviour of one-cycle rods. Ramp rates falling below 5 W/cm min did not cause any fuel rod failures.

In-situ ramp experiments with two-cycle fuel rods have indicated an upper limit for defect free operation at unrestricted reactor operation in the range of 420 W/cm. Above this boundary, a high failure probability has existed and the time to defection has become increasingly shorter with increasing LHGR levels.

First nondestructive testing has been performed in the Petten reactor pool and in the hot cells and have resulted in the following preliminary observations.

a) The defects are located mainly at pellet interfaces along the high power section of the rods;

b) Most of the defect rods showed several defect positions; the defects appeared pin-hole like;

c) Not all defect positions detected by eddy-current testing could be found visually;

d) Not all defect rods showed intrusion of water by neutronradiography;

e) Fission product peaking was detected at pellet interfaces and at pellet cracks;

f) Slight ridges were found along the high power section of the fuel rods, but the correlation of ridge height and local LHGR is not very conclusive. Also non-defective rods showed ridges.

g) Neutronradiography revealed a change of the crack pattern to many mid-pellet through-cracks, and a fast partial closing of the dishes.

7. Next Steps

Continuation of the ramp program

- to determine the LHGR limit for defect free operation of LWR fuel rods (improved and modified PWR fuel rods, standard BWR fuel rods) at unrestricted reactor operation by in-situ ramp experiments

- to determine the conditions for defect free operation at LHGRs above that limit and eventually restricted reactor operation by modified in-situ ramp experiments (reduced ramp rate at LHGRs exceeding 470 W/cm - one cycle fuel rods - and 420 W/cm - two cycle fuel rods - respectively)

8. Relation with Other Projects

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9. References

R. Holzer, D. Knödler, H. Stehle
"Pellet Clad Interaction: Experience, Testing and Evaluation". A KWU Review ANS Topical Meeting on

Water Reactor Fuel Performance, May 9-11, 1977, St. Charles, Illinois

10 Degree of Availability

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Classification: 11.1

| Title 1<br>Ramp Testing of $UO_2$-Zr Fuel | COUNTRY: Denmark<br>SPONSOR: Risø National Laboratory<br>ORGANIZATION: Risø National Laboratory |
|---|---|
| Title 2<br>Ramp Testing of $UO_2$-Zr Fuel | Project Leader:<br>P. Knudsen |
| Initiated: 1972    Completed:<br>Status: Progressing    Last Updating: | Scientists: |

1. General Aim

To examine the performance of $UO_2$-Zr fuel pins during overpower ramps.

2. Particular Objectives

To submit BWR- and PWR-type $UO_2$-Zr fuel pins of medium-to-high burnup to power increases mainly simulating nominally normal operating conditions in power reactors.

3. Experimental Facilities and Programme

DR 3 Reactor at Risø with associated loops.

4. Project Status

4.1. Progress to date: BWR- and PWR-type fuel pins have been tested, with burnups up to 35,000 MWD/t $UO_2$.

4.2. Fuel pin failures indicate the existence of limiting combinations of design and operating conditions.

5. Next Steps

Continue tests to support and extend present experience.

6. Relation with Other Projects

Results will be used for verification of fuel model calculations.

7. Reference Documents

(a) P. Knudsen, H.H.Hagen, J. Stiff, Atomwirtschaft 3 (1974) 135-136.
(b) P. Knudsen, K. Bryndum, Trans. ANS Winter Meeting 1974, p. 140.

8. Degree of Availability

Not generally available.

**Classification: 11.1**

| | |
|---|---|
| Title 1<br>Fuel Modelling | COUNTRY: Denmark<br>SPONSOR: Risø National Lab.<br>ORGANIZATION: Risø National Laboratory |
| Title 2<br>Fuel Modelling | Project Leader:<br>N. Kjær-Pedersen |
| Initiated: 1972    Completed:<br>Status: Progressing    Last Updating: | Scientists: |

1. General Aim

To develop and maintain an up-to-date computer model of fuel pin performance under nominally normal operating conditions.

2. Particular Objectives

The code considers one pellet-size segment of a fuel rod. A finite difference model of the pellet is coupled with a thin shell model of the cladding. Pellet cracking is directly accounted for, based on certain assumptions relating to crack patterns. Secondary and primary creep of fuel and clad are represented. The model aims at an analysis of local stresses and strains in the cladding, including ridge formation, as well as fission gas release, to form a basis for a failure probability estimation.

3. Experimental Facilities and Programme

Danish ramp testing programme.

4. Project Status

4.1. Progress to date: The second version of the model is in full production.

4.2. Essential results: Proven capability of predicting measured ridge-heights within a factor of two (ref. 2).

5. Next Steps

Develop next version, which will permit several axial nodes to be considered.

6. Relation with Other Projects

Utilizes data from Danish overpower test programme for verification.

7. Reference Documents


1. N. Kjær-Pedersen, "A New Version of the LWR Fuel Performance Model WAFER", 4th International Conference on Structural Mechanics in Reactor Technology, San Francisco, U.S.A., 15-19 August 1977, Paper No. D 1/3.

2. N. Kjær-Pedersen, " WAFER-2. A Code for Thermal and Mechanical LWR Fuel Performance Modelling", IAEA Specialists Meeting on Fuel Element Performance Computer Modelling, Blackpool, UK, 13-17 March 1978.

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| 140-1 -01<br>145-2 -01 | | 11-1 |
|---|---|---|
| Titre<br>Développement de codes de calcul de sûreté pour les éléments combustibles des réacteurs à eau ordinaire en fonctionnement normal et en régime accidentel. | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DgCS |
| Titre (anglais)<br>Development of safety codes for pressurized water reactor fuel elements under normal operation and accidental transient conditions. | | Organisme exécuteur<br>CEA/DMECN/DTech<br>SACLAY<br><br>Responsable<br>DTech - Saclay |
| Date de démarrage 1/1/77 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 31/12/80 | Dernière mise à jour 12/77<br>(mise à jour n°1) | |

1 - Objectif général :

Disposer, à terme, de codes de calcul pour les éléments combustibles des réacteurs à eau ordinaire sous pression rendant compte de leur comportement mécanique et thermique tant en fonctionnement normal qu'au cours des différents accidents envisagés.

2 - Objectifs particuliers :

1 - Juger de l'importance relative des différents phénomènes mis en jeu, et rechercher les modèles les plus aptes à les représenter. S'ils font défaut, les établir à partir de l'expérience acquise par le CEA.

2 - Tenir à jour et perfectionner le code de fonctionnement normal (RESTA) en vue de décrire les états de référence et de préciser les limites de sûreté.

3 - Développer les codes décrivant l'accident de dépressurisation (CUPIDON et DEMETER).

4 - Ultérieurement, développer un code décrivant l'accident d'insertion de réactivité (TRANSIT).

.../

4 - Etat de l'étude :

1 - Avancement à ce jour :

Un modèle de déformation de gaine au cours d'un accident de dépressurisation a été établi à partir des résultats du programme EDGAR.
Le code de fonctionnement normal, RESTA, est considéré comme opérationnel, mais doit faire l'objet de perfectionnements.
Un code relatif à l'accident de dépressurisation dans le cas particulier du programme PHEBUS, CUPIDON, a été écrit. Ce code est considéré comme opérationnel, mais une deuxième version, plus élaborée, doit lui être substituée.

5 - Prochaines étapes :

Modifier le modèle de déformation de gaine compte tenu de nouveaux résultats expérimentaux attendus.
Perfectionner le modèle de fluage et de relaxation de la gaine dans le code RESTA.
Etablir la deuxième version du code CUPIDON.
Développer le code DEMETER en utilisant l'acquis du code CUPIDON, mais en l'étendant au cas général portant sur un assemblage normal irradié.

6 - Relation avec d'autre études :

Ce programme est en relation, d'une part avec le programme EDGAR ZY qui fournit les données nécessaires à l'établissement d'un modèle de déformation de la gaine, et d'autre part avec le programme PHEBUS. Le code CUPIDON a été établi en premier lieu pour effectuer les calculs préliminaires de ce programme. Les résultats de celui-ci seront utilisés pour qualifier le code DEMETER.

7 - Documents de référence :

Rapports internes non disponibles .

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| | | |
|---|---|---|
| 144-1 -01 /4105-20<br>154-1 -01 | | 14<br>* 11.1 |
| Titre<br>Disponibilité de systèmes en attente périodiquement testés.<br>Approche analytique. | | Pays<br>FRANCE |
| | | Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br>Periodically tested standby systems availability<br>Analytical approach. | | Organisme exécuteur<br>CEA/DSN - SETSSR |
| | | Responsable<br>DSN-SETS - Fontenay |
| Date de démarrage 01/12/75 | Etat actuel<br>en suspens | Scientifiques |
| Date prévue d'achèvement 1978 | Dernière mise à jour 12/77 | |

| | |
|---|---|
| 144-1 -03<br>154-1 -03 | 14<br>x 11.1 |

| | |
|---|---|
| Titre<br><br>Etude de processus stochastiques<br>(processus de Markov non homogènes) | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br><br>Study of stochastic processes<br>(non homogeneous Markov Processes) | Organisme exécuteur<br>CEA/DSN - SETSSR<br><br>Responsable<br>-<br>DSN/SETS Fontenay |

| | | |
|---|---|---|
| Date de démarrage 1/1/78 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 31/12/79 | Dernière mise à jour 1.1.78 | |

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<table>
<tr><td colspan="2">144-1 -04 / 154-1 -04 /4112-20</td><td>14<br>x 11.1</td></tr>
<tr><td colspan="2">Titre<br><br>Utilisation des techniques de MONTE CARLO pour les calculs de fiabilité.</td><td>Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DSN</td></tr>
<tr><td colspan="2">Titre (anglais)<br><br>The use of MONTE CARLO techniques for reliability calculations.</td><td>Organisme exécuteur<br>CEA/DRE - SERMA<br><br>Responsable</td></tr>
<tr><td>Date de démarrage 01/01/77</td><td>Etat actuel<br>en cours</td><td rowspan="2">Scientifiques<br>--</td></tr>
<tr><td>Date prévue d'achèvement 31/12/78</td><td>Dernière mise à jour 12/77</td></tr>
</table>

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| 144-2 -01 / 154-2 -01 /4600-50 | | 14 * II.1 |
|---|---|---|
| Titre<br><br>Paramètres humains de la sûreté | | Pays<br>FRANCE<br><br>Organisme directeur<br>C.E.A/ DgCS |
| Titre (anglais)<br><br>Human factors in nuclear safety | | Organisme exécuteur<br>CEA/DSN - SETSSR<br><br>Responsable<br>DSN/SETS /Fontenay |
| Date de démarrage 1/01/78 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 31/12/81 | Dernière mise à jour 1/78 | |

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| | |
|---|---|
| 144-1 -09 / 154-1 -09 /4105-20 | 14 * 11.1 |

| | |
|---|---|
| Titre<br>Construction automatique d'arbres de défaillances à partir de diagrammes logiques. | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br>Automatic construction of fault trees from logic diagrams. | Organisme exécuteur<br>CISI<br><br>Responsable<br>DSN/SETS Fontenay |

| | | |
|---|---|---|
| .te de .emarrage 1/6/77 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 31/12/79 | Dernière mise à jour 1/78 | |

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| 144-1-07<br>154-1-07<br>164-1-02 | 14<br>* 11.1 |
|---|---|

| | |
|---|---|
| **Titre**<br>Modèlisation du taux de défaillance de vannes en fonction de différents paramètres. | **Pays**<br>FRANCE<br>**Organisme directeur**<br>CEA / DSN |
| **Titre (anglais)**<br>Modelisation of the failure rate of valves in terms of different parameters. | **Organisme exécuteur**<br>CEA/DSN - SETSSR<br>**Responsable**<br>DSN/SETS- Fontenay |

| | | |
|---|---|---|
| **Date de démarrage** 1/12/77 | **Etat actuel** en cours | **Scientifiques** |
| **Date prévue d'achèvement** 30/6/78 | **Dernière mise à jour** 1/78 | |

Classification 11.1

| | |
|---|---|
| Title 1<br>OVERPOWER TESTS | Country<br>Italy |
| Title 2 | Sponsor<br>CNEN |
| Initiated 1.3.1974 Completed sept. 75<br>Status Progressing Last updating End 74 | Organisation<br>CNEN<br>Project Leaders<br>C. Lepscky |

1. General aim
   Fuel cladding interaction and mechanical properties of irradiated cladding after rupture determined by power excursion. -

2. Particular objectives
   Investigate modes of rupture and material properties by neutron-radiography dimensional analysis etc. of irradiated cladding subjected to increasing power ramps up to power burst.
   Investigate the influence of gap (150, 230, 310 cold gap).

3. Experimental facilities and programme
   Irradiation(Halden reactor)and post irradiation examination at CSN Casaccia Center of CNEN

4. Project status
   4.1. Progress to date
   Irradiation completed at Halden (IFA 131) up to 30.000 MWD/t.
   Non destructive Post-irradiation tests of 6 rods prior to ovepower test already completed.
   First overpower test on 1 rod already performed including non destructive post-irradiation analysis of the same rods.
   Experimental procedure and facilities for subsquent over power tests alcready set-up

5. <u>Next steps</u>

Overpower tests performance will begin in 1975 on 16 rods with irradiation up to 30.000 MWD/t

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

None.

| Title 1 (Original language) | Classification |
|---|---|
| Prove di scoppio su guaina | 11.1 |
| **Title 2 (English)** | **Country** ITALY |
| Burst tests | **Sponsor** CNEN<br>**Organisation** CNEN |
| **Date initiated** January 1973<br>**Date completed** 1976<br>**Last updating** December 1976 | **Project Leader**<br>G.P. CALI' |

1. General aim

Mechanical properties of irradiated cladding under rupture conditions.

2. Particular objectives

Measurement of 1) Ultimate burst strength 2) Wall thinning at fracture 3) Total circumpherential elongation in irradiated cladding internally pressurized up to rupture conditions.

3. Experimental facilities and programme

Risö and Studsvik Laboratories

4. Project status

Burst tests have been completed within the following range of conditions: temperature: 280°C and room temperature; burn-up: 1.500 to 24.000 MWD/T; integrated fast neutron dose: $6 \div 30 \times 10^{20}$ n/cm$^2$; cladding diameter and type 15.06 mm ext.dia.(VDM), 14.26 mm ext.dia (Wolwerin).

5. Next steps

None

6. Relation with other projects

None

7. Reference documents

Internal reports

8. Degree of availability

| TITLE 1 (original language)<br>Sviluppo elementi di combustibile per HWR | Classification<br>11.1 |
|---|---|
| TITLE 2 (english)<br>HWR fuel development | Country: ITALY<br>Sponsor: CNEN<br>Organisation: CNEN |
| Date initiated 1975<br>Date completed In progress<br>Last updating June 1976 | Project Leader<br>G. VALLI |

1. General aim.

Development of a stress-strain correlation at high temperature for Zircalloy fuel cladding.

2. Experimental facilities and programme.

Joule effect heating facility in order to test samples with different heating rates in a temperature range of 800÷1100 °C. The inner pressure may be changed in a range of 3÷20 Ata. The sample is in an inhert atmosphere.

3. Project status.

Several samples are being measured to obtain a stress-strain correlation.

4. Next steps.

a) The same experimental facility will be modified to test in steam atmosphere.
b) To study the fuel behaviour at high temperature, an experimental facility will be designed with heaters held inside the Zr clad.

5. Degree of availability

The experimental facility is available at any time.

| PROJECT TITLE : Cladding in LOCA conditions | CLASSIFICATION 11.1 |
|---|---|
| SPONSORING COUNTRY : Italy | ORGANISATION : CNEN |
| DATE INITIATED : January 1975<br>DATE COMPLETED : Dec. 1976/79 | PROJECT LEADER : P. Cerretti |

Description :

1. General aim

Investigating modes of rupture and ballooning of irradiated and non irradiated zircalloy cladding under simulated LOCA conditions.

2. Particular objectives

Measurement of diameter of ballooned cladding at rupture, temperature, internal pressure and other relevant parameters in Zr-cladding, under simulated LOCA conditions.

3. Experimental facilities and programme

Facilities:

a) Elliptical irradiation oven (Casaccia)
b) Cladding heating by Joule effect (Casaccia)
c) In-pile loop (ESSOR reactor) (Ispra)

Programme

a) Tests on unirradiated and irradiated material under inert atmosphere (1975-76)
b) Tests on unirradiated material under oxidant atmosphere (1975-76)
c) In-pile tests on single rod and small cluster (1979).

4. Project status

4.1 Progress to date : construction of experimental facilities a) and b).

| Title 1 (Original language) Studio della conduttanza dell'intercapedine combustibile/guaina in barre di combustibile BWR | Classification 11.1 |
|---|---|
| Title 2 (English) Gap conductance in BWR fuel rods | Country ITALY<br>Sponsor CNEN<br>Organisation CNEN |
| Date initiated 1968<br>Date completed in progress<br>Last updating April 1977 | Project Leader<br>A. Nobili |

1. General Aim

The aim of the experimental program is to study the effect of the different parameters (linear heath rating, cold gap, power cycling, etc.) on the gap conductance.

2. Experimental facilities

The irradiation has been performed utilizing instrumented rigs capables to measure directly the power generated in the fuel as well as the fuel and cladding temperatures (1).
Reactor utilized: Avogadro, Siloè, Essor.

3. Project status

A first series of irradiation concerning the thermal behaviour of different fuel types has been completed (2).

4. Next steps

An experimental program is inprogress in order to improve the knowledge of the effects of the axial load, power cycling and cold gap on the gap conductance.
The irradiation will be performed in Essor and Siloè.

5. Relation to other projects

Preliminary studies to the ASCOT program.

6. Reference documents

The most important documents are:

6.1. A. Calza-Bini et al., "Esperienze di irraggiamento di capsule Cyranum per lo studio del comportamento termico della barra combustibile", RT/ING(70)15.

1110

| Title 1 (Original language) Studio della conduttanza dell'intercapedine combustibile/guaina in barre di combustibile BWR | Classification |
|---|---|
| | 11.1 |

6.2. A. Calza-Bini et al., "In-pile measurement of fuel cladding conductance for pelleted and vipac Zr-2 sheathed fuel pins", Nuclear Technology, Vol. 25, Jan. '75.

6.3. A.Calza-Bini et al.,"Esperienze Gioconda: studio sperimentale del comportamento termico di una barra combustibile ad $UO_2$: integrale di conducibilità e conduttanza dell'intercapedine tra combustibile e guaina", CNEN.

7. Budget, personnel involved

The budget for the second part of the program is 200 ML to irradiate 8 different fuel pins. This cost does not include the neutrons utilized in ESSOR reactor.

The personnel involved is 4 men for about 2 years.

8. Additional information

The results on gap conductance are utilized to calculate the stored energy in the fuel pin at the beginning of the LOCA accident.

| Title 1 (Original language) | Classification |
|---|---|
| Misura in pila della velocità di creep di una guaina di Zircalloy 2 di tipo Caorso | 11.1 |
| **Title 2 (English)** | **Country** ITALY |
| In-pile creep rate measurements of Zircalloy 2 cladding of Caorso fuel pin | **Sponsor** CNEN<br>**Organisation** CNEN |
| **Date initiated** 1977<br>**Date completed** 1977<br>**Last updating** April 1977 | **Project Leader**<br>Nobili |

1. General Aim

   Measurement of creep rate of a Zr-2 cladding under irradiation.

2. Experimental facilities

   The cladding sample will be placed in a CHOUCA Device in which the temperature is constant (± 1°C): the sample elongation is measured continuously by means of a resonant chamber system. The irradiation will be performed at Siloè Reactor.

3. Project status

   The sample fabrication is in progress; the irradiation will take place at the beginning of July.

4. The next step will be to set-up an experimental program in order to investigate the behaviour of the cladding of a fuel pin under irradiation.

5. Relation to the other projects

   Connection to the program performed by NUCLITAL.

6. Reference documents

   The most important reference documents are:

6.1. M. Masson, P. Millies, R. Wallop "Mesure en continue de deformation pendant l'irradiation en pile", CEA-CENG, Service des Piles $P_1$(NT) 461-103/68.

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| Title 1 (Original language) Misura in pila della velocità di creep di una guaina di Zircalloy 2 di tipo Caorso | Classification<br>11.1 |
|---|---|

6.2 R. Warlop "Ensemble de mesure de deformation", CEA-CENG/G, Service des Piles, NT/$P_i$ 293-05-124/70.

6.3 A. Calza-Bini et al. "Misura in pila di velocità di creep su tubi di guaina in acciaio inox in geometria veloce", CNEN/RT/ING(75)16.

7. Budget, personnel involved

The cost of the experiment will be about 50 ML for $2000^h$ of irradiation.
The personnel involved is one man/year.

8. Additional information

The results will be utilized to calculate the permanent deformation of the cladding under internal pressure.

| Title 1 (Original language) | Classification |
|---|---|
| Comportamento meccanico e chimico di guaine di Zr ad alta temperatura | 11.1 |
| **Title 2 (English)** | Country ITALY |
| High temperature mechanical and chemical behaviour of cladding tubes. | Sponsor CNEN<br>Organisation CNEN |
| Date initiated 1975<br>Date completed in progress<br>Last updating April 1977 | Project Leader<br>S. OMARINI |

1. General aim

Development of stress strain and oxide growth rate correlations at high temperature for Zircaloy fuel cladding.

2. Experimental facilities and programme

a) Joule effect heating facility for testing of Zr cladding tubes with different heating rate or temperature transients up to 1100°C. Inner pressure range 20-60a ta.

b) Radiant heat facility for the high temperature Zr oxidation studies.

3. Project status

The first tests on CIRENE HWR reactor and on CAORSO LWR geometries already tested. Tests completed in 1977.

4. Next steps

a) Test of cladding tubes with a variable inner pressure to simulate the differential pressure of a LOCA transient.

b) Development of internal heater for fuel rod simulation and H-Gap studies.

5. Reference documents

Internal reports.

6. Degree of availability

To a limited extent.

7. Budget, Personnel involved

8 persons.

1115

| | |
|---|---|
| PROJECT TITLE :<br>RELIABILITY OF CLASSICAL NDT | $\boxed{12.3}$<br>11.2.3 LWR<br>11.1 |
| SPONSORING COUNTRY :<br>Commission of the European Communities | ORGANISATION :<br>JRC Ispra |
| DATE INITIATED : 1977<br>DATE COMPLETED : 1980 | PROJECT LEADER :<br>S.J. CRUTZEN |

1117

| N.V. KEMA | CLASSIFICATION : 11.1 |
|---|---|
| TITLE : Splijtstofverlenging tijdens bedrijf | COUNTRY: THE NETHERLANDS |
| | SPONSOR : KEMA<br>ORGANIZATION : KEMA |
| TITLE (ENGLISH LANGUAGE): Fuel elongation during power rating | PROJECTLEADER : |
| INITIATED : - / LAST UPDATING : 1978 | SCIENTISTS : W.R. van Engen, J. Hoekstra, W. Slegers |
| STATUS : - / COMPLETED : 1977 | |

General aim

Pellet-clad interactions.

Particular objectives

To detect possible pellet-clad interaction situations in the core of a power reactor by means of

a) absolute elongation of the fuel rods or whole assemblies
b) change in time constant for heat transfer of the fission heat to the coolant

Experimental facilities

Dodewaard nuclear power plant.

Project status

Still in progress.

Next steps

To come to an additional method for incore measurements to have better knowledge over actual core conditions.

Relation to other projects

Measuring technique is based on the Halden incore instrumentation experience with differential transformers.

Reference documents

None.

Degree of availability

Through the organization KEMA.

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX

Classification 11.1

| | |
|---|---|
| Title 1 Fuel Clad Distortion under Accident Conditions. | Country UK |
| Title 2 | Sponsor UKAEA<br>Organisation<br>Reactor Fuel Laby.<br>Springfields |
| Initiated 1971 Completed<br>Status Continuing Last updating | Project leaders<br>Dr. K. M. Rose |

1 General Aim

To secure acceptable core conditions following a loss-of-pressure accident (LWR).

2. Particular Objectives

To discover whether internal pressure across zircalloy clad pins causes deformation which obstructs further cooling.

3. Experimental Facilities and Programme

Fuel bundles up to 1 m. length are heated in a furnace, through defined temperature and pressure sequences. The bundles are dismantled and distortions measured.

Project Status

Many such tests have been performed. Extension blockage can be produced particularly by low temperature (850°C) sequences.

The case is being considered for a more representative form of heating.

Reference Documents

SNI Report SNI 1/14. Oct. 1973

Classification

11:1

| Title 1 | COUNTRY<br>UNITED KINGDOM |
|---|---|
| HIGH TEMPERATURE CREEP OF ZIRCALOY | SPONSOR UKAEA |
| | ORGANIZATION<br>RFL SPRINGFIELDS |
| Title 2 | Project Leader<br>DR K M ROSE |
| Initiated 1971 Completed : | Scientists: |
| Status : Last updating | |

Description:

1. General Aim

To secure acceptable core conditions following a loss of pressure accident (LWR) and SGHWR.

2. Particular Objectives

To define limits of temperature and stress for which the Zircaloy can will not creep significantly.

3. Experimental Facilities and Programme

Short lengths of fuel can are taken through a defined sequence of temperature and internal pressure. The deformation of the can is measured throughout the transient.

4. Project Status

Many tests have been performed, originally with inert gas atmosphere, currently with a steam atmosphere as this significantly affects the results. The test is specific to the reactor temperature transient, so the tests are repeated from time to time as changes are made to the reactor design.

Reference Document

SNI Report SNI/1/14 Oct 1973.

| Classification 11.1 | |
|---|---|
| Title 1<br>FUEL CLAD DISTORTION UNDER ACCIDENT CONDITIONS | COUNTRY<br>UNITED KINGDOM<br>SPONSOR UKAEA<br>ORGANIZATION<br>RFL SPRINGFIELDS |
| Title 2 | Project Leader<br>DR K M ROSE |
| Initiated 1971 Completed :<br>Status : Last updating | Scientists: |

Description:

1. General Aim

To secure acceptable core conditions following a loss-of-pressure accident (LWR) and SGHWR.

2. Particular Objectives

To discover whether internal pressure across Zircaloy clad pins causes deformation which obstructs further cooling.

3. Experimental Facilties and Programme

Fuel bundles up to 2m length are heated in a furnace, through defined temperature and pressure sequences. The bundles are dismantled and distortions measured.

4. Project Status

Many such tests have been performed. Extended blockage can be produced particularly by low temperature (850°C) sequences.

The case is being considered for a more representative form of heating.

Reference Documents

SNI Report SNI/1/14 Oct 1973.

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31. 12. 77 | 11.2 | RS 245 |
| Vorhaben/Project Title<br>Großbehälter - Phase 1<br><br>Full Size Vessel - Phase 1 | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Staatl. Materialprüfungsanstalt (MPA) Universität Stuttgart |
| Arbeitsbeginn/Initiated<br>1. 10. 1976 | Arbeitsende/Completed<br>31. 12. 1978 | Leiter des Vorhabens/Project Leader<br>Dr. Sturm / Doll |
| Stand der Arbeiten/Status<br>continuing | Berichtsdatum/Last Updating<br>31. 12. 1977 | Bewilligte Mittel/Funds<br>8.983.828,-- DM |

1. General Aim

Elaboration of a rough specification for the research programme "Full Size Vessel" as well as purchase and transport of a boiling water reactor pressure vessel. Fabrication of a part of a shell made of material in boundary condition.

2. Particular Objectives

The rough specification shall contain the following urgent examinations to be performed on the full size vessel:

- nondestructive examinations (flaw detection probability) detection of the zero-condition
- removal of trepan to verify probable defects
- internal pressure test

3. Research programme

4. Experimental Facilities

Lining of testing cell has already begun (internal diameter 11,5 m, clear hight 20,0 m, Fig. 1). The underground and concrete work are underway.

5. Progress to Date

Rough specification:

Elaboration of section "Non-destructive examinations" (flaw detection probability) which has been carried out by the Institut fuer zerstoerungsfreie Pruefverfahren (IzfP), Saarbruecken, especially with regard to

- the description of a weakered part of the shell to be welded-in

- Control of field welds
- detection of the state as delivered concerning the total vessel including closure head, head, and studs
- observance of the flaw development and flaw propagation during loading tests
- elaboration of sections: detection of the zero condition and removal of trepan, cold pressure test
- construction of the frame

Fabrication of the vessel at Breda, Milan (compare also Fig. 1)

| | |
|---|---|
| part of head | 32 control rod drives and one Nozzle O have been welded |
| lower part of shell | Nozzle D 1 and D2 as well as Nozzle C have been closed by means of closure head |
| central part of shell (part 3) | ready for dispatch |
| "weakened" part of shell (part 4) | prepared for shipment to the FRG where it shall undergo subsequent treatment |
| Flange ring (part 5) | Nozzle B has been closed with the closure head. Surface crack examinations of welds have been carried out |
| Closure head (part 6) | final annealing has been completed. Flange holes are to be drilled. |

Transport has been prepared by KWU

Fabrication of the weakened part of shell (part 4)

- choice of material has been made (segregation melts)
- closure head semi-rings MPA-Melt 34/35 or KS 2/3
- Steam generator head MPA-Melt 44 or KS 4
- Forgins of ingots from closure head semi-ring and non-destructive examinations

## 6. Results

-

7. Next Steps

- rough specification to be completed
- parts of the vessel to be fabricated at Breda
- part of shell (part 4) to be shipped to the FRG (Jan. 78)
- Further closure head flange semi-rings to be forged
- Nozzles to be fabricated from steam generator heads
- tests certificates concerning the raw material of the pressure vessel to be evaluated
- Frame to be fabricated
- Testing cell to be completed

8. Relation with other Projects

RS 304 research programme of integrity of components

9. References

—

10. Degree of Availability

—

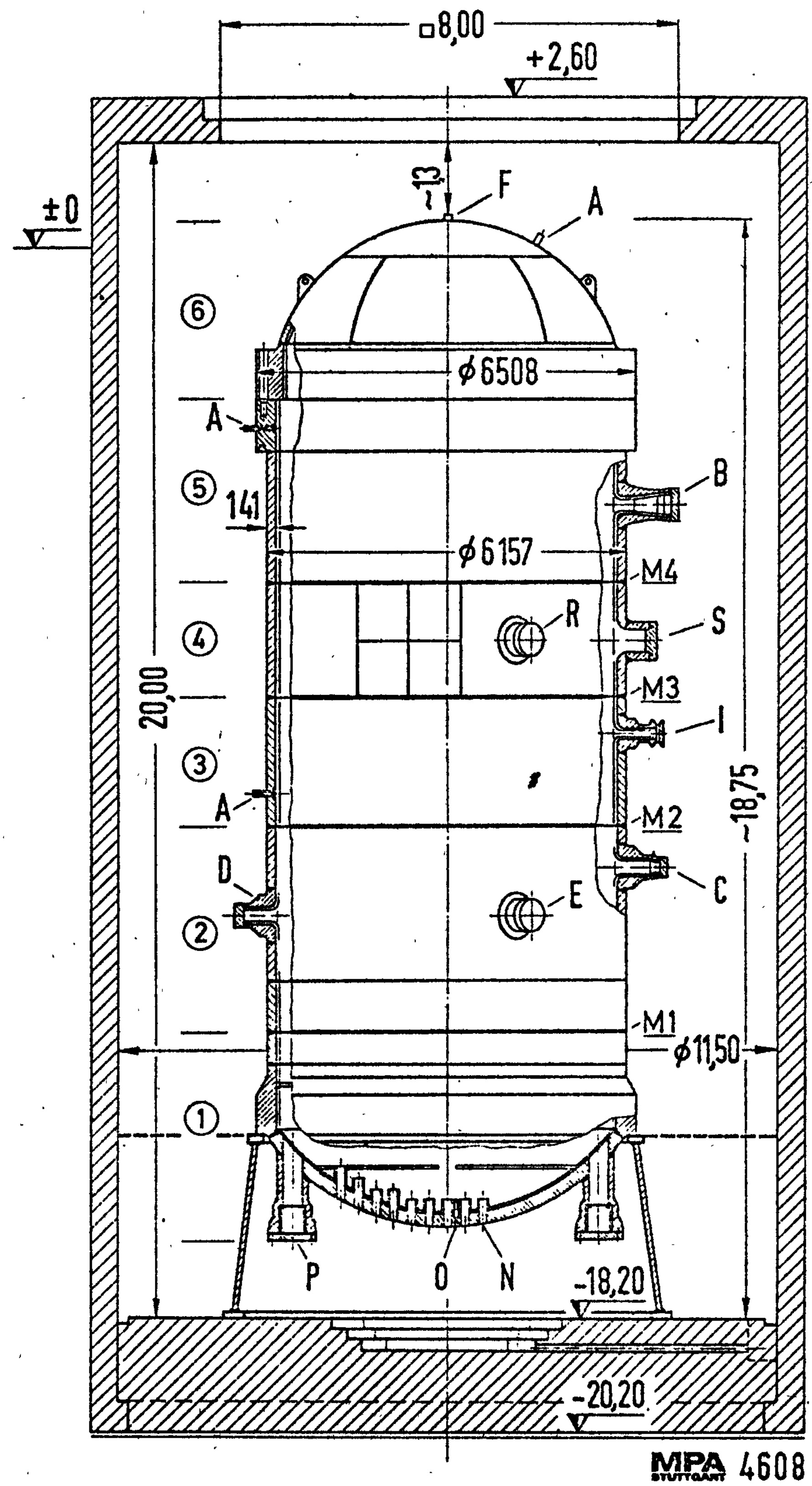


Fig. 1: Dimensions of the Full Size Vessel and the Testing Cell

Dimensions of the Vessel in mm

Dimensions of the cell in m

| Berichtszeitraum/Period 1.9.1977 - 31.12.1977 | Klassifikation/Classification 11.2 | Kennzeichen/Project Number RS 304 |
|---|---|---|
| Vorhaben/Project Title Research Programme "Integrity of Components" Forschungsprogramm Komponentensicherheit (FKS) | | Land/Country FRG; Fördernde Institution/Sponsor BMFT/Industry; Auftragnehmer/Contractor Staatliche MPA University of Stuttgart Pfaffenwaldring 32 7000 Stuttgart 80 |
| Arbeitsbeginn/Initiated 1.9.1977 | Arbeitsende/Completed 31.12.1978 | Leiter des Vorhabens/Project Leader Dr. Issler/Dr. Foehl |
| Stand der Arbeiten/Status Completed | Berichtsdatum/Last Updating December 1977 | Bewilligte Mittel/Funds DM 1.568,565,-- |

1. General Aim

The Research Programme Integrity of Components (F KS) which is jointly financed by the Federal Minister for Science and Technology (BMFT), system manufacturers, component and material manufacturers and utilities comprises a total volume of approximately 50 Mill. DM during a running period of five years. Its objective is to verify the safety of components which are relevant for light water nuclear power plants. Phase 1 of the F KS is focused mainly on the long time behaviour of the reactor pressure vessel in older plants for which the safety margin shall be investigated with regard to unfavourable state of material, flaw, and stress ("Lower-bound-concept).

-For budgetary reasons the programme is split up in formal projects for each fiscal year.-

2. Particular Objectives

The particular objectives of the F KS, are derived from the above mentioned conditions of real and assumed ranges of parameters. They must cover situations from limiting conditions, to those exceeding limiting conditions for manufacture, test, operation, accident and faulted conditions of the reactor pressure vessel and allow evaluation in view of their relevance to safety. Particular objectives are arranged as follows according to the sequence and strategy of the programme

2.1. Intentional and reproducible production of various materials, flaw and stress conditions ("procurement phase")

- covering operational influences (irradiation, corrosion, tempering, mechanical loading, cf. pt. 2.2)

2.2. Complete description of these conditions ("characterization phase") as well as their influence on limiting loading as related to crack initation, crack propagation, crack arrest and instability

- flaw detection probability by means of nondestructive methods
- correlation between metallographic results and mechanical behaviour
- applicability of material characteristics determined on small tensile specimens
- assessment of the applied HAZ-simulation techniques
- reliability of the theoretical stress analysis in order to calculate load and residual stresses

2.3. Effect of these conditions concerning safety ("applicability phase")

- application of these results under RPV conditions
- optimization of manufacture and operation of the RPV
- conclusions concerning safety concepts and codes

3. <u>Research Programme</u>

The research programme may be derived from different objectives and can be grouped according to section 2 in following steps

3.1. Procurement phase

- melting, forming, welding and heat treatment of the materials, simulation of heat affected zones
- aging, irradiation, corrosion and fatigue tests

3.2. Characterization Phase

- nondestructive (especially ultrasonic) testing and investigations in the field of materials science (chiefly metallographic) on the test materials and specimens (pre- and re-examinations)

- loading tests to define the mechanical properties (strength and toughness)
  small specimens up to 100 mm thickness (tensile, notch bar impact, drop weight, fracture mechanics, low cycle fatigue tests
  in the as received condition (base material, weld joint), and after HAZ-simulation, aging, irradiation and under corrosion
- large specimens up to 500 mm thickness (tensile, bending and centrifugal specimen) weld joints and base material
- intermediate size vessels up to 150 mm wall thickness with "welded in" pieces, nozzles and ring segments
-
- accompanying stress analysis and fracture mechanics calculations (linear-elastic, elastic-plastic)

3.3. Application Phase

- listing of real and hypothetical RPV conditions as well as safety concepts
- correlation with test results of the F KS
- development of the transferability of results
- proposing optimization procedures for manufacture and operation

4. Experimental Facilities, Computer Programme

In order to perform the research programme it is necessary to incorporate methods from all fields of material science, strength of materials and material examination. The following methods will be employed:

- nondestructive test methods
  (magnetic particle inspection (MP), liquid penetrant examination (PT), X-ray examination, ultrasonic examination)
- metallographic test methods (micro analysis, optical microscopy, electron microscopy such as SEM, TEM)

- HAZ-simulation technique
  (inductive, resistance and electron beam overheating)
- irradiation facilities
  (research and power reactors including dosimetry)
- corrosion facilities
  (static and refreshing-autoclaves, including measuring technique)

- mechanical properties testing facilities
  - small specimens
    (universal testing machines, static and oscillating, drop weight testing machine, impact testing machines both pendulum and tensile types, including measuring technique)
  - Large specimens
    (static tensile testing machine up to 100 MN
    fatigue testing machine up to 10 MN
    drop weight testing facility up to 100 000 Nm,
    centrifugal test facility
  - intermediate size vessel
    (burst chamber, pumps up to 2000° bar)
- computer programme
  - finite element programme including 3D-elastic-plastic (e.g. ASKA, MARK) including transient temperature fields (e.g. SMART) and cracked components (e.g. BIGIF)
  - systems of programmes to define the neutron spectrum and damage function for irradiation

5. Progress to Date

During the period of the report representing the first months of the programme it was intended to conduct work which includes in particular
- organization of the total programme
- specification of the year 1978
  (general and detailed net plans, bar charts),

moreover, it was possible to begin the initial activities of point 3.1 of the research programme

- procurement of available component heats
- specification and procurement of
  - special devices for testing facilities
  - peripherial facilities for computer processing
- preparation of documentation and evaluation programmes

6. Results (not applicable)

7. Next Steps

During the course of the planning phase it became necassary to extend the planning period until the end of March 1978. This phase shall specially serve the purpose of describing in detail the basis on which the tender 1978 will be based. Moreover, the material to be melted (component and special melts) will be chosen and supplied in view of analyse and melting parameters, in addition to continuing the organization of the work during the extended planning period.

8. Relation with other Projects

The F KS is closely related with the following projects

| | |
|---|---|
| - Urgency Programme | RS 101 (concluded) |
| - HDR Safety Programme | RS 123 |
| - Full Size Vessel | RS 245 |

9. References

The programme is based on the "Summary of the Detailed Specification" drawn up in connexion with the overall specification (RS 192). The results obtained during this report period are summarized in the appropriation document for the committee of experts "Materials and Strength" and the steering committee.

10. Degree of Availability of the Reports

The "Summary of the Detailed Specification" can be obtained from the GRS.

1135

| 141-1 -01 | | 11.2 |
|---|---|---|
| Titre<br><br>Etude de faisabilité d'essais sur cuve PWR à échelle 1 | | Pays<br>FRANCE<br><br>Organisme directeur<br>EURATOM |
| Titre (anglais)<br><br>Feasibility of tests to be made on scale 1 PWR pressure vessel. | | Organisme exécuteur<br>CEA/DSN/SETSSR<br><br>Responsable<br>J. DUFRESNE<br>DSN/SETSSR/Fontenay |
| Date de démarrage 1/1/77 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 1/7/78 | Dernière mise à jour 12/77 | |

1 - Objectif général :

Etudes de sûreté sur le comportement général des cuves de réacteur à eau .

Cette étude est exécutée dans le cadre d'un contrat entre EURATOM et l'Université de PISE . Celle-ci s'est entourée d'experts choisis dans les différents pays de la Communauté .

2 - Objectifs particuliers :

1) Enquête auprès des constructeurs, exploitants, et autorité de sûreté de France et de Belgique sur l'intérêt d'effectuer un programme d'études expérimentales à partir d'une cuve de réacteur à eau à échelle 1 : 1.

2) Recensement des cuves à échelle 1 : 1 existant en Europe, et dont la commande a été annulée. Etude des différents programmes d'essais nécessitant d'être effectués sur des cuves à échelle 1 : 1.

3 - Installations expérimentales et programme :

Cuves à échelle 1, le cas échéant.

.../

4 - Etat de l'étude :

1) Avancement à ce jour

4 réunions ont eu lieu au cours du semestre. Le rapport final est en cours de rédaction.

2) Résultats essentiels

- A la suite de l'enquête effectuée auprès des constructeurs européens, il résulte que ceux-ci ne possèdent plus de cuves fabriquées ou en cours de fabrication, et dont la commande a été annulée.

- Deux enquêtes ont été entreprises auprès des laboratoires, exploitants, constructeurs et autorité de sûreté. La première avait pour but d'identifier les principaux programmes actuellement en cours en Europe sur l'étude du comportement des matériaux utilisés dans la construction des cuves de réacteurs à eau. La seconde a permis de recenser l'intérêt que porteraient ces différents organismes à l'étude du comportement d'une cuve entière ou d'une partie de cuve, au moyen d'essais simulant les conditions de fonctionnement.
Les études proposées par les organismes consultés portent essentiellement sur :
- Les contrôles non destructifs ultra-sons. émission acoustique
- Le comportement des défauts naturels aux cyclages thermiques.
-Les critères de rupture en mécanique élastoplastique

5 - Prochaines étapes :

Rédaction du rapport final.

| 141-1 -02/4111-10 | | 11.2 |
|---|---|---|
| Titre<br><br>Evaluation de la probabilité de rupture d'une cuve de réacteur à eau pressurisée. | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br><br>Failure probability calculation of a PWR pressure vessel. | | Organisme exécuteur<br>CEA + EURATOM + FRAMATOME<br><br>Responsable<br>J. DUFRESNE<br>DSN/SETSSR/Fontenay |
| Date de démarrage 1975 | Etat actuel<br>en cours | Scientifiques<br><br>Répartis dans différents laboratoires. |
| Date prévue d'achèvement 1978 | Dernière mise à jour 12/77 | |

1 - Objectif général :

Evaluation de la probabilité de rupture d'une cuve de réacteur à eau pressurisée par une méthode probabiliste basée sur les lois de la mécanique des ruptures. Cette méthode devrait permettre d'aboutir à :

- une comparaison entre la probabilité de rupture de la cuve de réacteur et celle des canalisations du circuit primaire.
- définir l'importance relative des différents paramètres intervenant dans le mécanisme de la rupture.
- une comparaison entre les différentes méthodes de construction et d'exploitation.

2 - Objectifs particuliers :

Toutes les données sont introduites sous forme d'histogrammes. Le projet général se subdivise en 7 sous programmes.

.../

1) Revue et discussion des principales méthodes statistiques et probabilistes utilisées.

2) Collecte des défauts recensés dans les différentes cuves existantes. 3 contrats ont été signés avec des fabriquants européens pour recueillir les données sous formulaire standard.

3) Recueil des données relatives aux propriétés mécaniques de l'acier A 508 Cl 2 sous forme probabiliste. La plupart de ces données sont obtenues à partir de l'US-HSST programme.

4) Validation des formules de calcul du coefficient d'intensité de contrainte.

5) Validation des lois de propagation des fissures par vérification des données expérimentales obtenues sur des éprouvettes CT, et sur des plaques ou des tubes comportant des défauts elliptiques artificiels.

6) Analyse du système de contrôle du réacteur afin de définir les conditions de fonctionnement programmées et accidentelles ainsi que leur probabilité d'occurrence.

7) Mise au point d'un programme de calcul faisant entrée, sous forme d'histogramme, le défaut initial, la propagation pendant le fonctionnement, la résistance du matériau et les critères de rupture.

3 - Installations expérimentales et programme :

Travaux effectués avec le concours d'EURATOM et de FRAMATOME ainsi que, par contrats, avec d'autres industries et universités : Université de Technologie de Compiègne, Université de Metz - Creusot Loire (Saint-Etienne), - CETIM (Senlis), - STCAN (Paris).

Les calculs et la mise en oeuvre des codes sont effectués au Centre Commun de Recherche d'ISPRA.

4 - Etat de l'étude :

La 1ère partie est terminée : il en a été déduit que le nombre de cuve est actuellement trop insuffisant pour en tirer une valeur probabiliste de rupture.

La 4ème partie est terminée : 230 ruptures fragiles ont été recensées. On constate que les valeurs obtenues à partir des formules de SHAH et KOBAYASHI sont de 5 % inférieures à celles obtenues à partir des résultats expérimentaux.

La 6ème partie étudie 36 cas possibles ainsi que leurs probabilités d'occurrence.

L'étude du code de calcul est effectuée au Centre d'ISPRA.

Le code de base est " COVAL " ; les données d'entrée sont introduites sous forme d'histogramme.

- longueur et largeur du défaut initial.

-

- Coefficients de la formule de PARIS.
- résistance du matériau.

7 - Documents de référence : disponibles (partenaires de l'accord)

" Programme d'étude concernant l'évaluation de la probabilité de rupture d'une cuve de réacteur à eau ", J. DUFRESNE. Rapport DSN 94.

" Etude probabiliste de la rupture de cuve de chaudière nucléaire à eau ordinaire, premier rapport d'avancement Octobre 1976, J. DUFRESNE DSN 116 , et 2ème rapport d'avancement Mars 1977, DSN 145 .

| Title 1 (Original language) | Classification |
|---|---|
| Sull'inizio di propagazione di una rottura in un recipiente in pressione costruito con materiale duttile. | 11.2 |
| **Title 2 (English)** | **Country** ITALY |
| On the starting of a crack propagation in a cylindrical pressure vessel of ductile material. | **Sponsor** CNEN - CNR<br>**Organisation** University of Pisa |
| **Date initiated** 1968<br>**Date completed** 1980<br>**Last updating** May 1977 | **Project Leader**<br>CARMIGNANI Costantino |

## Description

The aim of this research program is to evaluate the possibility of using linear elastic fracture mechanics concepts to forecast the starting of crack propagation in a ductile material.

In the first phase it was developed a theoretical work to evaluate the elastic stress field around the tip of the crack (Ref. 1.2). After, several cylindrical pressure vessels with axial through thickness cracks, were bursted by internal pressure (Ref. 3.4.5). The results of this experimental work demonstrated the possibility of using cylindrical pressure vessels, like those above mentioned, to determine the critical parameters ($K_c$ or COD) of the material.

Now an experimental work is completed to evaluate size effect on the determination of the critical parameter with the method above indicated (Ref. 6).

At present we are working on an experimental research with the aim to evaluate the behaviour, by using linear and elasto-plastic fracture mechanics concepts, of cylindrical vessels with axial or circumferential throughwall, or partial-through, cracks.

Vessels will be loaded with internal pressure and/or bending moment.

Facilities used are a pressurization system with an electric/manual pump for bursting tests at room temperature and a frame for bending tests.

## Relation to other projects

Research outlined is strictly selected to joint experimental programs developed by CSM (Centro Sperimentale Metallurgico), Palermo University Laboratory and CNEN.

## Reference documents

1. CARMIGNANI C., DEL PUGLIA A.
Applicazioni di metodi di calcolo elettronico all'analisi del comportamento elastico di piastre rettangolari contenenti una fessura passante centrale, caricate normalmente al loro piano medio.
Atti Ist. Mecc. Appl. - Univ. Pisa, n° 11 - Anno Acc. 1968-'69.

1142

| Title 1 (Original language) | Classification |
|---|---|
| Sull'inizio di propagazione di una rottura in un recipiente in pressione costruito con materiale duttile. | 11.2 |


2. CARMIGNANI C., CELLA A.
Elastic Analysis of Cracked Thin Shells by the Finite Element Method.
Proc. 1st SMIRT, Berlin, 1971, vol. 5 pp. 57 ÷ 76.

3. CARMIGNANI C.
Ricerca teorica e sperimentale sulla propagazione delle rotture nei recipienti in pressione.
Atti del Convegno Nucleare di Pisa, vol. 1, CNEN, 1971, pp.279 ÷ 319.

4. CARMIGNANI C., REALE S.
Elaborazione dei risultati della ricerca sperimentale sulla propagazione delle rotture nei recipienti in pressione.
Atti Istituto di Impianti Nucleari - Univ. Pisa. RP 153(73), 1973.

5. CARMIGNANI C., CIBECCHINI P., REALE S.
Caratterizzazione di un acciaio al Nichel per impieghi a bassa temperatura con i criteri della meccanica della frattura.
Proc. 5th Int. Conf. on Exp. Stress Analysis, Udine, 1974, paper 18 pp. 2.5 ÷ 2.12.

6. CARMIGNANI C., REALE S.
Valutazione dell'effetto scala nella determinazione del parametro critico del materiale, mediante prove di scoppio di modelli cilindrici fessurati.
Atti del IV Convegno Nazionale AIAS, Roma, Sept. 1976.

| Title 1 (Original language) Ricerca sperimentale sull'accumulo delle deformazioni e sulle rotture per fatica di tubazioni soggette a carichi variabili ripetuti. | Classification<br>11.2 |
|---|---|
| Title 2 (English)<br>Experimental research on incremental collapse and plastic fatigue of piping components. | Country ITALY<br>Sponsor CNEN<br>Organisation University of Pisa |
| Date initiated 1976<br>Date completed 1980<br>Last updating May 1977 | Project Leader<br>NERLI G. |

This research is concerning with extending understanding of (i) incremental collapse phenomena, with progressively increasing deflections, and (ii) plastic fatigue phenomena in stainless steel piping components subjected to variable repeated loads.

Loads will be determined that produce (i) permanent strains and deformations in excess of the design limits and (ii) plastic fatigue ruptures in T branched pipes fixed at their flanged ends and loaded in two sections by variable repeated loads. The reference is to "emergency" and "faulted" load conditions as they are defined in ASME III Code.

The models are fabricated by stainless steel pipe and simulate some primary circuit piping components. The models will be tested using an existing experimental rig.

Experimental incremental collapse results will be compared with calculated ones by an existing computer program based on finite element technique. The program takes into account the effects of material strain hardening, where Section III of the ASME Code is based primarily on a limit analysis approach, that is on an elastic-perfectly plastic behaviour of the material.

One important objective of the research is to determine material and weld (of flanges) qualification parameters to be used for obtaining the best results by the calculus.

Some researches on plastic behaviour of carbon and stainless steel pipes were just performed at "Istituto di Impianti Nucleari" and at "Istituto di Ingegneria Meccanica" of Florence University/1,2,3,4,5/.

The results of the present research will be entirely published.

The research will be performed under a contract with CNEN.

References

1. - A. DEL PUGLIA, G. FILIPPELLI, G. NERLI
"Esperienze sul collasso di tubazioni iperstatiche non piane"
Disegno di Macchine N. 3 (1973).

1144

| Title 1 (Original language) Ricerca sperimentale sull'accumulo delle deformazioni e sulle rotture per fatica di tubazioni soggette a carichi variabili ripetuti. | Classification 11.2 |
|---|---|


2. - A. DEL PUGLIA, G. NERLI
"Experimental Research on Elastoplastic Behaviour and Collapse Load of Statically Indetermined Space Tubular Beams"
2nd Int. Conf. on Structural Mechanics in Reactor Technology, Berlin, Sept. 1973.

3. - P. CITTI, A. DEL PUGLIA, G. NERLI
"Experimental Study on the Effect of Variable Repeated Loads on Steel Piping Components"
3rd Int. Conf. on Structural Mechanics in Reactor Technology, London, Sept. 1975.

4. - P. CITTI, A. DEL PUGLIA, G. NERLI
"Esperienze su tubazioni soggette a carichi variabili ripetuti"
3rd Convegno Nazionale AIAS, Bologna, Oct. 1975.

5. - P. CITTI, P. RISSONE
"Calcolo e verifica sperimentale della deformata di una travatura in campo elastico-incrudente"
IV Convegno Nazionale AIAS, Roma, Set. 1976.

7143

19 45

| Title 1 (Original language) | Classification |
|---|---|
| Fatica ad alta temperatura di elementi strutturali | 11.2 |
| **Title 2 (English)** | Country ITALY<br>Sponsor CNEN<br>Organisation University of PISA |
| High Temperature Fatigue on Structural Elements | |
| Date initiated 1976<br>Date completed 1980<br>Last updating May 1977 | Project Leader<br><br>A. DEL PUGLIA |

Description:

Research program: to gather experimental information on the High Temperature Low Cycle Fatigue behaviour of simple tubular welded structures in AISI 304, 316 and other materials. The load conditions are characterized by:
- temperature (mainly comprised between 430°and 650°C)
- imposed strain (ranging from 0,5 to 2%)
- hold time (from 0 to 30 min and over).

A particular effort has been performed at properly characterizing the specimens and related manufacturing technology.

Up to now have been tested butt welded pipe segments with flanged ends, 500 mm long and with 60,3 mm outer diameter.

Facilities: test apparatus, which is able to house specimens up to 650mm lenght and to apply a 20 ton. maximum axial load.

Temperature and load cycle control instrumentation.

Load, axial and diametral deformation transducers and related instrumentation.

Acoustical emission instrumentation.

Reference documents

1. - DEL PUGLIA A.
Sul comportamento a fatica ad alta temperatura di elementi strutturali in acciaio inossidabile di tipo austenitico.
Atti Ist. Meccanica Applicata e Costruzione di Macchine - Università di Pisa n° 50 - Pisa (1974).

1146

| Title 1 (Original language) | Classification |
|---|---|
| Fatica ad alta temperatura di elementi strutturali | 11.2 |


2. - DEL PUGLIA A., BARTOLI G.
Prove preliminari ad alta temperatura di elementi tubolari saldati in acciaio inossidabile.
Atti III° Convegno AIAS - Bologna 1975.

3. - DEL PUGLIA A., MANFREDI E., CABRELE G., ZOLA S.
Fatica ad alta temperatura di elementi strutturali tubolari saldati in acciaio inossidabile: primi risultati.
Atti IV Convegno AIAS - Roma (1976).

4. - DEL PUGLIA A., MANFREDI E., TOMMASSETTI G.
High Temperature Fatigue Experiments on Welded Stainless Steel Structures.
4th International SMIRT Conference - San Francisco (1977)

1147

| Rotterdam Dockyard and others | CLASSIFICATION : 11.2 |
|---|---|
| **TITLE :** Breukanalyse Onderzoek aan Stompen (BROS) | COUNTRY: THE NETHERLANDS |
| **TITLE (ENGLISH LANGUAGE):** Fracture analysis research on nozzle intersections (BROS) | SPONSOR : Ministry of Economic Affairs<br>ORGANIZATION : Rotterdam Dockyard and others |
| | PROJECTLEADER : C.J. Drijver |
| INITIATED : March 1972 — LAST UPDATING : May 1978<br>STATUS: in progress — COMPLETED : End 1978 | SCIENTISTS : M.J.C. Broekhoven, H.J.M. van Rongen, A. de Sterke/M.J.A. Koning |

General aim

Crack extension behaviour in heavy section nuclear steel pressure vessels in areas of complicated geometry.

Particular objectives

In particular attention will be paid to crack occuring in areas of complicated geometry and stress distribution, notably nozzle corner regions. The program covers the following items:

- early detections of defects
- detailed surveillance of the growth of defects
- establishment of prediction for further growth of defects by fatigue and fracture.

Experimental facilities and program

The main activities are:

- theoretical research directed towards the development of efficient computation methods to calculate the crack extension behaviour for complicated configurations, such as nozzle corner cracks, using advanced analytical and finite element techniques,
- experimental investigations of crack extension behaviour in uni-axial and bi-axial loaded cracked nozzle-on-flat-plate models, manufactured mainly from nuclear grade pressure vessel steel (ASTM A508 Cl.2),
- determination of fatigue and fracture (elastic and elastic-plastic) related material parameters for A508 material,
- research on the applicability of Acoustic Emission techniques to detect, locate and characterize crack extension in the material of concern,
- the main test facility to be used is a bi-axial loading bench, suitable for plate dimensions 700x700x15 mm, capacity 200 tonnes static force, 200 tonnes dynamic loading, frequency 1Hz.

Project status

- A series of nozzle-on-flat-plate models has been uni-axially fatigue loaded and after sufficient crack growth, fractured by overload,
- Computer programs have been completed,
- Accoustic Emission testing has been completed,
- Fatigue and fracture ($J_{IC}$-) tests on standard specimens have been completed,
- The bi-axial loading bench has been constructed and tested,
- The testing of flat plate test pieces using the biaxial loading bench is in progress, and
- The publishing of the final technical reports (in English language) is in progress.

Next steps

The next steps will be the testing of flat plate test pieces of A508 material using the bi-axial bench and the publishing of the final technical report concerning the above mentioned tests.

Relation with other projects

Results of this project will be used within the "EPOSS" project.

Reference documents

A serie of about 50 technical and progress reports have been prepared. Nearly all these reports are written in the Dutch language.

Degree of availability

The reports have been submitted to the Ministery of Economic Affairs, Laan van Nieuw-Oost Indië 123, The Hague. Requests for obtaining copies should be sent to this address.

Budget

$f.\ 6.10^6$.

Personnel

46 manyears.

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 11.2.1 | RS 101/RS 101 A |
| Vorhaben/Project Title<br>Reactor Pressure Vessel<br>- Urgent Programme 22 NiMoCr 3 7 -<br>Part 13 - 10 000 Mp-Tensile Testing Machine<br><br>Construction of Testing Hall by the Staatliche Materialprüfungsanstalt in order to conduct the Research Programme Reactor Safety | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Staatliche Material-prüfungsanstalt (MPA) Stuttgart |
| Arbeitsbeginn/Initiated<br>1.12.1972 | Arbeitsende/Completed<br>31.12.1978 | Leiter des Vorhabens/Project Leader<br>Prof. Krägeloh/Doll |
| Stand der Arbeiten/Status<br>continuing | Berichtsdatum/Last Updating<br>31.12.1977 | Bewilligte Mittel/Funds<br>DM 15.246.897,-- |

1. General Aim

Quantification of Safety of Reactor Pressure Vessels.

2. Particular Objectives

Errection of a 10 000 Mp-Tensile Testing Machine and construction of the testing hall, Figs. 1 and 2, with fundaments and testing cell for the full size vessel (RS 245).

3. Research Programme

4. Experimental Facilities

The machine parts of the 10 000 Mp-Machine are completed and most of them are stored at the MPA. All necessary heavy transports were carried out Fig. 3. The fundaments for the machine are completed and the testing cell is under construction, Fig. 4.

5. Progress to date

10 000 Mp-Maschine: The cylinder (135 Mp) has been mounted on the fundament, Fig. 5, the complete shock absorbers as well and all other parts are being prepared.

1. Stage of construction (bays 1 - 4):

The hall is completed in the raw, the steel structure mounted and roofed. The 126 Mp-hangar crane functions and work for installations such as heating, cooling, plumbing and electrical fittings has been taken up.

2. Stage of construction (bays 5 - 7):
   The underground work is completed and the depth of the testing cell is -21 m, the work around the testing cell is nearly finished.

## 6. Results

## 7. Next Steps

Mounting of the 10 000 Mp-Machine is being continued including the total servo hydraulic, electrical equipment and electronic. Calibration and putting the machine into operation will be in Autumn 1978.

The testing cell with fundament for the full size vessel is to be armoured and reinforced with concrete, completion of the shell construction of the second stage, mounting of the steel construction and roof in the second stage, external panelling of the complete hall and the completion of the installation still to be carried out.

## 8. Relation with other Projects

RS 245 Research Programme Full Size Vessel
RS 304 Research Programme of Safety of Components.

## 9. References

Quarterly Reports RS 101/RS 101 A in GRS Research Reports, period reported on July 76 - December 77 (German).

## 10. Degree of Availability

Upon request at BMFT.

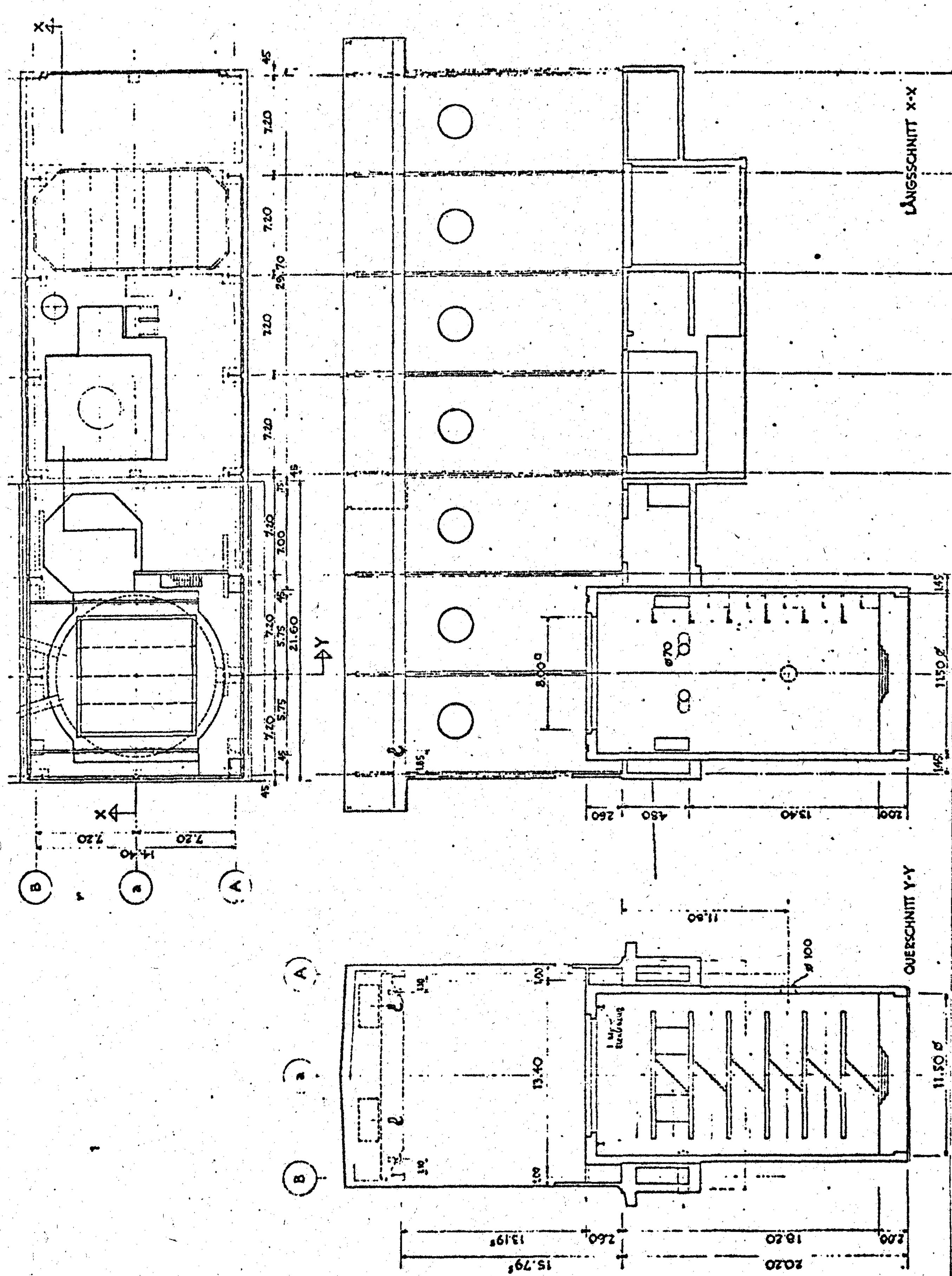


Fig. 1: Foundation plan and section of the complete testing hall

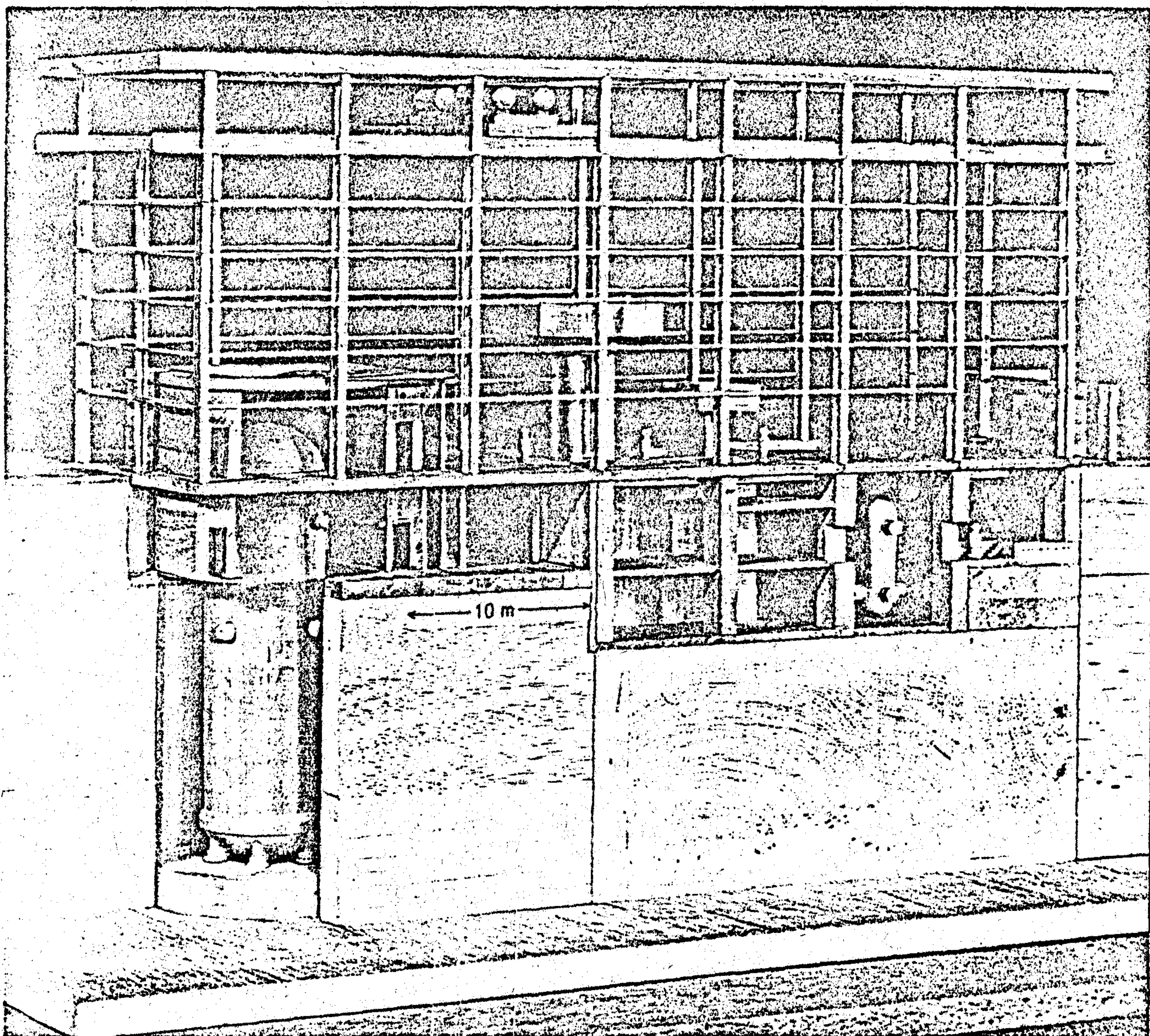


Fig. 2: Model of testing hall (section) with 100 MN-Machine and Full Size Vessel

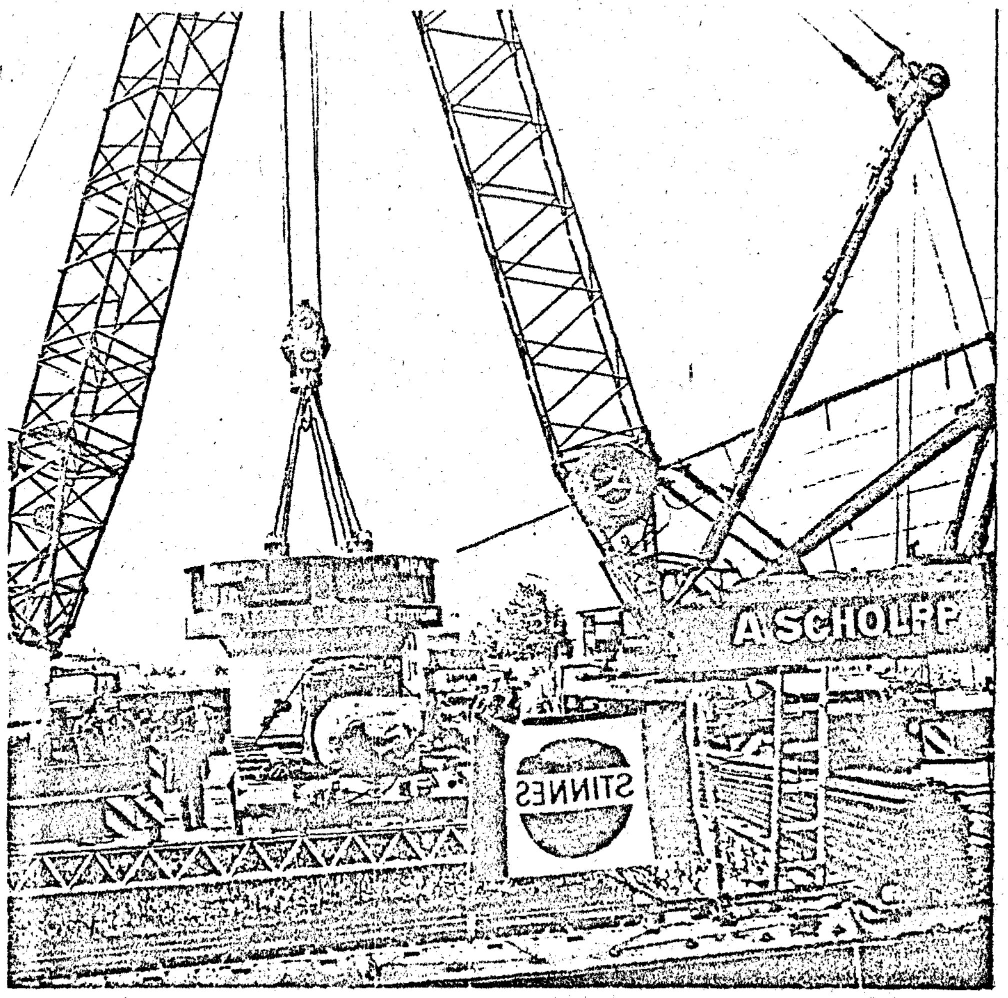


Fig. 3 Heavy transport in the harbour of Heilbronn (cylinder 135 Mp)

11 54

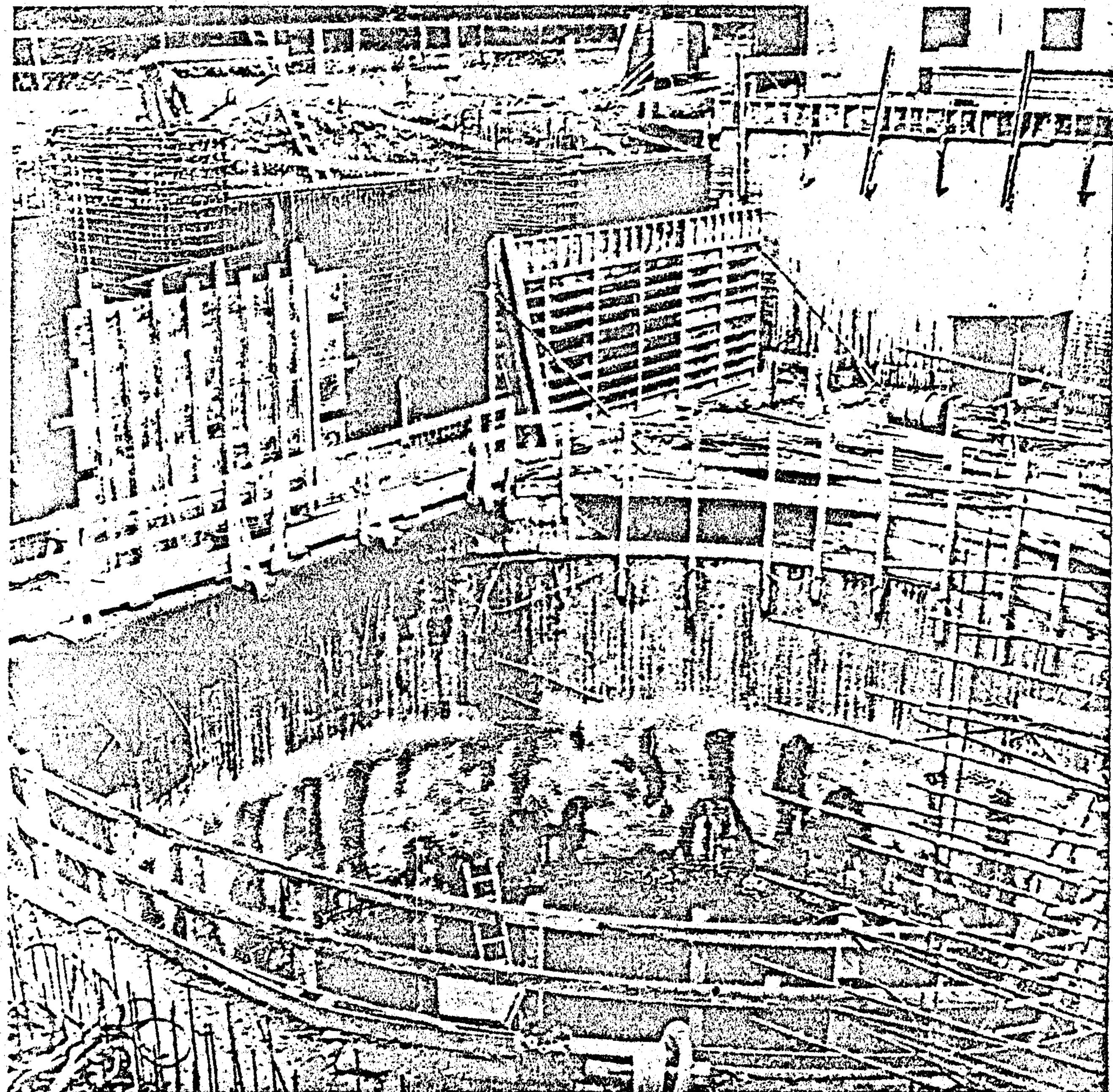

Fig. 4 Testing cell under construction

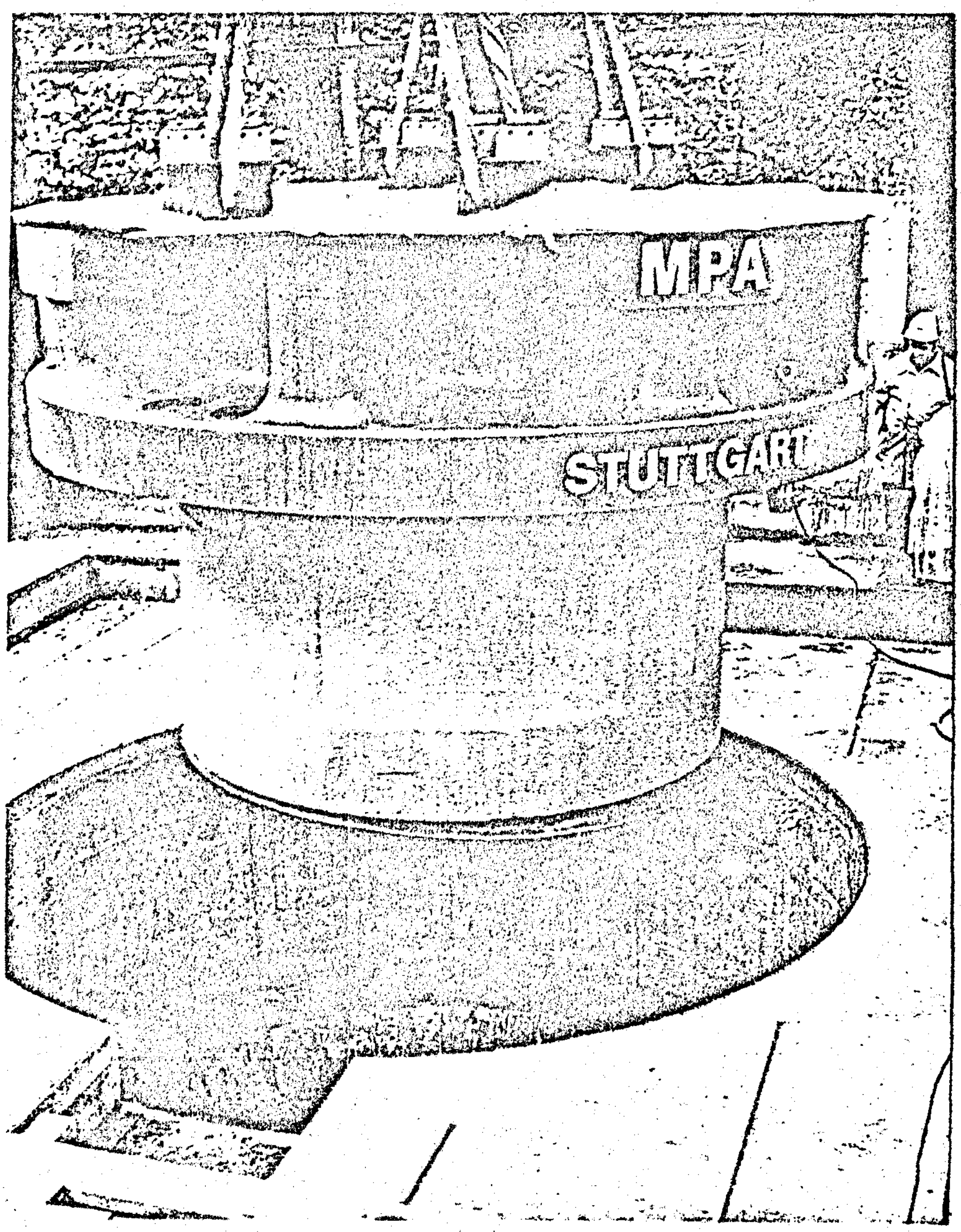


Fig. 5: Mounting of the cylinder (135 Mp) into the finished fundament (600 Mp)

1157

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. 77 - 31. 12. 77 | 11.2.1 | RS 207 |
| **Vorhaben/Project Title** | | **Land/Country** |
| Erweiterung des Einsatzbereiches kobaltfreier Werkstoffe für Reib- und Verschleißbeanspruchung im NDES von Leichtwasserreaktoren<br><br>Extension of the Use of Cobalt-Free Materials for Resistance of Friction, Wear and Corrosion in the Nuclear Steam Supply System (NSSS) of LWR's | | FRG<br>**Fördernde Institution/Sponsor**<br>BMFT<br>**Auftragnehmer/Contractor**<br>KRAFTWERK UNION AG<br>Reaktortechnik<br><br>R 41, Erlangen |
| **Arbeitsbeginn/Initiated** | **Arbeitsende/Completed** | **Leiter des Vorhabens/Project Leader** |
| 1. 5. 76 | 30. 9. 79 | L. Stieding |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | 31. 12. 77 | 322.698,-- DM |

1. General Aim

Reduction of contamination of the NSSS by products of corrosion and wear, which together with the coolant pass through the reactor core and thereby become activated.

2. Particular Objectives

a) Selection and (eventually in cooperation with the manufacturers) development of Co-free plating materials, which can also be exposed to impact stress conditions and which can be worked more easily than Everit 55. Investigations of the behaviour of these materials and conditions of sliding, impact and corrosion.

b) Selection and evaluation of plasma-sprayed materials with satisfactory corrosion and wear properties under PWR and BWR conditions.

c) Selection and evaluation of stainless steels which are expected to show better friction and wear behaviour than material No. 1.4550.

3. Research Program

3.1 Selection of test materials:
pre-selection, acquisition, fabrication of samples, adhesion tests, friction and wear tests at 20 °C.

3.2 Corrosion tests:
fabrication of samples, corrosion tests at 320 °C, tests

for resistance to integranular corrosion.

3.3 Friction and wear tests:
fabrication of samples, tests at 20 °C and 300 °C.

3.4 Mechanical tests:
fabrication of samples, adhesion tests, brittleness test

3.5 Evaluation of test results.

4. Test Facilities

The required test facilities have either been used in previous tests or have recently been developed (hydraulic apparatus for RT tests; stand for high-temperature wear tests at 300 °C with simultaneous influence of corrosion; static and refreshing autoclaves for long term corrosion tests) and are available for the above described tests.

5. Progress to Date

5.1 Plating materials

Friction tests were run with the materials

- Pantanax 25/0 Mo-G
- VEW alloys A-H
- Tribaloy T 700
- Nicrend C

Corrosion tests were started with Pantanax 25/0 Mo-G and the VEW alloys in PWR primary water (350 °C, 1500 ppm B)

5.2 Spray layers

A literature study was started, the chances of detonation- and plasmaspray layers were analyzed. UC-probes were worked out.

5.3 Structure materials

A literature study was started, a test program was worked out. Friction probes were assembled and tests were run in water at 20 °C with

- Material No. 1.4550 (for comparison)
- Nitronic 32
- Nitronic 60
- AT 18 m 9 h - WK
- Amagnit 3974-WK

6. Results

6.1 Plating materials

The wear mechanism Pantanax 25/0 Mo-G was a pure abrasion-wear process.
The alloys A-H were not suited as plate materials because the surfaces were roughened during the friction tests.

6.2 Spray-layers

The test program has been completed, the test materials are fixed. The producers of spray-layers were contacted.

6.3 Structure materials

First wear and friction tests with Nitronic 60 were successfull.

7. Next Steps

Wear tests with plate materials, corrosion tests with detonation- and plasma-sprayers and wear and friction tests with structure materials will be continued.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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Classification: 11.2.1.

| | |
|---|---|
| Title: Dynamisk brudmekanik | Country: DENMARK |
| | Sponsor: Risø National Lab. |
| Title: Dynamic Fracture Mechanics<br>Crack propagation in inhomogenous steel | Organization: Risø National Laboratory |
| Initiated date: 1973 Completed date:<br>Status: Some basic investigations finished | Scientists:<br>C. Debel |

1. General aim: To develop fracture mechanics approaches to evaluate crack propagation in inhomogenous steel in which the fracture toughness varies along the path of the crack.

2. Particular objectives: Welded connections in steel structures constitute areas within which the fracture toughness varies considerably. Welded connections are furthermore the most likely sites for defects able to cause failure of the steel structure. A particular aim of this project is to evaluate the critical size of a brittle volume as related to the general fracture toughness of the base metal.

3. Experimental facilities and programme: Tensile testing, fast bending and instrumented impact testing machines are available. A mathematical analysis has been made and an experimental investigation has been finished using duplex DCB test pieces of a type comparable to those used by G. Hahn, et.al., Battelle, Columbus, USA.

4. Project status: Danish reports on the above mentioned activities have been worked out and English versions are under way. More experiments are needed before significant conclusions can be drawn.

5. Next steps: Experiments to be continued.

6. Relation with other projects: Experiments will be designed to be carried out in the facility resulting from the project Pressure Testing Facility.

7. Reference documents: RISØ-M-1842. Carsten Engel: "Energetiske og kinetiske overvejelser vedrørende revners bevægelse over fasegrænser". RISØ-M-1897, C.P. Debel: "Dynamisk Brudmekanik. Licentiat-rapport".

8. Degree of availability: No limitations.

| 141-1-03/4111-65 | | 11.2.1 |
|---|---|---|
| **Titre**<br>Comportement à l'irradiation des aciers Creusot-Loire pour cuves PWR. | | **Pays**<br>FRANCE<br>**Organisme directeur**<br>CEA / DSN |
| **Titre (anglais)**<br>Research programme on irradiation embrittlement of pressure vessel steels | | **Organisme exécuteur**<br>CEA-DTECH<br>**Responsable**<br>-DSN/FAR<br>-DTECH |
| **Date de démarrage** 1/01/76 | **Etat actuel**<br>en cours | **Scientifiques**<br>DTECH |
| **Date prévue d'achèvement** 31/12/79 | **Dernière mise à jour** 1/78 | |

1 - Objectif général :

Le programme a pour but de déterminer le comportement sous irradiation neutronique de l'acier Creusot-Loire (type A508 C13) utilisé pour la fabrication des cuves PWR du programme nucléaire français.

2 - Objectifs particuliers :

Le programme comporte essentiellement :

- Les mesures de KCV et NDT
- Les mesures de $K_1d$ sur petites éprouvettes
- Les mesures de $J_1c$
- Les mesures de $K_1c$ sur grosses éprouvettes
- L'étude de la restauration

.../

3 - Installations expérimentales et programme :

Les irradiations sont effectuées dans le réacteur expérimental TRITON (CEN - Fontenay-aux-Roses)

4 - Etat de l'étude :

A) Le programme mis en oeuvre comporte d'une part un ensemble d'expériences destinées à apprécier d'une manière statistique le comportement sous irradiation des aciers de fabrication française et d'autre part, une série d'expériences spécifiques tendant à préciser divers points particuliers : effet de la composition chimique, effet du flux neutronique et de la température, effet du recuit sur le durcissement par irradiation.

Cette étude a permis de dégager les points suivants :

- L'abaissement des teneurs en éléments résiduels (P et Cu) permet de réduire la sensibilité à la fragilisation sous irradiation.

- L'addition de nickel au-dessus de 0,8 % introduit une sensibilité supplémentaire à la fragilisation par irradiation, tout en étant bénéfique hors irradiation pour l'amélioration de la ténacité et de l'homogénéité des propriétés mécaniques.

- Le décalage de la température de transition peut être prédit de manière sûre par la formule du Regulatory Guide 1.99 si la teneur en nickel reste faible (le décalage de la température de transition et le durcissement sont bien corrélés).

- La restauration du durcissement par irradiation à 290°C n'est sensible qu'au dessus de 350°C. Une restauration complète ne semble possible qu'au dessus de 400°C.

L'ensemble des résultats obtenus est donné dans la référence (1).

B) On a montré par ailleurs que de petites éprouvettes pouvaient être utilisées avec succès pour mesurer la ténacité sous irradiation des aciers de cuve. En particulier les essais sur éprouvettes CHARPY préfissurées donnent une mesure de $K_I d$ qui peut être considérée comme une borne inférieure du $K_I c$ du matériau irradié.
On a ainsi lancé en 1977 une campagne d'irradiation sur différents aciers de cuve (Fessenheim 1, 2, Bugey 3) ainsi qu'une soudure de Fessenheim 1 pour déterminer les courbes de transitoire de $K_I d$. Les résultats ont été acquis en 1977 mais le dépouillement complet des essais ne sera terminé que courant 1978.

.../

## 5 — Prochaines étapes :

Outre le dépouillement des essais $K_1d$, il est également prévu en 1978 le dépouillement des mesures de $J_1c$ sur matériaux irradiés avec éprouvettes de différentes tailles.

## 7 - Documents de référence :

(1) "Etude de la fragilisation par irradiation des aciers de cuve type A 508 Cl3 et A 533 Cl1. Influence de la composition chimique ",
P.PETREQUIN et P.SOULAT -
Colloque international de l'AIEA - 10-14 Octobre 1977.

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| 141-1 - 04/4111-66 | | 11.2.1 |
|---|---|---|
| Titre<br><br>Propagation de fissures de fatigue sur les matériaux de structure irradiés. | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br><br>Fatigue crack propagation on irradiated materials | | Organisme exécuteur<br>CEA/DTECH<br><br>Responsable<br><br>DSN-SETSSR-Fontenay |
| Date de démarrage 1/01/76 | Etat actuel<br>en cours | Scientifiques<br><br>(DTECH-SRMA) |
| Date prévue d'achèvement 31/12/79 | Dernière mise à jour 1/1/78 | |

1 - Objectif général :

Le programme a pour but de mettre au point des méthodes de mesure de vitesse de propagation de fissure de fatigue sur métaux irradiés.

2 - Objectifs particuliers :

Les essais sont plus spécialement exécutés sur les aciers des cuves pour réacteurs PWR.

3 - Installations expérimentales et programme :

La machine de fatigue en cellule commandée en 1974 a été opérationnelle en Septembre 1976 sur matériaux irradiés. En 1975, l'irradiation d'aciers de cuve a été lancée et leur exploitation à température ambiante a eu lieu au 2ème semestre 1976.

.../

4 - Etat de l'étude :

1) Avancement à ce jour

En 1977, la mise au point des essais de propagation de fissures à chaud (330°C) a été entreprise et une campagne d'irradiation a été lancée.

5 - Prochaines étapes :

Les essais proprement dits seront effectués courant 1978.

| 141-1 -06/4111-6-13 | | 11.2.1 |
|---|---|---|
| **Titre** Comportement des aciers pour cuves : tenue à l'irradiation - Programme coordonné de l'AIEA. | | **Pays** FRANCE<br>**Organisme directeur** IPSN/DSN |
| **Titre (anglais)** Research programme on irradiation embrittlement of pressure vessel steels. AIEA Coordinated Programme | | **Organisme exécuteur** CEA/DTECH<br>**Responsable** B.BARRACHIN DSN-SETSSR-Fontenay |
| **Date de démarrage** 1/01/78 | **Etat actuel** démarrage | **Scientifiques** |
| **Date prévue d'achèvement** 31/12/80 | **Dernière mise à jour** 1/01/78 | |

1 - Objectif général :

L'Agence Internationale pour l'Energie Atomique (AIEA) organise un programme coordonné d'irradiation d'aciers de cuves de réacteurs PWR.
Il est prévu d'examiner dans au moins 5 pays le comportement à l'irradiation de 2 aciers français, 2 aciers japonais, et 2 soudures.

2 - Objectifs particuliers :

- il est prévu tout d'abord de fournir et caractériser avant irradiation un acier et un joint soudé d'origine française qui seront irradiés dans les différents pays.
- les aciers français, les aciers japonais et les soudures seront ensuite caractérisés du point de vue de l'irradiation (Charpy V, Traction, $J_1c$ , Charpy préfissuré)

3 - Installations expérimentales :

Les irradiations seront effectuées dans le réacteur expérimental de TRITON (CEN.Fontenay)

…/

4 - Etat de l'étude :

Fin 1977, une réunion organisée par l'Agence a permis de préciser le programme, à savoir aciers proposés (tableau 1), participants (tableau 2), et programme CEA (tableau 3);

5 - Prochaines étapes :

voir tableau 3

| Repères | Dénomination | Produit | Origine | Nombre de blocs proposés x | Nombre de blocs demandés x |
|---|---|---|---|---|---|
| 1<br>LJ | A 533 B<br>cl. 1 | Tôle ép. 250 mm | Nippon Steel<br>Corporation | 46 | 10 |
| 2<br>FJ | A 508<br>cl. 3 | Morceaux de virole<br>Ø 5550 mm ép. 290 mm | Japan Steel Works | 46 | 26 |
| 3<br>LF | A 533 B<br>cl. 1 | Tôle ép. 320 mm | Marrel Frères | 50 | 15 |
| 4<br>FF | A 508<br>cl. 3 | Débouchure de virole<br>Ø ∼400 mm ép. 240 mm | Framatome | 34<br>+ matière en stock | 24 |
| 5<br>SJ | Joint soudé | ép. 250 mm<br>Fil : Kobe steel | Mitsubishi<br>Heavy Industries | 18 | 11 |
| 6<br>SF | Joint soudé | ép. 240 mm<br>longueur 3500 mm | Framatome xx | 18 - 20 | 12 |
| 7<br>SA | Joint soudé | non précisé | KWU | 18 | 16 |

x Blocs 150 x 150 mm x épaisseur

xx Fournisseur proposé : à confirmer

TABLEAU I - ACIERS PROPOSES

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| Matériau / Pays / Organisme / Rep. | A | A1 | B | C | D | E | F | G | H | I | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | Jaeri<br>Japon | Risø<br>Dane-<br>mark | GKSS<br>RFA | NRL<br>USA | Skoda<br>Tchécos-<br>lovaquie | CEA<br>Saclay<br>France | Jülich<br>RFA | KWU<br>RFA | AB<br>Atome<br>Suède | BARC<br>Trombay<br>Inde | Total |
| 1 | 6 | 1 | | 1 | 2 | 1 | 5 | 1 | | 2 | 19 |
| 2 | 6 | 1 | 4 | 2 | 2 | 1 | 5 | 2 | 1 | 2 | 6 |
| 3 | 2 | 1 | | 1 | 1 | 2 | 5 | 1 | | 2 | 15 |
| 4 | 2 | 1 | 4 | 2 | 3 | 2 | 5 | 2 | 1 | 2 | 24 |
| 5 | 3 | 1 | | 2 | 1 | 1 | | 1 | | 2 | 11 |
| 6 | 2 | 1 | | 2 | 1 | 2 | | 1 | 1 | 2 | 12 |
| 7 | 2 | 1 | 4 | 1 | 3 | | | 2 | 1 | 2 | 16 |
| Référence | 1 | | 1 | | | 1 | | 1 | | 1 | 5 |

TABLEAU II - Participants au programme d'irradiation - Nombre de blocs demandés

| Repère acier \ Type d'essai | Charpy V | | Traction | | $J_{IC}$ | | Charpy préfissuré | |
|---|---|---|---|---|---|---|---|---|
| | $\phi_1$ | $\phi_2$ | $\phi_1$ | $\phi_2$ | $\phi_1$ | $\phi_2$ | $\phi_1$ | $\phi_2$ |
| 1 LJ | E | E | E | E | | | | |
| 2 FJ | E | E | E | E | | | | |
| 3 LF | E | E | E | E | E | E | (E) | (E) |
| 4 FF | E | E | E | E | E | E | (E) | (E) |
| 5 SJ | (E) | E | (E) | E | | | | |
| 6 SF | E | E | E | E | E | E | (E) | (E) |
| 7 SA | (E) | | (E) | | | | | |
| Ref | E | | E | | | | | |

Température d'irradiation : 290 °C

$\phi_1$ = 3 à 4 $10^{19}$ n/cm$^2$ (E > 1 MeV)

$\phi_2$ = 7 à 9 $10^{19}$ n/cm$^2$ (E > 1 MeV)

E : prévu dans la première phase (avant 1980) (E) programme ultérieur

TABLEAU III - Programme proposé au CEA

| 145-2 -02 | | 11.2.1 |
|---|---|---|
| Titre<br>EDGAR ZY : comportement des gaines en zircaloy au cours d'un accident de dépressurisation. | | Pays<br>FRANCE<br>Organisme directeur<br>CEA /DgCS .- EDF/SEPTEN |
| Titre (anglais)<br>EDGAR ZY : zircaloy cladding behaviour under loss of coolant accidental conditions. | | Organisme exécuteur<br>CEA/DMECN/DTech<br>Responsable<br>DTech - Saclay |
| ...ce de démarrage 1/1/74 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 12/80 | Dernière mise à jour 12/77 | |

1 - Objectif général :

Déterminer les paramètres d'éclatement, vitesse de déformation et épaisseur de la couche oxydée des gaines en zircaloy des éléments combustibles des réacteurs à eau sous pression, lorsqu'elles sont soumises à des variations transitoires de température et de pression représentatives d'un accident de perte de caloporteur.

2 - Objectifs particuliers :

1 - Vérifier que le comportement des gaines permet le respect des critères de température et d'épaisseur oxydée maximales imposés par les organismes de sûreté.

2 - Elaborer un modèle de déformation de gaine utilisable dans les codes CUPIDON et DEMETER.

3 - Contribuer à la qualification de ces codes.

.../

3 - Installations expérimentales et programme :

1 - Pour les essais réalisés jusqu'à présent, des gaines neuves ont été utilisées exclusivement, dans une installation du DTech-SRMA à Saclay. Cette installation comporte un chauffage du tube par effet Joule et une pressurisation interne programmables. Les essais peuvent être réalisés sous vide, ou sous une atmosphère externe inerte, ou bien de vapeur d'eau.

2 - Après étude, un dispositif analogue est en cours de montage dans une cellule chaude du DTech-SECS (LECI) à Saclay en vue de l'étude complémentaire du comportement de gaines préirradiées.

4 - Etat de l'étude :

1) Avancement à ce jour

1 - La première phase du programme, ayant pour objet de vérifier que les caractéristiques de déformation à pression et vitesse de chauffe constantes sont en bon accord avec les résultats publiés à l'étranger, est terminée.

2 - Des essais à vitesses de pressurisation et de chauffe programmées d'après les résultats de calculs de thermohydraulique au cours d'un accident et sous vapeur d'eau ont été réalisés, sans bouclage entre calcul et expérience.

3 - Un modèle de déformation de gaine utilisable dans le code CUPIDON a été élaboré.

2) Résultats essentiels

Les résultats obtenus ont été généralisés sous forme :

1 - d'abaques donnant le temps de fluage nécessaire pour atteindre une déformation donnée, dans des conditions de pression initiale et de vitesse de chauffage données.

2 - de courbes maitresses de déformation en fonction de la contrainte pour des températures et vitesses de chauffage données.

3 - d'une relation donnant la vitesse instantannée de déformation en fonction de la contrainte , de la température et de la vitesse de variation de la température (modèle destiné à CUPIDON)

.../

5 - Prochaines étapes :

1 - Reprise des essais à vitesses de pressurisation et de chauffe programmées, avec bouclage entre le calcul et l'expérience

2 - Utilisation du dispositif expérimental EDGAR comme l'un des moyens de qualification du code CUPIDON et, ultérieurement, du code DEMETER.

3 - Vérification de l'influence de l'irradiation sur le comportement des gaines et, éventuellement, vérification de l'influence de la présence de pastilles dans la gaine. En 1978, mise au point du dispositif en cellule chaude .

6 - Relation avec d'autres études :

Cette étude est en relation directe, d'une part avec l'analyse de sûreté des réacteurs à eau sous pression (vérification du respect des critères de sûreté) et d'autre part avec l'élaboration de modèles de déformation de gaine pour les codes de calcul (CUPIDON , DEMETER) et avec la qualification de ces derniers. Elle doit, d'autre part, apporter une contribution à l'exploitation des résultats du programme PHEBUS.

7 - Documents de référence :

Les principaux résultats obtenus jusqu'à présent ont été publiés à la réunion de spécialistes du CSNI à SPATIND (Norvège) en septembre 1976 : Comportement du gainage en zircaloy des éléments combustibles des réacteurs à eau sous pression pendant un accident de refroidissement. Programme EDGAR.
Note DTech - RMA/76-710 de septembre 1976).

| 145-3 -01/4111-104 | | 11.2.1 |
|---|---|---|
| Titre<br>Comportement des lignes de tuyauteries lors d'un accident. | | Pays<br>FRANCE<br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br>Behaviour of pipings in the event of an accident | | Organisme exécuteur<br>CEA/DEMT<br>Responsable<br>(DSN-SETSSR)<br>(DEMT) |
| Date de démarrage 1/03/77 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 31/12/82 | Dernière mise à jour 1.12.77 | |

1 - Objectif général :

Le programme a pour but d'étudier le comportement des tuyauteries en cas de rupture ( eau pressurisée ). Il concerne essentiellement les tuyauteries de petit diamètre appartenant aux circuits auxiliaires . Les essais sont réalisés dans des conditions proches de l'échelle réelle.

2 - Objectifs particuliers :

Le programme comporte essentiellement, outre la mise au point du dispositif expérimental :
- La détermination des critères de fouettement.
- La mesure des efforts de réaction.
- La prise en compte des interactions possibles.
- Le dimensionnement des supportages et autres limiteurs de débattement.
- La tenue des organes d'isolement.

3 - Installations expérimentales et programme :

Boucle HUREPOIX (CEN-SACLAY).

.../

## 4 - Etat de l'étude :

Actuellement seuls quelques essais ont été réalisés. Ils avaient pour but essentiel de mettre au point le dispositif de rupture brutale de la tuyauterie, et les chaines de mesures associées.

## 5 - Prochaines étapes :

Détermination des critères (fin 78)
Dimensionnement des supportages et comportement des organes d'isolement (Fin 79)

| PROJECT TITLE : Fracture Mechanics for Ductile Materials | CLASSIFICATION 11.2:1 |
|---|---|
| SPONSORING COUNTRY : ITALY | ORGANISATION : PISA - UNIVERSITY |
| DATE INITIATED : 1968<br>DATE COMPLETED : 1977 | PROJECT LEADER : C. CARMIGNANI |

Description :

The aim of this research program is to evaluate the possibility of using linear elastic fracture mechanics concepts to forecast the starting of crack propagation in a ductile material.

In the first phase it was developed a theoretical work to evaluate the elastic stress field around the tip of the crack (Ref. 1.2). After, several cylindrical pressure vessels of diameter 100 ÷ 200 mm, thickness 5 ÷ 6 mm, with on axial through the thickness cracks of length 60 ÷ 100 mm, were bursted by internal pressure (Ref. 3.4.5). The results of this experimental work demonstrated the possibility of using cylindrical pressure vessels, like those above mentioned, to determine the critical parameters ($K_c$ or COD) of the material.

Now is under execution an experimental work devoted to evaluate size effect on the determination of the critical parameter with the method above indicated.

Facilities used are a pressurization system with a manual pump for bursting test at room temperature, using oil, and an electric pump for bursting test at low temperature (-170 °C), using liquid nitrogen.

The research program was initiated with a research contract with the CNEN and now it is presented with an analogy contract with the CNR.

REFERENCES

1. C. CARMIGNANI, A. DEL PUGLIA
Applicazioni di metodi di calcolo elettronico all'analisi del comportamento elastico di piastre rettangolari contenenti una fessura passante centrale, caricate normalmente al loro piano medio.
Atti Ist. Mecc. Appl. - Univ. PISA, n° 11 - Anno Acc. 1968-'69.

2. C. CARMIGNANI, A. CELLA
Elastic analysis of cracked thin shells by the finite element method.
Proc. 1st SMIRT, Berlin, 1971, vol. 5 pp. 57÷76.

3. C. CARMIGNANI
Ricerca teorica e sperimentale sulla propagazione delle rotture nei recipienti in pressione.
Atti del Convegno Nucleare di PISA, vol. 1, CNEN, 1971, pp. 279÷319.

4. C. CARMIGNANI, S. REALE
Elaborazione dei risultati della ricerca sperimentale sulla propagazione delle rotture nei recipienti in pressione.
Atti Istituto Impianti Nucleari - Univ. PISA. RP 153(73), 1973.

5. C. CARMIGNANI, P. CIBECCHINI, S. REALE
Caratterizzazione di un acciaio al Nichel per impieghi a bassa temperatura con i criteri della meccanica della frattura.
Proc. 5th Int. Conf. on Exp. Stress Analysis, Udine, 1974, paper 18 pp. 2.5 ÷ 2.12.

| | |
|---|---|
| TNO - Metaalinstituut | CLASSIFICATION : 11.2.1 |
| **TITLE :** Literatuurstudie:<br>(1) Invloed van omgevingscondities op vermoeiings-scheurgroei.<br>(2) Invloed van veroudering op de mechanische materiaaleigenschappen | COUNTRY: THE NETHERLANDS |
| **TITLE** (ENGLISH LANGUAGE): Literature study:<br>(1) Influence of Environment on Fatigue crackgrowth.<br>(2) Influence of aging on fracture related material properties. | SPONSOR : Ministry of Social Affairs<br>ORGANIZATION : TNO-Metaalinstituut<br>PROJECTLEADER : H.J.M. van Rongen |
| INITIATED : June 1974 — LAST UPDATING : May 1978 | SCIENTISTS : H.J.M. van Rongen, G.H.G. Vaessen |
| STATUS : completion — COMPLETED : Oct. 1977 | |

General aim

To improve the understanding of material behaviour during lifetime of nuclear vessels and other pressurized components as influenced by aquous environment and aging effects. A "state of the art" report will be submitted.

Particulare objectives

To review the knowledge in the areas

. influence of reactor coolant on fatigue crackgrowth in LWR pressure vessels, with particular attention to chemical composition of the coolant, type of pressure vessel steel, frequency, amplitude and threshold phenomena. Interpretation will be in fracture mechanics terminology.
. phenomena which deteriorate the fatigue and fracture related material parameters of nuclear grade pressure vessel steels, such as thermal aging and high strain aging.

Experimental facilities and program: Not applicable

Project status

Progress to date: final report submitted on Oct. 21 1977.

Relation with other projects: -

Reference documents:

1. Embrittling/strengthening phenomena
2. Environmental effects on fatigue crack growth in relation to nuclear reactor vessel steels under relevant conditions by H.J.M. van Rongen and G.H.G. Vaessen. nr. 77M/93/09913/ROH/VAE/KAG/ROS.

Degree of availability:

Through the Ministry of Social Affairs, C.R.V., Postbus 69, Voorburg, The Netherlands.

Budget: HFL. 50.000,--

Personnel: 2 scientists.

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX

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| NIL-MI-TNO | | CLASSIFICATION : 11.2.1/ 11.2.3 |
|---|---|---|
| **TITLE :** Literatuurstudie van materiaalparameters die nodig zijn bij de interpretatie van defect-localisatie met AE-apparatuur. | | COUNTRY: THE NETHERLANDS |
| | | SPONSOR : Ministry of Social Affairs and others |
| **TITLE (ENGLISH LANGUAGE):** Literature survey of material parameters which are necessary for the diagnosis of defects localized with AE techniques. | | ORGANIZATION : NIL-MI-TNO |
| | | PROJECTLEADER : Kloots |
| INITIATED : Sept. 1974 | LAST UPDATING : May 1978 | SCIENTISTS : Boerstoel v.d. Brink |
| STATUS : Completed | COMPLETED : June 1978 | |

General aim

This particular survey is a preliminary study of effects to be included in an extensive study which will be initiated in the future and which can be characterized as follows:

"Evaluation of the practical application of Acoustical Emission (AE) techniques during construction, testing and operation of welded constructions, in particular pressurized components to be used in the energy and process industry in order to improve safety, reliability and economic construction of components".

Optimum utilization of AE-apparatus for defect localization and diagnosis is only possible if sufficient data on the AE behaviour of constructural material are available (AE material parameters). This study comprises the use of AE material parameters for nuclear vessels and their application to the diagnosis of defects localized in experiments.

Particular Objectives

Study and inventarization of AE research in literature.
Visits to industry and institutes in the Netherlands.

Experimental facilities and program: none

Project status: completed

Next steps: -

Relation with other projects: see under General aim.

Reference documents: NIL-lastechniek, 40e jaargang, no. 5, mei 1974.
NIL-lastechniek "Literatuurstudie betreffende de toepassing van akoestische emissietechniek bij materiaalonderzoek aan drukvatstalen" samengesteld door Dr. B.M. Boerstoel en Dr. ir. S.H. van den Brink, rapport nr. 77M/93/03465/BRI/KAG

Degree of availability: Through Nederlands Instituut voor Lastechniek, Den Haag.

Budget: Approx. Hfl. 30.000,--

Additional information:

Survey has been executed and completed by Metaal Instituut TNO.

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX.

Classification

11.2.1

| | |
|---|---|
| Title 1<br>METAL FRACTURE (CRACK GROWTH) | COUNTRY<br>UNITED KINGDOM |
| | SPONSOR UKAEA |
| | ORGANIZATION<br>RFL SPRINGFIELDS |
| Title 2 | Project Leader<br>DR B TOMKINS |
| Initiated 1971    Completed :<br>Status :    Last updating | Scientists: |

Description:

1. General Aim

To recognise and assess modes of failure of nuclear reactor pressure vessels and other circuit components.

2. Particular Objectives

To examine the possibility of component defects growing to a critical size as a result of reactor transients.

3. Experimental Facilities and Programme

A number of special purpose machines have been assembled, which permit specimens of about 1 in. section to be stress cycled and thermally cycled under well-controlled conditions.

4. Project Status

a) Crack growth under thermal shock loading. Crack propagation from short initial surface defects has been monitored in specimens which undergo repeated surface cooling shocks of between 100°C and 250°C. The crack growth rates and varying degrees of crack penetration into the material have been analysed using the basic elastic and elastic-plastic crack growth data derived under simple push-pull conditions in an earlier part of the programme. The tests are being carried out on types 304 and 316 stainless steel. The information from these tests is being used to assess the growth of under-clad cracks as a result of service transients and the behaviour of defects in reactor circuits. Further tests will examine the growth of cracks in combined thermal and mechanical loading situations and a new rig will impose heating shocks on to cracked specimens.

b) Fatigue crack growth in pressure vessel steel. This work programme has now been extended to examine the growth of fatigue cracks at low frequency in A533-B steel in a simulated reactor water environment. Current tests are at ambient temperature and pressure. In particular the effects of mean stress and varying frequency are being examined. The results of these tests are

contd......

1186

Classification

| | |
|---|---|
| Title 1 | COUNTRY |
| | SPONSOR |
| | ORGANIZATION |
| Title 2 | Project Leader |
| Initiated  Completed : | Scientists: |
| Status :  Last updating | |

Description:

enabling corrosion and fatigue mechanics effects to be correlated with crack opening modes. Further work will attempt to define bounds on the accelerations in growth rate produced.

Reference Documents

Internal documents.

| | | |
|---|---|---|
| | Classification 11:2.1 | |
| Title 1<br>METAL FRACTURE (STABILITY) | | COUNTRY UNITED KINGDOM<br>SPONSOR UKAEA<br>ORGANIZATION RFL SPRINGFIELDS |
| Title 2 | | Project Leader<br>DR B TOMKINS |
| Initiated 1971 | Completed : 1976 | Scientists: |
| Status : | Last updating | |

Description:

1. General Aim

To recognise and assess modes of failure of nuclear reactor pressure vessels.

2. Particular Objectives

To develop a way of producing a sharp crack in a component, to enable the critical crack length to be measured.

3. Experimental Facilities and Programme

A high pressure technology exists at Springfields; steady pressures of 100,000 psi (6000 bar) may be employed and cycles of about half that magnitude. This has been applied to locally fatigue crack the base of a notch and thus produce well-defined crack starter conditions.

Reference Documents

Internal documents.

Classification

11:2.1

| Title 1 | COUNTRY UNITED KINGDOM |
|---|---|
| MECHANICS OF PLANE STRAIN FRACTURE | SPONSOR UKAEA |
| | ORGANIZATION RFL SPRINGFIELDS |
| Title 2 | Project Leader |
| | DR B TOMKINS |
| Initiated 1972   Completed : | Scientists: |
| Status :   Last updating | |

Description:

1. General Aim

To recognise and assess modes of failure of nuclear reactor pressure vessels.

2. Particular Objectives

To measure the deformation in a thick cracked plate up to the point of fracture.

3. Experimental Facilities and Programme

A rig has been constructed in which a high pressure (circa 3000 bar) may be applied within a crack in the plate.

The use of internal pressurisation means that fracture criteria in thick section material can be studied under well defined boundary conditions. Direct surface strain measurement around the crack tip will be done and fractography employed to stress fracture development prior to fast fracture propagation.

Reference Documents

Internal documents.

7191

| | |
|---|---|
| | Classification<br>11.2.1 |
| Title 1<br>VESSEL FAILURE TESTS | COUNTRY<br>UNITED KINGDOM<br>SPONSOR UKAEA<br>ORGANIZATION<br>REML RISLEY |
| Title 2 | Project Leader<br>DR A COWAN |
| Initiated 1969   Completed :<br>Status :   Last updating | Scientists: |

Description

1. General Aim

To quantify critical crack length for use in decisions arising in inspection.

2. Particular Objectives

To compare $K_{IC}$ and COD correlations for failure of intermediate-size pressure vessels above valid $K_{IC}$ critical temperatures.

3. Experimental Facilities and Programme

The test vessels are 5' dia and 12' long. Thicknesses to date, 1" and 3". Currently defects (partial and full thickness) are placed in the membrane regions.

4. Project Status

Investigation of the factors governing cyclic crack growth and fracture in pressure vessel steels is continuing. Three tests have been completed in a series on a 3" thick x 5' dia vessel to investigate the effect of plate thickness. The tests were made at temperatures of 84 and 10°C corresponding to the whole of the ductile/brittle transition range. Relationships between small scale fracture toughness test pieces cut from the vessels and vessel failure conditions are being investigated.

Reference Documents

Internal documents.

Classification

11.2.1

| Title 1 | COUNTRY UNITED KINGDOM |
|---|---|
| FRACTURE BEHAVIOUR OF STRUCTURAL STEELS | SPONSOR UKAEA |
| | ORGANIZATION REML RISLEY |
| Title 2 | Project Leader DR A COWAN |
| Initiated 1969 Completed : | Scientists: |
| Status : Last updating | |

Description

1. General Aim

To apply fracture mechanics concepts for assessing structural integrity of pressure retaining components.

2. Particular Objectives

i to investigate relationships between small scale fracture toughness tests and full scale experimental pressure vessel tests,

ii to assess the applicability of COD and J data at temperatures where LEFM methods are invalid.

iii to determine the effect on fatigue crack growth rate of mixed mode (I/II) loading.

iv to develop a laboratory test to examine factors influencing crack initiation (and arrest) under thermally induced shock loading.

3. Experimental Facilities

Equipment for failure tests on 1.5m dia x 36.5m long x 75mm thick pressure vessels. Laboratory test equipment of up to 1000 KN (static)/500 KN (dynamic) capacity.

4. Project Status

i tests on 1.5m dia pressure vessels containing partial thickness and full thickness defects have been completed. Fracture toughness data from small scale fracture mechanics test pieces are being compared with vessel failure data.

ii the influence of test piece geometry (bend vs tension) on COD and J has been assessed using HY130 steel. The programme is being repeated using A533B steel. Particular attention is focussed on measurement of initiation toughness and its relevance to plant operation in the upper shelf regime.

contd.......

1194

| Classification | |
|---|---|
| Title 1 | COUNTRY |
| | SPONSOR |
| | ORGANIZATION |
| Title 2 | Project Leader |
| Initiated / Completed : | Scientists: |
| Status : / Last updating | |

Description

iii tests are in progress using angle notched centre cracked plates and modified compact tension test pieces to establish the influence of shear loading on fatigue crack growth rates. Test specimens are 12.5 and 25mm thick low alloy pressure vessel steel.

iv the essential features of thermal shock induced crack initiation and arrest are being studied by applying a known thermal shock to statically loaded three point bend test pieces. Preliminary calculations have been made and a series of test pieces are being prepared.

Reference Documents

Internal documents.

-1195

Classification 11.2.1

| | |
|---|---|
| Title 1 Mechanics of Plane Strain Fracture | Country UK |
| Title 2 | Sponsor UKAEA<br>Organisation<br>Fuel Element Labs. Springfields |
| Initiated 1972 Completed<br>Status Continuing Last updating | Project leaders<br>Dr. B. Tomkins |

1 General Aim

To recognise and assess modes of failure of nuclear reactor pressure vessels.

2. Particular Objectives

To measure the deformation in a thick cracked plate up to the point of fracture.

3. Experimental Facilities and Programme

A rig has been constructed in which a high pressure (circa 3000 bar) may be applied within a crack in the plate.

The use of internal pressurisation means that fracture criteria in thick section material can be studied under well defined boundary conditions. Direct surface strain measurement around the crack tip will be done and fractography employed to assess fracture development prior to fast fracture propagation.

Reference Documents

Internal documents.

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.5. - 31.12.77 | 11.2.2 | RS 262 |

| Vorhaben/Project Title | |
|---|---|
| Untersuchung zur Zeitfestigkeit im zweiachsigen Spannungsfeld mit reinem Zug und Druck<br><br>Study on the finite fatigue strength in a biaxial stress system subjected to purely tensile and compressive loading | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Fried. Krupp GmbH Krupp Forschungsinstitut |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.5.77 | 30.4.79 | Dipl.-Ing. Spandick |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | Dec. 31, 1977 | 247.281,-- DM |

## . General

The codes of practice for the design of pressure vessels permit the yield point to be exceeded considerably in the case of secondary stresses. Pressure vessel components are therefore often to be designed for finite fatigue strength. In doing so, multi-axial cyclic thermal deformations, partly or even fully prevented, must be allowed for up to the plastic range. The design practice according to the ASME code is based on uni-axial, mechanical tests and uses high safety margins. Theoretical studies known from literature give rise to serious doubts as to whether it is admissible -

- to compare multi-axial loads in the higher temperature range with uni-axial loads via the reference stress, e.g. according to v. Mises, and
- to treat thermal cycles merely as mechanical cycles at an equivalent temperature.

For safety reasons it is therefore necessary to more closely investigate the loading limits for the materials in question.

## 2. Objectives

In order to obtain experimental data in support of the design of pressure vessel components subjected to thermal multi-axial cycles loading this research project provides for thermal biaxial-load-cycle-tests to be carried out on selected materials. In these tests the fatigue life as a function of the restrained deformations is to be determined. Furthermore, the stress-strain curve is to be plotted as a Bauschinger loop for each temperature cycle.

## 3. Test program

As set forth in the application the research project centers on the following main program points:

3.1. Evaluation of publications on recent work in the field of low-cycle fatigue throughout the term of the project.

3.2. Design and manufacture of suitable test facilities specially adapted to meet the project objectives.

3.3. Establishing the testing procedure

3.4. Testing

3.4.1. Tests on the material TTStE 29
Heating to 200 °C, 300 °C, 400 °C, 500 °C, holding time 0 min., cooling to approx. 50 °C.

3.4.2. Tests on the material No. 4541 (stainless steel) like 3.4.1.

3.4.3. Tests on the material No. 4948 (stainless steel) like 3.4.1., however, only for one temperature cycle

3.4.4. For comparison purposes specimens of the three above-mentioned materials will be subjected to mechanical uni-axial cycles at a maximum deformation of 5 °/oo, a constant testing temperature of 300 °C and a holding time of 0 min.

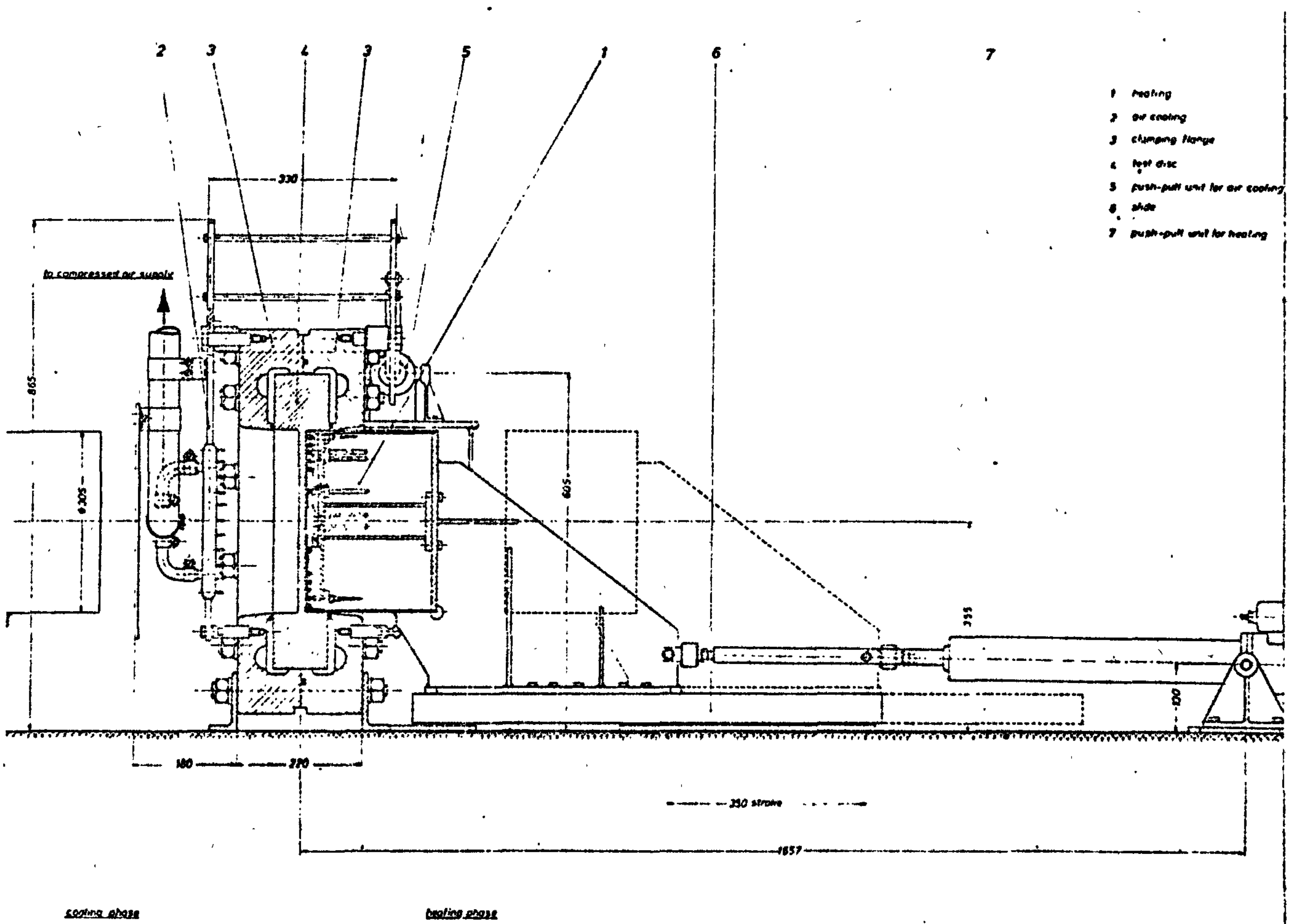
3.4.5. Tests as under 3.4.4. but with a holding time of 20 min.

3.5. Evaluation of the individual test results.

4. Test facilities, EDP programs

The following testing equipment was installed:

The test disc is constrained by a heavy water-cooled ring which largely prevents any thermal deformation. The disc is heated on both sides by electric radiating elements. The latter can be electromotively retracted for cooling of the disc. A cycle time of 5 - 10 min is reached. The heating coils of the radiating elements are suitably arranged to ensure an approximately constant temperature over the disc. The temperature distribution is measured by means of thermocouples. Temperature-compensated high-temperature strain gauges are used to measure the restrained deformations.




The drawing shows the principle of the test set-up

## 5. Work performed

ad 3.1.

The following publications touching on the project have been evaluated:

- Analysis of biaxial-fatigue damage at elevated temperatures, S.Y. Zamrik and O.G. Bilir (Pennsylvania State University)

- Fatigue strength under superposed biaxial static loading, E. El-Magd and S. Mielke (RWTH Aachen)

- Testing of welded joints of different steels for thermal fatigue, N.M. Korolev, Yu. A. Butov and A. K. Kovyazin (Moskow)

Because of the parameters used for these tests (either the yield point was intentionally not exceeded or the temperature was left out of consideration, etc.) the results cannot be readily compared with those that are to be obtained from the project under consideration.

At the VDI conference on "the Behaviour of Thermally Stressed Materials and Components" held in Munich on November 24 and 25, the objectives of this project RS 262 mentioned under items 1 and 2 were cited as being a very difficult, as well as necessary task. Then many machines and apparatus are likely to fail under the thermal loading conditions involved. In spite of the great number of excellent investigations already carried out successfully, we are still far from making reliable predictions of thermal fatigue life. The principal methods for computing the damage have been presented:

- the method of linear superposition of creep- and oscillate-damage according to Taira
- the extension of the Coffin-Manson rule to influences of frequency by Coffin
- the subdivided strain-area-method by Manson, Halford and Hirschberg.

Unfortunately, the formulations given for computation only contain characteristic values for the materials which can only be determined by suitable thermal tests. Low-cycle-fatigue experiments simulating actual service conditions are therefore necessary before the theoretical work can be continued.

ad 3.2.

The testing equipment described under 4 was designed and built. A test disc fitted with thermo-couples was used to adjust the equipment so as to ensure optimum heating and cooling of the specimens. The temperature obtained for each disc centre temperature are shown in item 6 of this report.

ad 3.3.

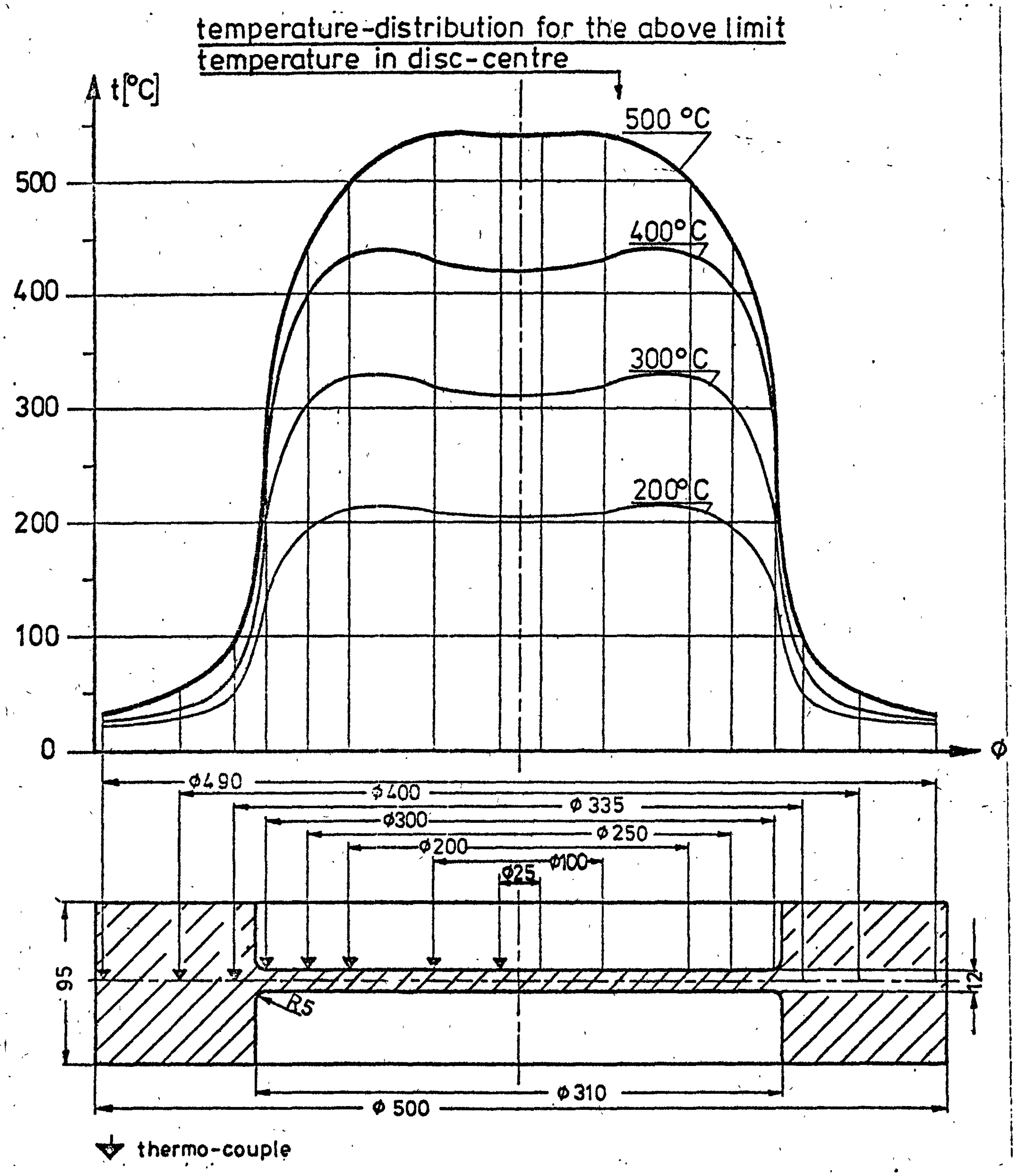
The test procedure was established and suitable forms prepared.

ad 3.4.1

The first test disc made of the material TTStE 29 was fitted with thermo-couples and strain gauges in accordance with the measuring point chart. Cyclic testing was started between the temperature limits of 50 °C and 300 °C.

6. Results

The sketch shows the test disc with its thermo-couples, which was used to adjust the equipment to optimum heating and cooling conditions. The temperature distribution over the disc is shown for each disc centre limit temperature.



## 7. Further work planned

ad 3.1.

Evaluation of any new publications concerning this project.

ad 3.4.

Continuing the cyclic loading between 50 °C and 300 °C. On completion of this cycle, carrying out the next test at the limit temperatures mentioned in the test program (see item 3).

ad 3.4.4. and 3.4.5.

Uni-axial tests intended for comparison purposes will be prepared and commenced concurrently with the disc tests.

## 8. Relations to other projects

The Interatom Company intends to carry out uni-axial mechanical cyclic loading tests at constant temperatures on the ferritic material 10 Cr MO 9 10 (Nb-stabilized) and the austenitic material X 6 Cr Ni 18 11 (No. 1.4948). Similar tests are to be carried out at the Jülich Nuclear Research Centre on materials retaining high strength at elevated temperatures. During the performance of the project a discussion of the results is provided.

## 9. Literature

-

## 10. Accessibility of Reports

-

1205

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 01.08.77 - 31.12.77 | 11.2.2 | RS 0275 |

| Vorhaben/Project Title | |
|---|---|
| Untersuchungen zur Übertragbarkeit von Kennwerten für den Bruch und Rißfortschritt von Proben auf Bauteile im Hinblick auf die Sicherheit von Kernkraftwerkskomponenten<br><br>Transposition of Fracture and Crack Propagation Characteristics from Specimens to Structures Considering the Safety of Components of Nuclear Power Stations | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Bundesanstalt für Materialprüfung (BAM)<br><br>Berlin<br>Abteilung 1 |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 01.08.1977 | 31.07.1980 | Dr. Aurich |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | December 1977 | 1.567.000,-- DM |

1. General Aim

Clarification of the influence of different states of nominal stresses, as usual in pressurized components of nuclear power stations and in conventional specimens, on the fracture behaviour of material with defects. In usual specimens, an uni-axial state of nominal stress is produced, but in pressurized components a multiaxial state of nominal stresses occurs. Such components are the cylindrical parts and the bottoms of pressure vessels, containments and pressurized pipes under tension and momentum load. The induced principal stresses in such structures should be simulated in test specimens, neglecting the influence of radial stresses. The effect of defects on the local stress distributions should be analyzed and their influence on the fracture behaviour in dependence of the nominal stress state should be evaluated.

2. Particular Objectives

2.1 Development of test specimens

in which states of nominal stresses can be simulated as they might occur in pressurized components of nuclear power stations by means of a FEM-stress analysis.

2.2 Analysis of local states of stresses

near defects, especially analysis of the K-factors and J-values in dependence of the nominal stress state by means of numerical and experimental procedures. Elastic and elastic-plastic material behaviour as far as possible should be taken into account.

2.3 Test performance
on specimens with defined defects in dependence of temperature and under a nominal stress state as it occurs in components. Comparison of these results with those on conventional uniaxial stressed specimens including fracture mechanics tests. Cleavage fracture, ductile fracture and fatigue crack propagation should be considered. The results should be represented in dependence of the nominal stress state.

2.4 Development of transposition criterias
for characteristics of fracture, crack propagation and critical temperatures from one stress state to another stress state.

## 3. Research Program

3.1 Test specimen development by means of stress analysis

3.1.1 Development of the specimens shape and the shape of the joints, taking into consideration the experiences of pretests which were performed on our own account.

3.1.2 Evaluation of $K_I$- and $J_I$-values, if possible.

3.2 Evaluation of $\sigma_{ij}(r, \varphi, z)$
for the state of small scale yielding and general yield as far as financial means are available.

3.3 Investigation of Fracture Behaviour

3.3.1 Specimen with cracks
should be investigated for clarification of the influence on the fracture behaviour under different states of nominal stresses.

3.3.2 Conventional specimens
should be investigated in addition to determinate the basic mechanical properties such as yield strength in dependence of temperature, the cleavage fracture strength and so on.

3.4 Transposition concept
should be tried on the basis of results under 3.2 in combination with the fracture hypotheses.

4. Experimental Facilities and Computer Codes

4.1 Specimen Development
The stress analysis should be performed by the FE-programs of the SAP-family, especially NONSAP and the advanced development ADINA. The MARC-CDC-program is not regarded for the time being owing to reasons of too high costs.

4.2 Evaluation of $\sigma_{ij}(r, \varphi, z)$
The elastic-plastic stress analysis should be performed by the NONSAP-FE-program. Additional possibilities should be considered.

4.3 Investigation of Fracture Behaviour
For cooling down the big specimens, a special cooling device is necessary. Experience in this field is available, however, the detailed development of the device cannot be started before a clarification of the specimen geometry is reached, even so the scientific instrumentation plan needs clarification of the specimen geometry.

4.4 Concept of Transposition

5. Progress to Date

5.1 Development of Test Specimens
Preliminary work for the use of the FE-programs of the SAP-family and the own development of additional programs were performed for the application on machines of types TR 440, CD 6500 and CYBER 175 which have be used by BAM. Elastic stress analysises were done for several specimen geometries of disks and plates. Cooperation with the section "Static of Building Structures" of the Technical University of Berlin has been started with the aim to perform stress analysises in the elastic-plastic range of materials behaviour by the NONSAP/

ADINA-FEM-program.

5.2 Evaluation of $\sigma_{ij}(r, \varphi, z)$

For the evaluation of stresses and deformations near the crack as well as for the evaluation of $K_{Ic}$ and $J_{Ic}$ values, a co-operation for exchange of experience in the field of important FE-programs, special crack elements and methods for the determination of fracture mechanics values have been started by contacting other research institutes in Western Germany.

5.3 Investigation of Fracture Behaviour

Steel 20 MnMoNi 5 5 has been ordered. The material is already casted and partly forged. It should be delivered as soon as the specimen geometry is finally fixed.

Tests were carried out on specimens of the pretests (see fig. 1 and 2) prepared with double-edge notches of 5 mm radius in the center line of the specimen. The strains in the notch tips were measured during loading at room temperature. Plastic zone sizes should be determined by micro hardness tests and metallographical methods.

5.4 Concept of Transposition

6. <u>Results</u>

6.1 Specimen Development

The results of FEM-calculations for specimens without crack and for elastic material behaviour demonstrate that plate-shaped specimens have advantages in comparison with disk-shaped specimens, especially if the loading capacity of the testing device to be used is considered.

6.2 Evaluation of $\sigma_{ij}$ (r, φ, z)

Review of literature and experience exchange with other research institutes demonstrate the possibility to use isoparametric finite elements (2D, 3D) for simulating the influence of cracks on the stress-deformation distribution in specimens or in structures. This can be done in principle by the NONSAP/ADINA-FE-program for elastic material behaviour as well as for elastic-plastic material behaviour.

6.3 Investigation of Fracture Behaviour

Tests on specimens of the pretests demonstrate that even at room temperature higher stresses $\sigma_{zz}$ parallel to the notch tip are induced in the type 2 specimen (see fig. 2) than in the type 1 specimen (see fig. 1) under the same net stress. In the type 2 specimen tensile deformations ($d\varepsilon_{zz} > 0$) are observed in the notch tip up to high nominal stresses, but in the type 1 specimen under the same test conditions compression strains ($d\varepsilon_{zz} < 0$) in the notch tip already occur at the beginning of loading. So it follows: Full constraint cannot operate near the notch tip in a notch of 5 mm radius and therefore plane strain conditions could not be reached in the type 1 specimen ($d\varepsilon_{zz}<0$). A nominal stress parallel to the notch tip as induced in the type 2 specimen however influences the deformation behaviour in the notch tip so as $d\varepsilon_{zz}>0$ can be reached even under high net stresses (type 2 specimen).

6.4 Concept of Transposition

7. Next Steps

7.1 Specimen Development

Further FEM-calculations for determination of the optimal specimen geometry by comparison with the results for plates and disks and taking additionally into account the loss of rigidity by the crack and the possible inaccuracy owing to the force initiation.

7.2 Evaluation of $\sigma_{ij}(r, \varphi, z)$

Preparation and testing of the NONSAP/ADINA-program with focused isoparametric elements for elastic material behaviour.

7.3 Investigation of the Fracture Behaviour

Ordering of offers for machining of specimens. Tests on models of the final specimen shape for comparison with results of nominal stresses, evaluated and determined by tests. Checking of the influence of inaccuracies produced by the force initiation.

7.4 Concept of Transposition

8. Relation with other Projects

—

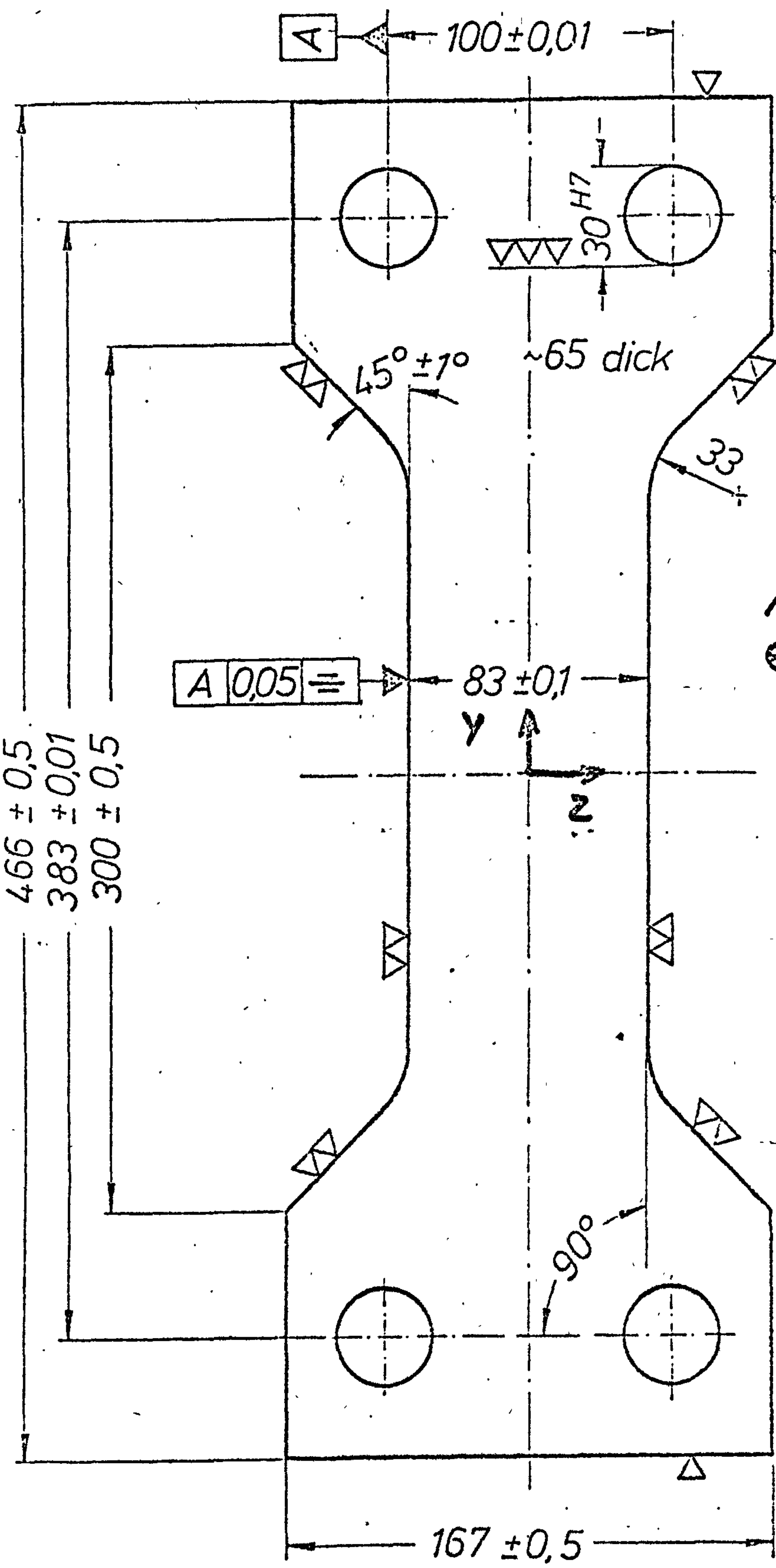
9. Literature

—

10. Degree of Availability of the Reports

—

Figure 1

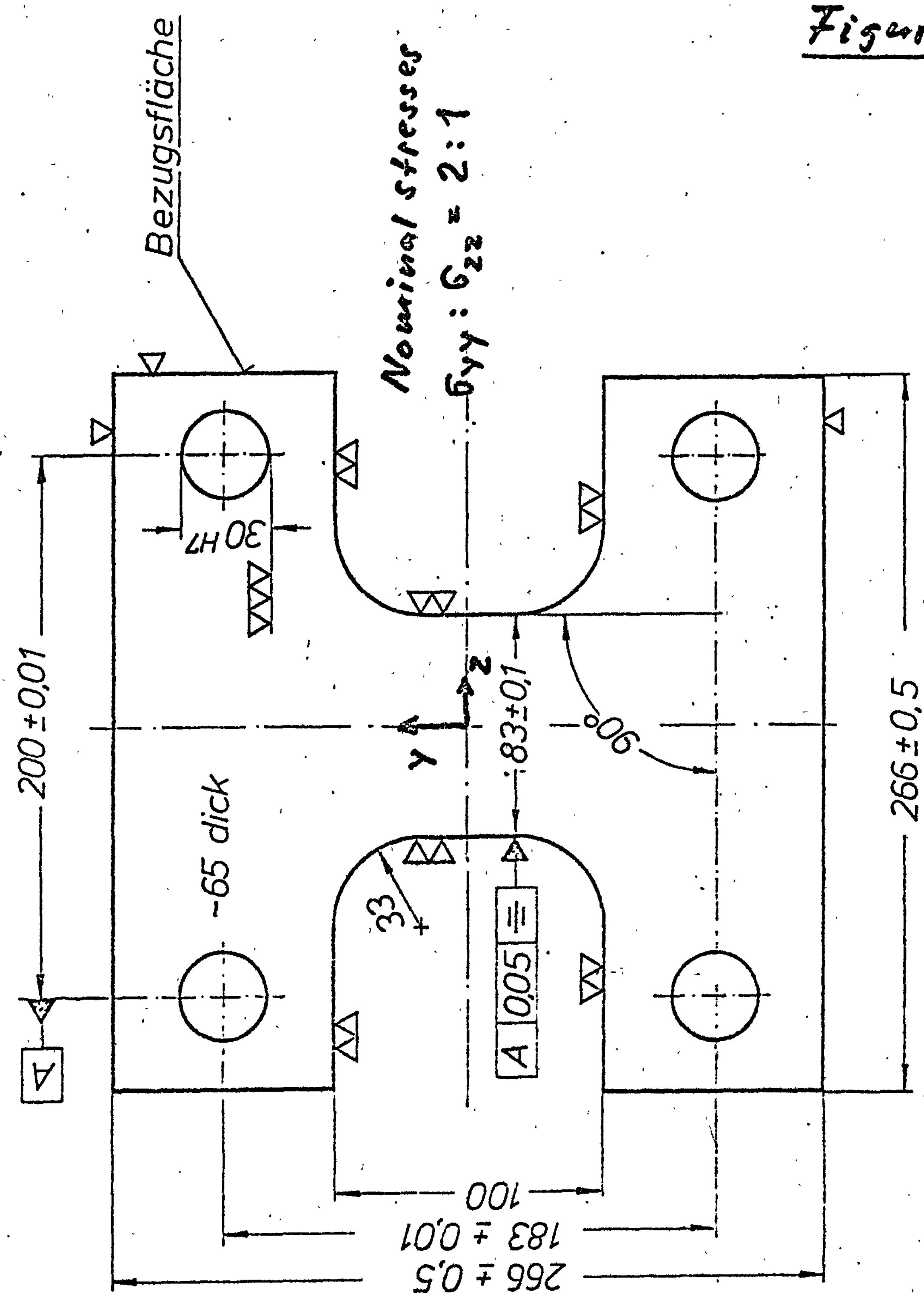



Nominal stresses
$\sigma_{yy} : \sigma_{zz} = 1:0$

specimen
type 1

M.12

Figure 2




specimen type 2

1213

Classification
11.2.2/11.2.3

| Title 1<br><br>Spændingsanalyse af primære trykbærende stålkomponenter: Sammenligning mellem beregnede og målte tøjninger og spændinger i en BWR pumpestuts. | COUNTRY Denmark<br><br>SPONSOR Risø National Lab.<br><br>ORGANIZATION Risø National Lab. |
|---|---|
| Title 2<br>Stress analysis of primary steelcomponents:<br>A comparison between calculated and measured stress and strain in a BWR main circulation pump nozzle. | Project leader<br>S.I. Andersen<br><br>Scientists: |
| Initiated (date) 74.04.01     Completed: 1977 | S.I. Andersen |
| Status: concluded     Last updating(date) April 1977 | |

1. General aim. The purpose of the project is to establish the accuracy, which can be obtained by stress analysis of a complicated pressure vessel component — and to determine the degree of sophistication, required in such calculations.

2. Particular objectives. The pump nozzle in a BWR-steel pressure vessel has been chosen as object for this investigation. The nozzle is located in the transition zone between the spherical bottom head and the cylindrical vessel part. The nozzle axis is parallel to the centerline of the vessel.

3. Experimental facilities and programme. During the manufacturers hydrotest of the vessel, strain measurements has been performed on the pump nozzle.

4. Project status

4.1. Progress to date A 3-dimensional finite element model of the nozzle has been generated, and 3 load cases run: hydrotest, stresses due to stationary temperatures, and stresses during normal operation conditions. Besides, the experimental program, i.e. strain measurements, has been performed.

4.2. Essential results The stresses and strains due to the internal pressure has been obtained, and both calculated and measured values are shown to be in good accordance with each other. Besides, the results from simplified calculations (2 D) has been compared to the 3D- and experimental results.

5. Next steps

None

6. Relations with other projects The work is related to

1) Risø investigations of the validation of structural computer codes

2) Risø work on the safety of primary pressure system.

7. Reference documents


[1] S.I. Andersen, J. Reynen, P. Engbæk:
"Stress Analysis of a Main circulation Pump Nozzle".
Risø-TPM-76/1, Jan. 1976.

[2] S.I. Andersen, T. Henriksson, J. Reynen:
"Stress Analysis of a MCP Pressure Vessel Nozzle".
Paper G 8/4 at the 4th Int. Conf. on
Structural Mech. in Reactor Techn., San Francisco 1977.


8. Degree of availability A limited amount of the results may be freely available.

| | | |
|---|---|---|
| 141-1 -05/4111-617 | | 11.2.2 |
| Titre<br>Mesure d'une caractéristique d'arrêt de fissure dans les aciers pour cuves de réacteurs. | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br>Crack arrest methodology for nuclear pressure vessel steels | | Organisme exécuteur<br>CEA/DTECH<br><br>Responsable<br>DSN-SETSSR-Fontenay |
| Date de démarrage 01/01/78 | Etat actuel<br>démarrage | Scientifiques<br>(DTECH-SRMA) |
| Date prévue d'achèvement 01/01/81 | Dernière mise à jour 01/01/78 | |

1 - Objectif général :

Le programme proposé a pour but d'étudier les conditions mécaniques et métallurgiques qui conduisent à l'arrêt d'une fissure en cours de propagation.

- Objectifs particuliers :

Le programme concerne essentiellement les aciers pour cuves de réacteurs PWR. La première phase de l'étude consiste en une revue bibliographique et la mise au point de méthodes d'essais.

4 - Principales étapes : 1978 Etude bibliographique
Définition de l'appareillage
1979 Mise au point des essais
1979-80 essais

| TITLE 1 (original language)<br>Vibrazioni di sistemi di tubazioni | Classification<br>11.2.2 |
|---|---|
| TITLE 2 (english)<br>Structural and acoustic vibrations of piping systems | Country: ITALY<br>Sponsor: CNEN-CNR-Italimpianti<br>Organisation: University of Pisa |
| Date initiated 1973<br>Date completed 1976<br>Last updating June 1976 | Project Leader<br>A. De Paulis |

Description :

Research program:

A multi-purpose computer program has been developed with the aim to study structural and acoustic vibrations of piping systems.

This program will be useful to minimize objectionable piping vibrations at resonance and to control and/or eliminate excessive vibrations developed in service.

Facilities:

IBM 370/168 Computer belonging to CNUCE of Pisa.

Reference documents:

1. C. CARMIGNANI, A. CELLA, A. DE PAULIS
Structural Dynamics by Finite Elements: Modal and Fourier Analysis
Proc. 2nd Int. Conf. S.M.i.R.T. - Berlin 1973.

30 juin 1978

| | |
|---|---|
| | Water reactor 11.2.3. |
| Emission Acoustique | Belgium<br>IRSIA |
| Acoustic Emission | Association Vinçotte<br>P. CAUSSIN |
| December 75 \| In progress | L. JACQUES-HOUSSA<br>W. SYS |

## 1. General aim

Analysis of the diagnosis capabilities of multi-channels acoustic emission (AE) systems used for monitoring of proof tests of pressure vessels.

## 2. Particular objectives

Study of the correlation between the AE signal characteristics and the initiation and the development of cracks in welded products.

## 3. Experimental facilities and program

AE monitoring of
- tensile tests of 5 1000 x 600 x 20 mm steel specimens at room and - 20°C temperature
- tensile tests of 20 1000 x 800 x 50 mm steel specimens at room and - 20°C temperature
  (the specimens contain different kind of stress raisers including weld defects)
- 2 large pressure vessels.

The facilities include :

- a 8 M Newton tensile machine
- a 60 M Newton tensile machine
- a computarized 24 channels AE system fitted with 6 transient recorders for waveform analysis (FFT, ...)
- Equipments for C.O.D. measurements, Moiré photographs etc...

## 4. Project status

Specification and acquisition of the AE system.
Check of the tensile machines for noise pollution in the 0.1 to 1 MHz frequency range.

## 5. References

Diagnosis abilities of acoustic emission multichannel systems : experimental set-up.

P. CAUSSIN, L. JACQUES-HOUSSA, W.SYS.

Proceedings of the Institute of Acoustic, Conference on Fundamental Aspects and applications of A.E., London, December 20-21, 1976.

## 6. Co-operation

This program is conducted in co-operation with the Laboratory for the Strength of Materials, Ghent University.

## 7. Availability

Details available at

ASSOCIATION VINCOTTE
Département Etudes
B. 1640 RHODE-SAINT-GENESE

30 juin 1978

| | | |
|---|---|---|
| | | Water reactors 11.2.3 |
| Contrôle ultrasonore des aciers austénitiques | | Belgium |
| Ultrasonic testing of austenitic steels | | Association Vinçotte |
| | | P. CAUSSIN |
| 1973 | | J. CERMAK |
| In progress | | |

## 1. General aim

Improvement of the capability of ultrasonic testing austenitic stainless steel structures, including welds and castings.

## 2. Particular objectives

Development of a technique complying with the ASME code requirements for the inspection of pressure vessels and piping systems of nuclear installations.

## 3. Experimental facilities and programme

Development of focused, longitudinal wave angle probes and of procedures for applying them.
Facilities are available for manufacturing anc characterizing the probes.
The performances are tested on different kinds of welded blocks and castings.

## 4. Project status

14 probes with refracted angles between 45 and 70 degrees have been developed and are used for inspecting in the field specimens from 20 up to 100 mm thick. The structures inspected include pipe welds, safe-end welds, castings, etc. In that thickness range forgings, plates, statically and centrifugally cast pieces, etc. are inspected in compliance with the ASME code.
Procedures have been developed for the inspection of those items and also for testing thin (5 to 20 mm thick) specimens using standard probes.

## 5. Next steps

Developments are still necessary to improve the inspectability of heavy castings up to 200 mm in thickness.

## 6. References

Ultrasonic testing of austenitic steel castings and welds

J.P. PELSENEER, G. LOUIS

Br. J. of NDT, July 1974, pp.107-113.

Ultrasonic testing of austenitic stainless steel structures.

P. CAUSSIN

Association Vinçotte report to the OECD-CSNI Working Group on Mechanical and Material Problems relating to Safety Aspects of Steel Components in Nuclear Plants, revision 1, October 18, 1976, 15 p.

## 7. Availability

Details available at

ASSOCIATION VINCOTTE
Département Etudes
B 1640 RHODE-SAINT-GENESE (BELGIUM)

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 11.2.3 | RS 102-20/2 |

| Vorhaben/Project Title | Land/Country |
|---|---|
| Akustische Holographie - Labor- und anwendungstechnische Versuche mit dem Holscan 200<br><br>Acoustical holography - laboratory and application tests with the Holscan 200 | FRG<br><br>**Fördernde Institution/Sponsor**<br>BMFT<br><br>**Auftragnehmer/Contractor**<br>Fraunhofer-Ges., München<br><br>IzfP, Saarbrücken |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.8.1975 | | Dr. V. Schmitz |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Completed | Dec. 31, 1977 | DM 626.100,-- |

## 1. General Aim

In non-destructive testing of nuclear power plants one of the main aims is to get an identical three-dimensional picture of flaws, respectively of critical fault areas. An exact knowledge of these flaws is necessary to determine the influence on the stability or safety of these components.

## 2. Particular Objectives

The acoustical holography links the conventional method of pulse-echo through-transmission or tandem method with the imaging capability of optical holography. In determining amplitudes and phases of the flaw echo it is possible to image inhomogeneities like flaws, cracks, segregations or shrink holes. We are engaged in determining artificial or natural flaws in thick-walled ferritic or austenitic materials or interfaces between them like claddings, both in laboratory and in the field.

## 3. Research Program

3.1. Investigation of the mathematical methods in creating a hologram and in reproducing the picture of the flaws.

3.2. Experiments on specimens with artificial flaws, varying frequency, focal length of transducers, inclination and depth of flaws.

3.3. Experiments on specimens with natural flaws.

## 4. Experimental Facilities, Computer Codes

From the methodological point of view ultrasonic holography is better suited to the examination of thick-walled materials as it works in the far-field length of a focal probe. It is possible to use piezoelectric transducers with water-coupling or in contact technique and electromagnetoacoustic transducers. After emitting a spherical wave the reflected ultrasound is detected. To conservate the information of the phase, too, a coherent phase-shifted reference signal is superimposed. This is realized in the holographic equipment Holscan 200. By scanning rectilinearly the surface of the inspected components the holographic signal is written on a storage oscilloscope in a manner that follows the transducer motion. The hologram is then placed in the optical processor to reconstruct the image.

## 5. Progress to Date

The quality of the image depends on the axial and lateral resolution. The lateral resolution of holography is determined by the resolution of the focal probe. To get a good resolution the focal length should be short, the diameter big and the testing frequency high. The practically available resolution corresponds to one wavelength. This means flaws separated by at least one wavelength can be resolved in the image independent of the flaw depth. The axial resolution is not good. The determination of the flaw depth should be measured by determining the time of flight between the surface pulse and the flaw echo. This can be done with extremely high accuracy.

The size of the image depends upon the wavelength and the depth. The deeper the flaw the smaller the image.

It is possible to investigate the whole depth of the material in only one hologram. Flaws in different depth can be imaged on a screen one after the other. Concerning the signal-to-noise ratio it is sometimes better to gate the signal and to investigate smaller zones of the specimen. Naturally the inspection time increases. For the investigation of very big plates or rods, for instance 500 mm, it is possible to use probes with focal length of about 600 mm and to lay the focus into the interior of the material. The real distance of the flaw is now the distance to the focal plane and not to the surface of the specimen.

The application of the holographic method until now is limited to the inspection of pieces with plane or nearly plane surfaces.

Very important is that the opening angle of the probes should be large vice versa the flaws and that the flaw should be large vice versa the wavelength. If this is sure you get an excellent picture of the flaw in lateral direction. To determine the extension of the flaw in axial direction you have to produce more than one hologram out of different directions.

The time needed for a hologram is determined by the scan-velocity of the probe. A mean value is about 15 minutes for an aperture of 150 mm x 150 mm. This time can be drastically diminuished by using arrays with electronical switching the elements, one of our future efforts.

The quality of the reconstruction depends upon the quality of the surface roughness. A roughness of claddings of about 0,1 mm does not allow good images. By diminuishing the surface roughness to 0.005 mm the influence of the remaining cladding and the interface can be neglected and is comparable to experiments without claddings.

## 6. Results

All experiments have shown that the equipment has a high technical standard. The ultrasonic holography has shown that it is the most promising method of testing thick walls. Problems have arisen in the vertical mounting and couplant system. The technology of the fluid filled coupler should be improved or replaced by probes in contact technique. The scanner has shown to be very reliable, but there is a need for a smaller one for analysing cracks or flaws in welded regions which are situated in complicated or narrow places.

## 7. Next Steps

The potential development lies in the replacement of the optical processing by a numerical processing scheme. Then the information of the amplitudes and phases must not be combined in intensities but can be independently fed into a computer. The reconstruction is done numerically and can be presented on monitor or plotter. The mathematical treatment allows the expansion of application to curved surfaces, the improvement of the quality of the pictures and an easier handling of the measurement of shape, geometry and flaw location.
In 1978 our main effort lies on the systematic investigation of the two-dimensional numerical reconstruction, both in laboratory and in practical field application.

## 8. Relations with other projects: none

## 9. References

A survey of about 350 references is given in a technical report (IzfP No. 770418). Most of the experiments cited above are discussed in the technical report IzfP No. 770533.

## 10. Degree of Availability of the Reports

by GRS, Glockengasse 2, 5000 Köln 1.

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 01.01. - 31.12.1977 | 11.2.3 | RS 132 |

| Vorhaben/Project Title | |
|---|---|
| Durchführung von Untersuchungen zur Rißerkennung an druckführenden Reaktorbauteilen mit Hilfe der optischen Holographie<br><br>Crack detection in pressurized vessels and reactor components by optical holography | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Institut für Verfahrenstechnik der T.U. Hannover<br>Callinstr. 36<br>3000 Hannover 1 |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| July 1974 | June 1978 | Prof.Dr.-Ing. F. Mayinger |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | December 1977 | 627.600,00 DM |

## 1.. General aim

The aim of the activities is to develop a quick and reliable optical method in order to detect cracks in reactor components during fabrication and later in repeating tests.

## 2. Particular objectives

In the last few years a new nondestructive technique - holographic interferometry - was devevoped by which material defects are detectable. Up to now it was, however, only used for nonmetallic components.

Our aim is to apply this technique also for the testing of steel. The special objective of this investigation is to detect cracks in or near the surface of pressure vessels using surface waves which are recorded double-pulse holography.

## 3. Research program

At first, basic experiments with test pieces of simple geometric shape were made. The aim of these preliminary experiments was to find out the smallest crack that can be measured with holographic interferometry in connexion with the impact excitation of the test pieces. The indication of cracks is influenced by various parameters:

1. crack-parameters (shape, size, orientation and position of the crack)
2. test piece-parameters (geometric shape, thickness and material)
3. test-method-parameters (impact-parameters sensitivity of the method)
4. influence of a small liquid layer on the surface of the test piece.

Further experiments shall be carried out with test pieces of any geometric shape. After these preliminary investigations experiments will be made with real reactor components like pipelines, elbows, T-pieces, etc.

to proof the reliability of this new nondestructive test-method. For these measurements in a nuclear power plant during a repeating test the holographic set-up must be reconstructed to a compact and versatile device.

## 4. Experimental facilities

The principle of this holographic technique is to visualize the deformation of surface and bending waves due to small irregularities in the material. These waves cause surface deformations which can be measured with holographic interferometry.

To produce the waves it is usually necessary to strike the test section in a suitable manner. This impact excitation may be generated e.g. by a free falling steel ball, by the bullet of an air-gun, an ultrasonic transducer, or any other stress wave generator. The local impact is only a short time excitation after which the waves will travel in all directions. In order to record this very fast event holographically a Q-switched ruby laser is used which produces two short light pulses. The first giant pulse illuminates the test piece shortly before the impact and thus the test piece in its unstrained condition is recorded. The second laser pulse appears at a certain time after the impact to make visible the propagation of the wave. The whole experimental set-up was already discussed in the annual reports A 74, A 75.

## 5. Progress to date

The experiments with test pieces of simple geometric shape like plates in connexion with the described holographic technique are now finished. From this experience it is possible to describe the detectable cracks as a function of different parameters.
Furthermore investigations were made to use different impact generators to produce waves of higher frequencies than agitated by the falling steel ball. Experiments were carried out with an airgun bullet and an ultrasonic wave generator (1 MHz). Fig. 1 shows the experimental arrangement which was used to produce and to store the surface motion excited by an ultrasonic longtudinal wave generator.
To prove the efficiency of the holographic nondestructive test method, experiments were carried out with an structural part of a nuclear power plant. This part is a section of a feedwater boiler where a crack occured

in the vicinity of a welding seam after a period of operation.

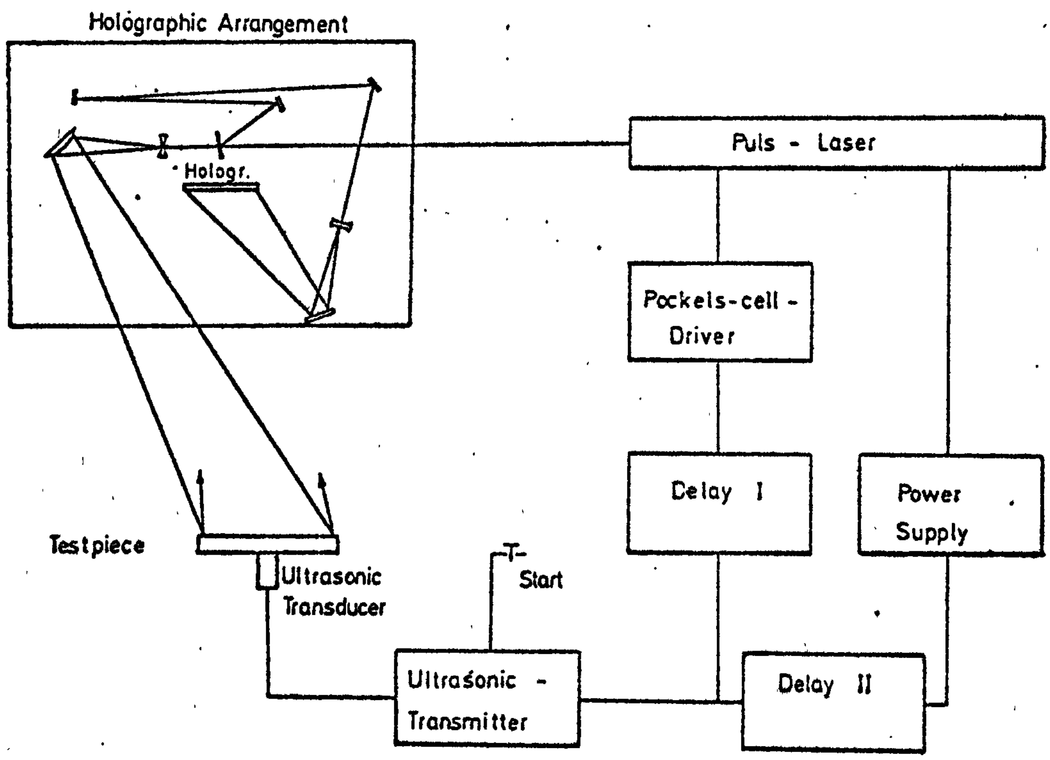



Fig. 1: Experimental arrangement for the detection of ultrasonic surface wave motion

## 6. Results

Theoretical and experimental investigations have shown that the method has a very high sensitivity as long as the cracks are near the surface. The described holographic technique certainly has limitations, too, due to the fact that the wave disturbance is extremly small for crack geometries of interest. Fig. 2 shows the results of several measurements which were made with and without an adequate magnification of the surface motion by the aid of an liquid layer of water.

As can be seen in fig. 2 it is possible to detect cracks of very small geometric size, when these cracks are located in the surface of the test piece. In addition it is noticeable that the thickness of the material is not limited if a liquid layer is used. This layer served in amplifying

the extremly small surface wave amplitudes and their distortion due to such small cracks.
The influence of two different impact generators are shown in fig. 3. and fig. 4.

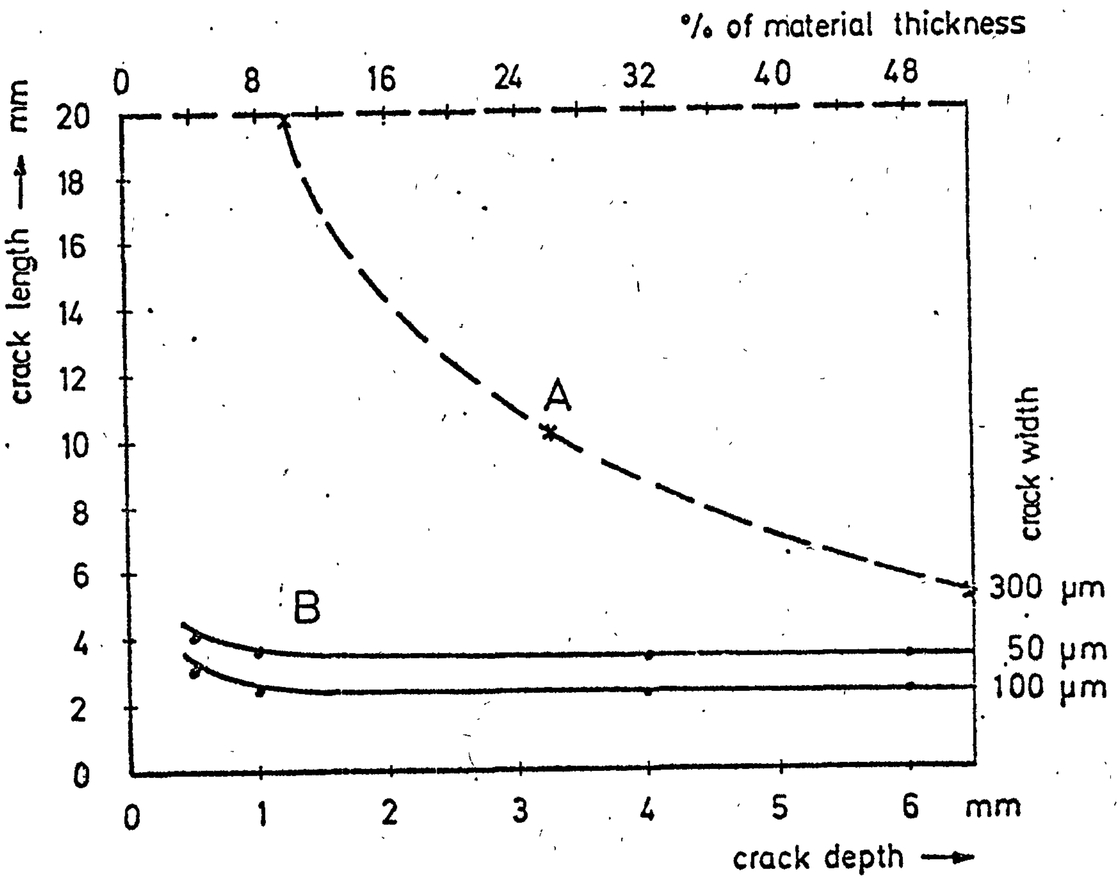



Fig. 2: Holographically detectable surface cracks
A) without the aid of a liquid layer
material thickness smaller than 40 mm
B) with the aid of a liquid layer
material thickness unlimited

Both results were obtained with the same test piece with a surface crack of 0,5 mm depth, 100 µm width and a length of 10 mm. Fig. 3 shows the result , which was obtained with the aid of a falling steel ball (diameter: 12 mm) and fig. 4 was obtained with the aid of a small airgun bullet (diameter: 4,5 mm). In both figures the distortion of the waves due to the crack is recognizable, but in fig. 4 the crack is surrounded by a closed interference fringe and therefore even better visible than in fig. 3.
After these results, experiments werde made with even smaller cracks, and as fig. 5 shows, surface cracks of a length of 3 mm, a depth of 0,5 mm

and a width of only 50 µm are clearly recognizable in the interference pattern.

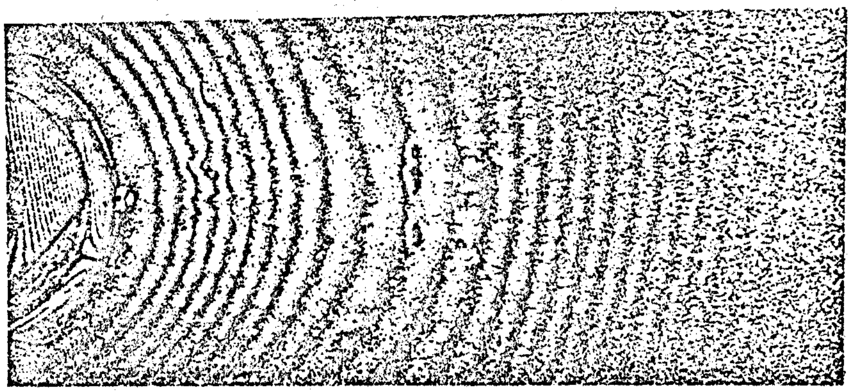


Fig. 3: Impact generator: falling steel ball 12 mm diameter

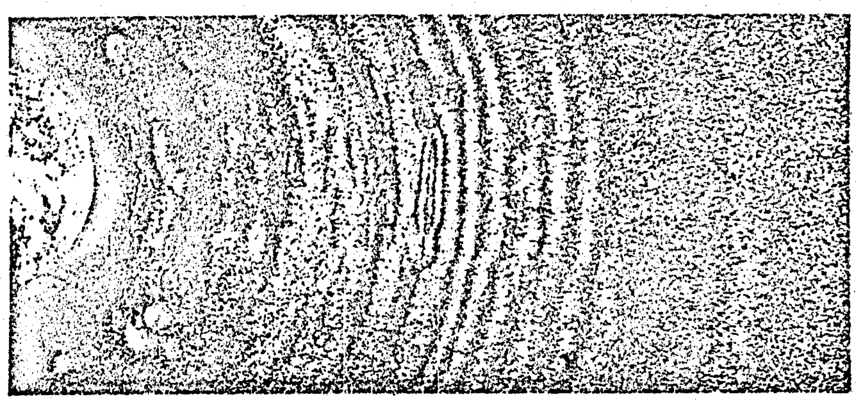


Fig. 4: Impact generator: air-gun bullet 4,5 mm diameter
Distortion of the interference pattern due to a crack of 10 mm length, 0,5 mm depth and 100 µm width

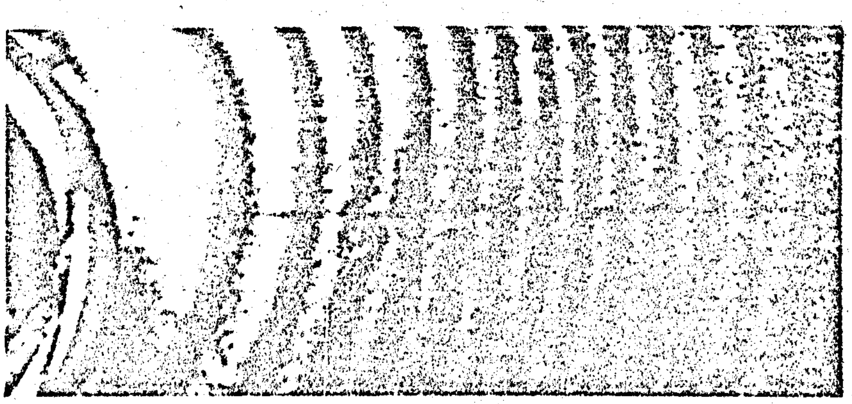


Fig. 5: Holographic detection of a surface crack
(length 3 mm, depth 0,5 mm, width 50 µm)
Impact generator: air-gun bullet 4,5 mm diameter

Further experiments with a test piece of a complicated geometric shape werde made (photo - fig. 6). This test piece is a section of a feed water boiler of a nuclear power plant, where a crack had occured in the vicinity of a welding seam. Since this crack is located in a geometrically complicated area the problem arose how to detect this material fault with the conventional nondestructive test methods. Therefore it was of interest if this crack can be detected by the aid of holographic interferometry. Two kinds of lads of the test piece werde chosen by means of which an appropriate holographically measurable surface deformation could be obtained, namely the short impact load by a free falling steel ball and a load occuring during the operation of the feedwater boiler. In such vessels positive pressure as well as vacuum can occur. The abnormal for the holographic detection of the crack necessary surface deformation in the vicinity of the crack is the highest, when the crack is expanded by the applied load.

Fig. 7 shows the result when the load exists of the small impact due to the falling steel ball. The crack in the melding seam is clearly visible by the break of some interference fringes. The result of the other load is shown in fig. 8, where a deformation of the feedwater boiler was applied similar to that under working condition. In the interferogram (fig.8) the crack is clearly recognizable also by the break of the interference fringes.

Further investigations were made with an ultrasonic transducer as an impact generator. This transducer consists of a longitudinal wave generator which is driven by a impulsively loaded transmitter. The amplitude of the generated wave is so small that it is not possible to detect the surface motion interferometrically without the aid of a thin liquid layer. Fig. 9 shows the arrival of the longitudinal wave which was generated on the rear side of a 85 mm thick steel plate. The area of the surface motion is surrounded by a sharp circular interference fringe. This interferogram was made 15 sec. after generating the wave - this time is exactly the time which the wave needs to travel through the steel plate.

Later on an ultrasonic standing wave occures , due to the reflexion on the surface and the edges of the test piece (fig. 10). The amplitude of the surface motion is even smaller than 50 n m and only by the aid of the liquid layer it is possible to measure such small vibrations.

1233

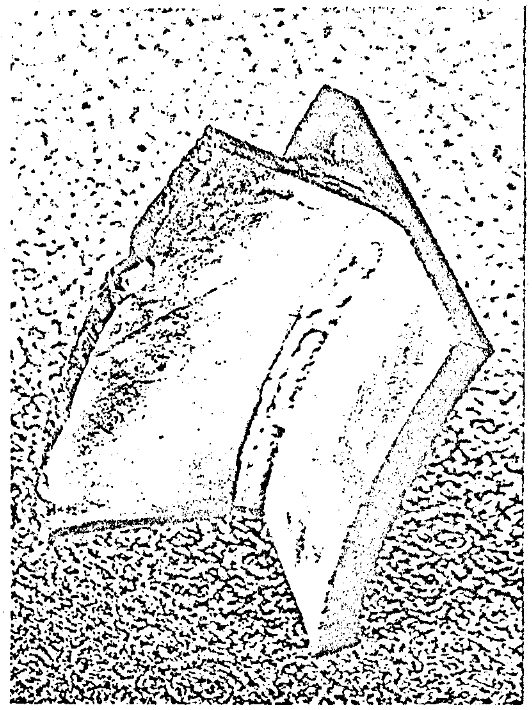


Fig. 6

Photo of a segment of a feedwater boiler of a nuclear power plant

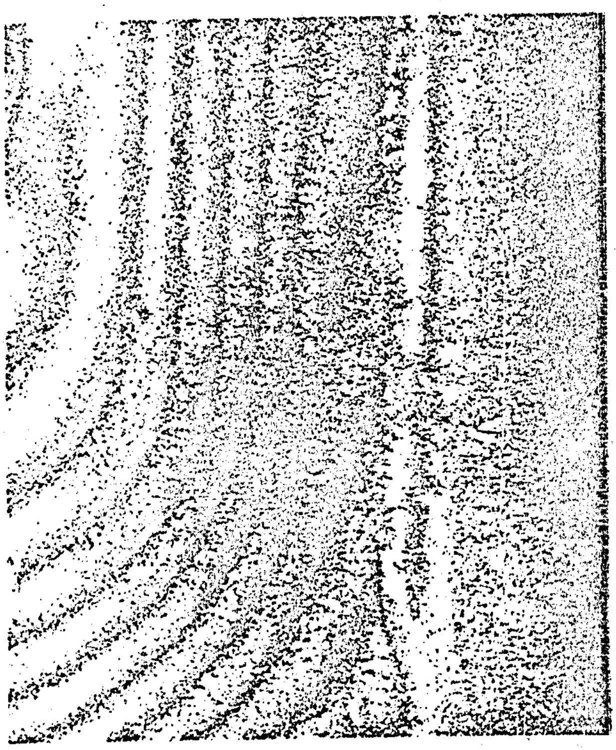


Fig. 7: Detection of the crack in the welding seam with the aid of a impulse load of the test piece.

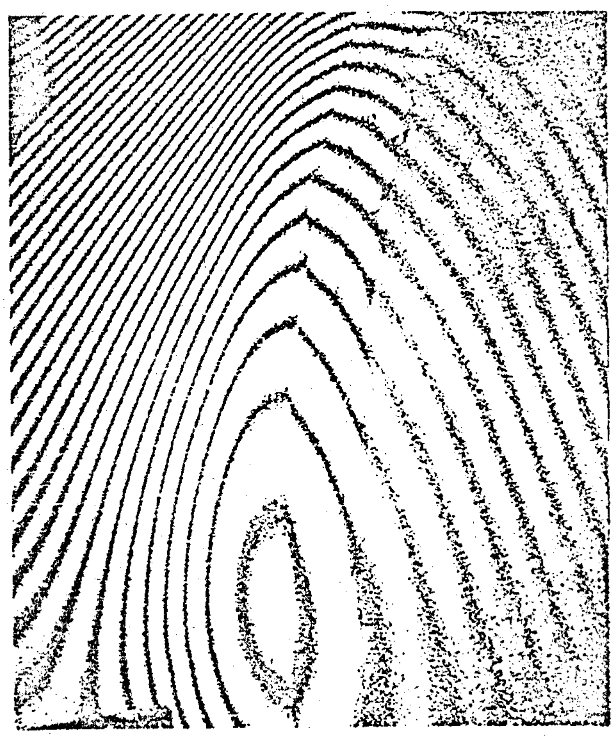


Fig. 8: Detection of the crack by a static bending load due to a contraction of the inner wall of the boiler.

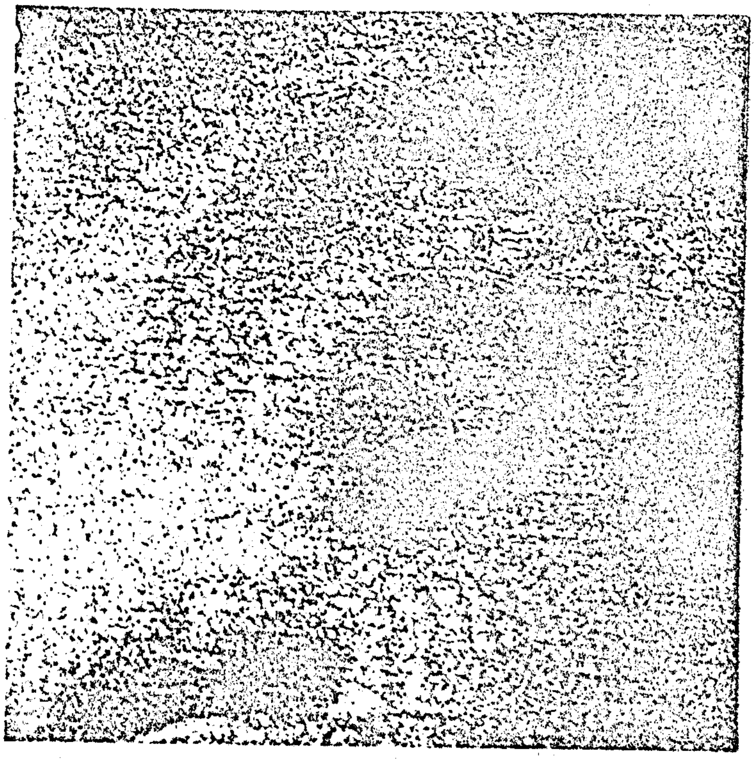


Fig. 9: Arrival of a ultrasonic longitudinal wave at the surface of a 85 mm thick steel plate.

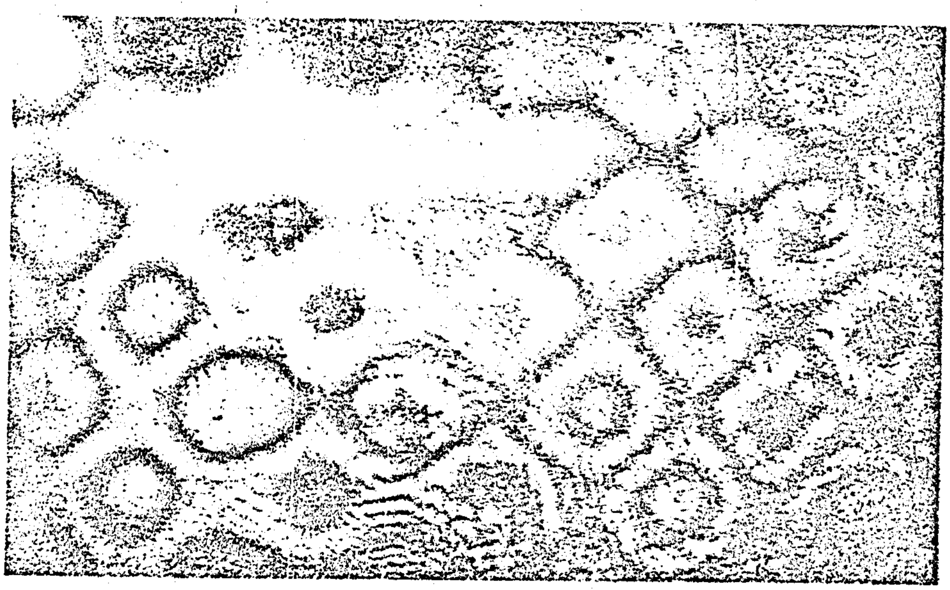


Fig. 10: Ultrasonic standing wave pattern

## 7. Next steps

It was shown in this report that it is possible to detect the ultrasonic motion on the surface of a test piece. More elaborated measurements with high power ultrasonic transducers (longitudinal-, transverse- and surface-wave transducers) and test pieces with material faults will be made to

improve this new test method.
Furthermore experiments will be made with some more pieces of the described feedwater boiler to show the reliability of the holographic nondestructive test method.

8. Relations with other projects

See annual reports A 74, A 75

9. Reference documents

Annual report A 74, A 75, A 76

Quarterly reports in the series GRS-Forschungsberichte

| | | | | | | |
|---|---|---|---|---|---|---|
| Report-period | Jan. | 1977 | - | March | 1977 | IRS-F. |
| " | April | 1977 | - | June | 1977 | IRS-F. |
| " | July | 1977 | - | Sept. | 1977 | IRS-F. |
| " | Oct. | 1977 | - | Dec. | 1977 | IRS-F. |

10. Degree of availability

The quarterly reports are available by Gesellschaft für Reaktorsicherheit (GRS).

1237

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. 77 - 31. 12. 77 | 11,2,3 | RS 298 |

| Vorhaben/Project Title | |
|---|---|
| Universalsteuerpult<br><br>Universal Control Panel | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 512, Erlangen |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1. 10. 77 | 31. 12. 78 | K. Ruthrof |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | 31. 12. 77 | 907.490,-- DM |

1. General Aim

Under the scope of developing devices for automated testing of reactor presure vessels, a universally applicable prototype control panel is to be developed and constructed which would perform by computer control US-testing for complicated geometries, by simultaneous multidirectional motion.

2. Particular Objectives

A panel for manipulator control is to be developed which a) will be relatively independent of both the manipulator drive method and power, and b) allows for complicated geometries to be followed reproducibly with a high degree of accuracy. Furthermore control of up to six different, independent and - in some applications - dependent motions of the manipulator will be required. Various modes of control operation including manual and automated operation have to be provided to ensure the highest possible serviceability.

The control panel is presently being developed as a prototype panel. If during the testing period the experiences with various manipulators are found to be favourable, follow-on models of lower cost will be constructed.

3. Research Program

Based on the experiences to date, various independent modes of operation will be used as a base for the panel concept:

3.1 Possible operating modes of the control panel
- operating modes without process computer
- operating modes with process computer

3.2 Hardware
Control panel
Computer portion

3.3 Software package
Support programs
Operation programs

4. Test Facilities
No special equipment and computer programs are existing fot this program.

5. Progress to Date
Design work for construction of prototype control panel for manipulator control was initiated.

6. Results
No special results exist up to now.

7. Next Steps
Preparation of a specification framework.

8. Relation with Other Projects
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9. References
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10. Degree of Availability
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123.9

| Berichtszeitraum/Period 9/1/77 - 12/31/77 | Klassifikation/Classification 11.2.3 | Kennzeichen/Project Number RS 267 |
|---|---|---|
| Vorhaben/Project Title | | Land/Country FRG |
| Ermittlung der optimalen Ankoppelspaltdicke für die mechanisierte Kontakttechnik-Ultraschallprüfung von Reaktordruckbehältern<br>Investigating the optimal thickness of the coupling layer for mechanized ultrasonic inspections of reactor pressure vessels using the contact technique | | Fördernde Institution/Sponsor BMFT<br>Auftragnehmer/Contractor Krautkrämer GmbH, Köln<br>KWU AG, Erlangen |
| Arbeitsbeginn/Initiated 9/1/77 | Arbeitsende/Completed 8/31/78 | Leiter des Vorhabens/Project Leader Dr. H. Seiger |
| Stand der Arbeiten/Status continuing | Berichtsdatum/Last. Updating 12/31/77 | Bewilligte Mittel/Funds DM 195.874,-- |

## 1. General aim

Decrease of the sensitivity variations when ultrasonically testing reactor pressure vessels.

## 2. Particular objectives

Determination of the optimal thickness of the coupling layer for the mechanized ultrasonic inspection of reactor pressure vessels using the contact technique.

## 3. Research programme

For the determination of the optimal thickness of the coupling layer the following two contradictory effects are to be taken into consideration:

a) Decrease in the oscillations of the transmission factor with increasing thickness of the coupling layer.

b) Increase in the interferences of the frequency spectrum with increasing of the thickness of the coupling layer.

Thereby the following values are to be measured as a function of the probe location using the thickness of the layer as a parameter: reflector echo amplitudes, acoustic noise as interference level, echo frequency spectra. The thickness of the layer is increased in steps of 0.1 mm starting with 0 mm until the oscillations of the transmission factor fade away. The measurements are to be carried out on the pressure vessel specimen for 1 and 2 MHz using differently sized ultrasonic transducers and different beaming angles using the tandem and single probe techniques.

4. Experimental facilities, computer codes

Pressure vessel specimen, ultrasonic testing device, manipulator device, recorder, probes, path pick-up for recording the variations in the thickness of the coupling layer, holders.

5. Programme to date

The necessary probes and part of the equipment have been produced or ordered and, as far as possible, tested for their properties and faultless functioning.

6. Results

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7. Next steps

1. Establishing a detailed testing schedule for the measurements on the pressure vessel specimen.
2. Carrying out the measurements.
3. Evaluation of the measurements and documentation of the results.

8. Relation with other projects

RS 27

9. References

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10. Degree of availability of the reports

Gesellschaft für Reaktorsicherheit
Glockengasse 2
5 Köln 1

1241

Classification: 11.2.3.

| Title: Akustisk Emission fra Stålkonstruktioner | Country: DENMARK |
|---|---|
| | Sponsor: Risø National Laboratory |
| Title: Acoustic Emission from Steel Structures | Organization: Risø National Laboratory |
| Initiated date: 1977 Completed date: <br> Status: In progress | Scientists: W.E.. Swindlehurst <br> C.P. Debel |

1. General aim: To provide a basis for the evaluation of AE signals obtained during surveillance of steel structures.

2. Particular objectives: Weld metals and heat affected zones will be given particular attention.

3. Project status: Laboratory investigations as described in previous status document, are almost completed. Planning of data treatment methods has been started. Pressure vessel testing facility has been set up and the initial testing has begun.

4. Next steps: Completion of pressure- vessel test programme.

5. Relation with other projects: This project is related to industrial projects on application of the AE technique.

6. Reference documents:
Risø-M-1896. Arved Nielsen "AE surveillance Methods". AE as a supplementary tool for inspection of reactor pressure components".

7. Degree of availability: As decided by the sponsors (European Coal and Steel Community, Danish Teknologirådet).

| Classification 11.2.3. | |
|---|---|
| Title 1<br><br>Anvendelse af Akustisk Emission | COUNTRY Denmark<br>Risø National Lab.<br>SPONSOR et. al.<br><br>ORGANIZATION<br>Risø National Lab.<br>Danish Wel. Inst. |
| Title 2<br>Industrial application of Acoustic Emission | Project leader<br>Arved Nielsen |
| Initiated: 1970    Completed:<br>Status : progressing    Last updating | Scientists:<br>W. Swindlehurst<br>S.E. Iversen<br>N. Thorp |

## 1. General aim

Increase of the effeciency of non-destructive control and surveillance by application of AE methods.

## 2. Particular objectives

Development of a system to store on magnetic tape information from a large number of AE transducers in order to do a careful location analysis after testing or surveillance.

Development or testing of commercially available discriminators to operate with the above mentioned storing system or to operate in connection with other applications of AE than location.

## 3. Experimental facilities and programme

Suitable tape recorders and suitable electronic equipment is available. Large steel structures in which the need for advanced surveillance is severe are present particularly within the conventional power production.

Laboratory experiments are planned in order to provide information on materials properties with respect to acoustic activety. This is to form basis for further development of discriminators.

## 4. Project status

The storing system has been finished and is ready for field use when a set of discriminators have been build.

A more simple system for surveillance of smaller pressure vessels during periodic inspection is operating under field conditions.

## 5. Next steps

Development of more advanced discriminators and development of software to handle the information stored on tape.

## 6. -

## 7. Reference documents

A. Nielsen: Acoustic Emission Surveillance Methods. Risø Report No. 277.

A. Nielsen: European Progress Report on Acoustic Emission. July 6, 1974. IIW X-752-74.

## 8. Degree of availability

For commercial reasons the availability of information of the storing system might be restricted.

1245

Classification

11.2.2/11.2.3

| | |
|---|---|
| Title 1<br><br>Spændingsanalyse af primære trykbærende stålkomponenter: Sammenligning mellem beregnede og målte tøjninger og spændinger i en BWR pumpestuts. | COUNTRY Denmark<br><br>SPONSOR Risø National Lab.<br><br>ORGANIZATION Risø National Lab. |
| Title 2<br><br>Stress analysis of primary steelcomponents: A comparison between calculated and measured stress and strain in a BWR main circulation pump nozzle. | Project leader<br>S.I. Andersen<br><br>Scientists: |
| Initiated (date)<br>74.04.01<br><br>Status: concluded | Completed: 1977<br><br>Last updating(date)<br>April 1977 |
| | S.I. Andersen |

1247.

| Title 1 (Original language) Controllo della integrità dei recipienti in pressione mediante monitoraggio dell'emissione acustica. | Classification<br>11.2.3. |
|---|---|
| Title 2 (English) Assessment of nuclear pressure vessel integrity through acoustic emission monitoring. | Country ITALY<br>Sponsor ENEL<br>Organisation CISE |
| Date initiated October 1971<br>Date completed 1978<br>Last updating March 1977 | Project Leader<br>G. POSSA - F. TONOLINI |

1. General Aim

To develop non-destructive techniques based on acoustic emission monitoring for the assessment of nuclear pressure vessel integrity.

2. Particular Objectives

To develop acoustic emission monitoring systems and methods to be utilized 1) during pressure vessel hydrotests in cold unirradiated conditions, 2) during pressure vessel hydrotests in hot and possibly irradiated conditions, 3) during power operations as a continuous automatic pressure vessel surveillance.

3. Experimental facility and programme

3.1. Experimental facitilies: none

3.2. Programme

3.2.1. Development of an acoustic emission instrumentation system capable of detection and precise location of emitting defects during pressure vessel cold hydrotests in unirradiated conditions.

3.2.2. Development of an acoustic emission instrumentation system for the automatic continuous monitoring of a nuclear pressure vessel during power operation.

3.2.3. Installation of a prototype of the instrumentation system developed in 3.2.2. on a power reactor pressure vessel, with the main objective of determining the actual sensitivity in the presence of operational noises.

1248

| Title 1 (Original language) Controllo della integrità dei recipienti in pressione mediante monitoraggio dell'emissione acustica. | Classification 11.2.3. |
|---|---|

4. Project status

4.1. Progress to date

- (3.2.1.): 12 channel system for on-line acoustic emission source location designed, manufactured and tested; activity in progress on a 24 channel system.
- (3.2.2.): Prototype of the instrumentation system designed; components purchased or manufactured.
- (3.2.3.): Installation of a two channel instrumentation system prototype on the BWR Caorso pressure vessel to be completed within next May 1977.

4.2. Essential results obtained

Know-how and know-why regarding the technique based on acoustic emission monitoring for defect location in pressure vessels during hydrotests.

5. Next steps

Additional extensive tests of acoustic emission instrumentation in various steel pressure vessel hydrotests (3.2.1.).
Measurements of acoustic emission of the CAORSO BWR pressure vessel during the preoperational tests (3.2.2. & 3.2.3.).

6. Relation to other projects

None

7. Reference documents (Main titles)

1) E. Fontana, G. Grugni, B. Pirovano, G. Possa, F. Tonolini: "Controllo non distruttivo di recipienti a pressione mediante analisi dell'emissione acustica nel corso della prova a pressione idrostatica", Energia Nucleare, Vol. 21 n. 10, ottobre 1974.

2) E. Fontana, G. Grugni, C. Panzani, B. Pirovano, G. Possa, F. Tonolini: "Acoustic Emission Monitoring During Hydrotests of a Thin Wall Pressure Vessel", Energia Nucleare, Vol. 22 n. 5, maggio 1975.

3) E. Fontana et alii: "Acoustic Emission Measurements during the first pressure vessel hydrotest at ENEL - CAORSO BWR", Paper presented to the Conference "Periodic Inspection of Pressure Vessels" - London, Sept. 20/22, 1976.

8. Degree of availability

Various means of cooperation may be envisaged: for example, Italian participation in research programs carried out by partner countries, cost sharing, etc.

9. Budget

The estimate cost of this R&D program is about 200 millions lire per year.

1249

| | |
|---|---|
| PROJECT TITLE :<br>RELIABILITY OF CLASSICAL NDT | $\boxed{12.3}$<br>11.2.3 LWR<br>11.1 |
| SPONSORING COUNTRY :<br>Commission of the European Communities | ORGANISATION :<br>JRC Ispra |
| DATE INITIATED : 1977<br>DATE COMPLETED : 1980 | PROJECT LEADER :<br>S.J. CRUTZEN |

| N.V. KEMA | CLASSIFICATION : 11.2.3 |
|---|---|
| TITLE : Automatisch ultrasoon onderzoek van lassen en componenten | COUNTRY: THE NETHERLANDS |
| | SPONSOR : P.Z.E.M. |
| | ORGANIZATION : KEMA |
| TITLE (ENGLISH LANGUAGE): Automatic ultrasonic examination of welds and components | PROJECTLEADER : K. Boer |
| INITIATED : 1975 / LAST UPDATING : 1978 | SCIENTISTS : De Jong, Tempelman |
| STATUS : to be continued / COMPLETED : - | |

General aim

The performance of inservice inspection (ISI) by ultrasonic examination in accordance with the ASME-code and the Dutch-code.

Particular objectives

Development of an automatic system for the volumetric inspection by ultrasonic to remote areas where manual access is retricted for manual inspection operations.
Parts of the inspection system are special developed ultrasonic probes, manupulators, electronics and computer. The special purpose computer is used for control and acquiring all inspection data in accordance with the codes. On-line and off-line evaluation is possible.

Programm

The programm consists of:
- ultrasonic inspection of longitudinal, meriodinal and circumferential welds and nozzles of steamgenerators, pressurizer, pumps, main loops and reactor vessel.
- ultrasonic inspection of studs, threaded holes and ligaments of pumps and reactor vessel.

Project status

Tests are performed during the shutdown at February 1976, February 1977 and November 1977 for a PWR.

Next steps

The work is continued during the coming shutdown in 1978 and 1979 to complete the first interval of a PWR. Simular steps are taken for the second interval of a BWR.

Relation to other projects

Not applicable.

Reference documents

General description.

Degree of availability

Through the organization KEMA.

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX

1253

| NIL-MI-TNO | CLASSIFICATION : 11.2.1/ 11.2.3 |
|---|---|
| **TITLE :**<br>Literatuurstudie van materiaalparameters die nodig zijn bij de interpretatie van defect-localisatie met AE-apparatuur. | COUNTRY: THE NETHERLANDS<br>SPONSOR :<br>Ministry of Social Affairs and others |
| **TITLE (ENGLISH LANGUAGE):**<br>Literature survey of material parameters which are necessary for the diagnosis of defects localized with AE techniques. | ORGANIZATION :<br>NIL-MI-TNO<br>PROJECTLEADER :<br>Kloots |
| INITIATED : Sept. 1974 — LAST UPDATING : May 1978<br>STATUS : Completed — COMPLETED : June 1978 | SCIENTISTS :<br>Boerstoel<br>v.d. Brink |

| | |
|---|---|
| | Classification<br>11.2.3 |
| Title 1<br>DEVELOPMENT OF ACOUSTIC EMISSION MEASUREMENT | COUNTRY<br>UNITED KINGDOM<br>SPONSOR UKAEA<br>ORGANIZATION<br>REML RISLEY |
| Title 2 | Project Leader<br>P BENTLEY |
| Initiated 1969 Completed :<br>Status : Last updating | Scientists: |

Description:

1. General Aim

To determine from measurements made during a non-destructive pressure test, whether a vessel has a significant crack.

2. Particular Objective

To characterise emissions from cracks in many practical test conditions.

3. Experimental Facilities

The mobile laboratory is being used . locate signal sources during tests on vessels and small test pieces. Further development is necessary to permit identification of the types of acoustic signals due to the differing types of defect eg, cracks, inclusions - slag or porosity, crack growth, localised yielding, brittle cracking and ductile tearing.

To permit the signals produced in different testz to be compared in a more quantitative manner, calibration devices are under development.

Reference Documents

Internal documents.

1257

Classification

11.2.3

| Title 1 | COUNTRY UNITED KINGDOM |
|---|---|
| DEVELOPMENT OF HOLOGRAPHY AND LASER SPECKLE PHOTOGRAPHY | SPONSOR UKAEA |
| | ORGANIZATION SRD CULCHETH |
| Title 2 | Project Leader D J V MARTIN |
| Initiated 1973 Completed : | Scientists: |
| Status : Last updating | |

Description:

1. General Aim

To determine from measurements made during a non-destructive pressure test whether a vessel has a significant crack.

To evaluate strain by laser speckle photography.

2. Particular Objectives

To characterise holograms of typical vessel regions - membrane, junction, nozzle, etc - with and without cracks.

To holographically inspect regions of a reactor pressure vessel.

3. Experimental Facilities

The photographic equipment - including a pulsed laser - is being assembled. Tests are in hand on: a 5' vessel at Culcheth; a 3" test plate in a 4000 T machine at Risley; a 2" test plate in a machine at Windscale; any thickness of test plate (currently 3") in the hydraulic pressuriser at Springfields. Part through defects in a series of small cylinders are being hologrammed. Proposals for the inspection of a reactor pressure vessel have been issued.

Reference Reports

AEA internal reports.

Martin D J V    Holographic method giving stress levels and visualisation of defects in thick cylinders. To be published.

Martin D J V    Holographic interferometry and laser speckle photography as aids to assessment of pressurised components. Paper No. G3.10 at 3rd Int Conf on Structural Mechanisms in Reactor Technology, September 1975.

1259

Classification: 11.2.4.

| | |
|---|---|
| Title: Trykprøvningsfacilitet | Country: DENMARK |
| | Sponsor: Risø National Laboratory |
| Title: Pressure Testing Facility | Organization: Risø National Laboratory |
| Initiated date: 1977 Completed date: 1978<br>Status: In progress | Scientists: C. Debel |

1. General aim: To provide a facility for testing to failure of intermediate size vessels under controlled conditions.

2. Particular objectives: Testing of experimental vessels particularly under gas pressure in order to facilitate the testing at different temperatures.

3. Experimental facilities and programme: The aim is to provide a facility for fracture mechanics and acoustic emission experiments on vessels.

4. Project status: Designing is in progress.

5. Next steps: Construction.

6. Relation to other projects: The facility should be available for the projects Dynamic Fracture Mechanics and Acoustic Emission.

7. Reference documents: None so far.

8. Degree of availability: In principle no limitations.

496 1

| Title 1 (Original language) | Classification |
|---|---|
| Programma di ricerca sul comportamento del calcestruzzo per contenitori di reattori nucleari. | 11.3.1 |
| **Title 2 (English)** | Country: ITALY |
| Research Program on Concrete for Nuclear Reactor Vessels | Sponsor: ENEL<br>Organisation: ENEL |
| Date initiated: 1970<br>Date completed: -<br>Last updating: April 1977 | Project Leader<br>P. Bertacchi |

Description:

1. General Aim

Determination of concrete properties to provide data for prediction of vessel behaviour.

2. Particular Objectives

- Systematic investigation on the behaviour of concrete subjected to multiaxial stresses.
- Determination of concrete strength after wet and dry thermal treatments
- Study of concrete creep.

3. Experimental Facilities ad Programme

All the tests are carried out both at ENEL - Niguarda Laboratory (Milan) and at ISMES Laboratory (Bergamo).
These investigations are carried out in the framework of a joint research program with CEGB - CERL.

4. Project Status

- Systematic tests have been carried out on concrete specimens (cubical and cylindrical) subjected to bi-and triaxial stresses with an aim at determining the "rupture surface".
- Investigations have been carried out on the mechanics of onset and propagation of micro-cracks to gather information on the possibility of defining the limit of concrete elastic behaviour under multiaxial stress conditions.

1262

| Title 1 (Original language) Programma di ricerca sul comportamento del calcestruzzo per contenitori di reattori nucleari | Classification 11.3.1 |
|---|---|

Sonic methods have mainly been used with equipment for measuring propagation velocity of costant-frequency signals.

- A large number of specimens have been subjected to thermal treatment at temperatures of 90° and 120° C, both in dry and in moist ambients; moreover, they have been subjected to thermal cycling and to continuous treatment.
  The effects of thermal treatment on the bending and compression strength and on Young's modulus (static and dynamic) have been determined.

5. Next steps.

a. Study of concrete creep.
The installation of automatic load control equipment at the Niguarda Laboratory, with the possibility of subjecting concrete prismatic and cylindrical specimens up to a max load of 100 tons, makes it possible to start systematic research in order to determine the specific surface of creep and to study the influence exerted on concrete creep by various factors, such as the shape and size of the specimens subjected to loads which are kept constant with time.

b. Development of novel techniques (optical holography) for determining the state of strain and early stage of fissuring in concrete.

6. Relation to other Projects

Development of advance solutions for pre-stressed concrete pressure vessels (thin-wall solutions) (11.3.2 - 11.3.4)

7. Reference Documents


- R. Bellotti, "Joint Research Programme of ENEL-DSR/CEGB-CERL on concrete for Nuclear Reactor Vessels- Enel Contribution. State of the research as at December 1973." CRIS n. 2425.
- R. Bellotti, P. Rossi, "New Prospects for Evaluating the Degree of Safety in Concrete Structures Subjected to Multiaxial Stresses". III - 5; Seminar on: Concrete Structures Subjected to Triaxial Stresses - 17 th - 19 th May, 1974 - ISMES - Bergamo.

12 63

Classification
11.3.2/11.3.4

| Title 1 | COUNTRY Denmark |
|---|---|
| Brudundersøgelse af Betontank-lågmodeller | SPONSOR Risø National Laboratory |
| | ORGANIZATION<br>Risø National Laboratory |
| Title 2<br>Overload behaviour and ultimate load capacity of PCRV-closures for BWR-plants. | Project leader<br>S.I. Andersen<br>Scientists: |
| Initiated (date) Completed: 1976<br>Status: concluded Last updating (date) April 1977 | N.S. Ottosen<br>S.I. Andersen |

1. General aim. To study the overload behaviour, failure model and ultimate load capacity of PCRV-closures for a Nordic BWR-PCRV reference design.

2. Particular objectives. To investigate the influence from different design parameters, such as depth-to-span ratio, reinforcement and supporting flange geometry, etc., and to propose an optimized closure design.

3. Experimental facilities and programme. The test facility, situated at Risø, includes a steel pressure vessel, in which model specimens in scale 1:11 of the reference vessel closure can be pressurized to a max. hydraulic pressure of 450 bars.

The programme included tests on 9 different closure models and a series of comparative calculations by means of a finite element computer programme P-479, which has been developed by Risø for the analysis for PCRV-structures, taking into account the effects of concrete creep, plasticity and cracking together with steel plasticity. A high degree of experimental verification of the programme has been achieved from these tests.

4. Project status. 9 closure models have been tested, and good agreement between calculations and experimental data has been obtained, provided the plasticity and cracking of the concrete and plasticity of the steel parts

are taken into account. A new failure criteria and a proposal for a constitutive model for concrete are used by the finite element program.

5. Next steps. None.

6. Relations with other projects. The programme is part of a joint Nordic development work on a PCRV for BWR application.

7. Reference documents.

Ultimate load behaviour of PCRV top closures.
S.I. Andersen, N.S. Ottosen.
Paper H 4/3, 2. Int. Conf. on Struct. Mech. in Reactor Technology
Berlin (1973).

Theoretical and Experimental Studies for Optimization of PCRV Top Closures.
N.S. Ottosen, S.I. Andersen.
Paper H 3/6, 3. Int. Conf. on Struct. Mech. in Reactor Technology
London (1975).

A Failure Criterion for Concrete
N.S. Ottosen
Journal of the Engineering Mechanics Division, Proceedings of the American Society of Civil Engineers. Vol. 103, No. Eh4, Aug. 1977.

Structural Failure of Thick-Walled Concrete Elements.
N.S. Ottosen, S.I. Andersen.
Paper H 4/3 4. Int. Conf. on Structural Mech. in Reactor Technology.
San Francisco (1977).

8. Degree of availability

A limited amount of the results may be available on an exchange basis.

1265

Classification

11.3.2/11.3.4

| Title 1 | |
|---|---|
| Brudundersøgelse af Betontank-lågmodeller | COUNTRY Denmark |
| | SPONSOR Risø National Laboratory |
| | ORGANIZATION Risø National Laboratory |
| **Title 2** | **Project leader** |
| Overload behaviour and ultimate load capacity of PCRV-closures for BWR-plants. | S.I. Andersen |
| | Scientists: |
| Initiated (date) — Completed: 1976 | N.S. Ottosen |
| Status: concluded — Last updating (date) April 1977 | S.I. Andersen |

1267

| | |
|---|---|
| Title 1 (Original language) Sviluppo di soluzioni avanzate per i recipienti in pressione in calcestruzzo precompresso (soluzioni a pareti sottili). | Classification<br>11.3.2<br>11.3.4 |
| Title 2 (English) Development of advanced solutions for Pre-stressed Concrete Pressure Vessels (Thin-wall solutions). | Country ITALY<br>Sponsor ENEL<br>Organisation ENEL, ISMES, CISE |
| Date initiated 1970<br>Date completed --<br>Last updating April 1977 | Project Leader<br>F. Scotto |

General Aim

To develop advanced economical and safe solutions for PCPVs by better exploiting the resistance of triaxially stressed concrete.

2. Particular Objectives

Reduction of the wall thickness of current PCPVs by applying novel design philosophies and tendon layouts.

Development of a pre-stressed concrete top lid and of a closure-and-sealing system similar to those adopted for steel pressure vessels.

3. Experimental Facilities and Programme

The proposed solutions are investigated by means of suitable design tools, such as physical and mathematical models of the structure and rheological models of the material (concrete).
Most of the tests are carried out at the ISMES Laboratory at Bergamo. Sophisticated small-scale physical models (scale 1:10, 1:20) have been developed, built and tested to cover the entire range of structural behaviour up to collapse.
In order to obtain additional information on the deformation states of the physical models a new approach, based on holographic techniques will be tried.

4. Project Status

a. Progress to date

The following models have been built and tested up to collapse:
- One thick-and three thin-wall models (scale 1:20) for a 300-MWe HTR.
- One thin-wall "continuous" model (scale 1:10) for a 1000-MWe BWR.
- Two thin and one "super-thin" models of the top lid (scale 1:10) for a 1000 MWe BWR.
- One integral model (main structure plus removable lid) scale 1:10 for a 1000 MWe BWR.

Use has been made of mathematical models for the selection of the optimal geometrical solutions by means of an interactive type of man-machine analysis.
A computer program was developed to take into account also the frictional interaction between the lid and the main cylindrical structure.

| Title 1 (Original language) Sviluppo di soluzioni avanzate per i recipienti in pressione in calcestruzzo precompresso (soluzioni a pareti sottili). | Classification<br>11.3.2<br>11.3.4 |
|---|---|

b. Essential Results
The proposed thin-wall solution can lead to a reduction in the PCPV weight from about 37,000 tons to 14,000 tons for a 300-MWe HTR, and from 8000 to 4000 tons for a 1000 MWe BWR.

5. Next Steps
   a. Experimental and theoretical verification of the designs.
   b. Recycling of the results of the physical model tests to the mathematical model.
   c. Testing of the final solution of the thin-walled PCPV by means of one or more physical models (scale 1:10).
   d. Application of thin walled concept to Gas Cooled Fast Breeder Reactors.

6. Relation to Other Projects
   a. "Research Program on concrete for nuclear reactor vessels" (11.3.1)
   b. ENEL-A.B. Atomenergy Agreement concerning "Collaboration for the Development of PCPV for LWRs".

7. Reference Documents

- F. Scotto, "Triaxial State of Stress of Thin-Walled PCPVs for HTGRs; Comparison with a Conventional Thick Solution". IABSE Seminar on Concrete Structures Subjected to Triaxial Stresses, Bergamo, 17-19 May, 1974.
- M. Fanelli et al., "Finite-Element Analysis of Prestressed - Concrete Pressure Vessels". IABSE Seminar, 17-19 May, 1974.
- E. Fumagalli and G. Verdelli, "Small-Scale Models of PCPVs for High-Temperature Gas Reactors. Modelling Criteria and Typical Results". IABSE Seminar, 17-19 May 1974.
- F. Scotto, "Thin Walled" Concept and a new top lid applied to the Scandinavian PCRV for a boiling reactor. Paper (H3/4) presented at III SMIRT* - London, September 1 - 5, 1975.
- R. Riccioni, G. Robutti and F. Scotto, "Finite Element Structural Analysis of a P.C.P.V. for a BWR. Paper (H2/7) presented at III SMIRT* - London - September 1 - 5, 1975.
- E. Fumagalli, G. Verdelli, "Research on PCPV for BWR" - Physical model as design tool - Main Results. Paper (H3/5) presented at III SMIRT* London - September 1 - 5, 1975.

* SMIRT - Conference on structural mechanics in reactor technology.

1269

| Title 1 (Original language) Sviluppo di soluzioni avanzate per i recipienti in pressione in calcestruzzo precompresso (soluzioni a pareti sottili). | Classification<br>11.3.2<br>11.3.4 |
|---|---|

8. Degree of availability
Exchange of information can be arranged.

9. Budget
About $60 \cdot 10^6$ Lire/year.

1271

| | |
|---|---|
| Title 1 (Original language) Sviluppo di soluzioni avanzate per i recipienti in pressione in calcestruzzo precompresso (soluzioni a pareti sottili). | Classification $\boxed{11.3.2}$ 11.3.4 |
| Title 2 (English) Development of advanced solutions for Pre-stressed Concrete Pressure Vessels (Thin-wall solutions). | Country ITALY<br>Sponsor ENEL<br>Organisation ENEL, ISMES, CISE |
| Date initiated 1970<br>Date completed --<br>Last updating April 1977 | Project Leader<br><br>F. Scotto |

CLASSIFICATION
11.4.1

| | |
|---|---|
| TITLE 1 ENCEINTE DE CONFINEMENT POUR PWR. DOUBLE ENCEINTE SANS PEAU D'ETANCHEITE. | COUNTRY FRANCE<br>SPONSOR E.D.F.<br>ORGANIZATION E.D.F. |
| TITLE 2 PRIMARY CONTAINMENT FOR PWR WITHOUT METALLIC SKIN FOR TIGHTNESS. | Project Leader<br>E.D.F./SEPTEN/M<br>Scientists<br>M. COSTAZ |
| Started 1970 Completed 1975<br>Status Last updating : 20.01.75 | |

I - GENERAL AIM . Définition d'un nouveau type d'enceinte de confinement plus économique, sans peau d'étanchéité. Enceinte double en béton.

II - PARTICULAR OBJECTIVES

La conception et la réalisation des peaux d'étanchéité métalliques d'enceintes en béton armé ou précontraint présentent de nombreuses difficultés provenant de la nécessité de limiter l'épaisseur des tôles à des valeurs faibles (6 ou 9 mm) pour des raisons économiques.

Un nouveau système de confinement a été mis à l'étude, il est composé de deux parties :

- une enceinte interne en béton précontraint, sans peau d'étanchéité métallique
- une enceinte externe en béton armé.

En cas d'accident, les fuites traversant l'enceinte interne, sont récupérées dans l'espace annulaire et filtrées avant rejet.

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1274

## III - EXPERIMENTAL FACILITIES AND PROGRAMME

Les études et moyens d'essais associés sont :

- étude de la structure sous l'angle du Génie Civil,
- essais d'étanchéité du béton,
- essais d'étanchéité des peintures,
- essais de contrôle d'étanchéité du béton en cours de construction.

## IV - PROJECT STATUS

### 4.1 - Progress to date

- étude d'un avant projet détaillé d'enceinte (1973)
- essais d'étanchéité du béton armé ou précontraint (1973).

### 4.2 - Essential Results

Faisabilité démontrée.

Choix par E.D.F., en accord avec les Autorités de Sûreté Française, de ce type d'enceinte pour équiper les tranches nucléaires à venir W 1350.

## V - NEXT STEPS

- études d'avant projet détaillées de l'enceinte W 1350,
- essais d'étanchéité de revêtements élastomères sur béton,
- essais concernant le contrôle de l'étanchéité du béton en cours de construction.

## VI - RELATION WITH OTHER PROJECTS

Néant.

...

## VII - REFERENCE DOCUMENTS

Note E.D.F. SEPTEN GC 72-30. Double enceinte sans peau. Spécification technique.

- note E.D.F. SEPTEN GC 73-01. Double enceinte sans peau. Synthèse des études et essais réalisés en 1972,
- note E.D.F. SEPTEN GC 73-09. Présentation et analyse des essais réalisés.

## VIII - DEGREE OF AVAILASILITY

Internes E.D.F.

1277

| Berichtszeitraum/Period<br>January - December 1977 | Klassifikation/Classification<br>11.5 | Kennzeichen/Project Number<br>PNS 4239 |
|---|---|---|
| Vorhaben/Project Title<br>Untersuchungen zum Einfluß der Größe und Form von Kühlkanalblockaden in der Flutphase eines Kühlmittelverluststörfalles<br><br>Influence of the Size and Shape of Coolant Channel Blockages upon Core Cooling in the Reflood Phase of a LOCA | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>KfK<br>Projekt Nukleare Sicherheit (PNS)<br>Institut für Reaktorbauelemente (IRB) |
| Arbeitsbeginn/Initiated<br>1972 | Arbeitsende/Completed<br>1979/80 | Leiter des Vorhabens/Project Leader<br>S. Malang |
| Stand der Arbeiten/Status<br>continuing | Berichtsdatum/Last Updating<br>December 1977 | Bewilligte Mittel/Funds |

1. General Aim

The influence of coolant channel blockages upon the cooling effect during the reflood phase of a loss of coolant accident is investigated.

2. Particular Objectives

The objective of the program is to investigate the influence of size and shape of coolant channel blockages on the flow and the heat transfer conditions in the vicinity of blockages. Not considered is the influence of the cooling conditions on the propagation of the cladding deformation.

3. Research Program

The program consists of three major steps to investigate separate effects of the reflood cooling conditions in a PWR-geometry:
(FEBA, Flooding Experiments with Blocked Arrays)

3.1 Experiments with a 5-rod row, all subchannels blocked by the same blockage ratio, variation of size and shape of the blockages. These tests serve mainly for optical observation of the two-phase flow and for qualitative studies of the cooling effects.

3.2 Experiments with a 25-rod bundle.
- Similar objectives as in 3.1, but quantitative results.
- First tests to study the two-phase flow bypassing blockages.

3.3 Experiments in larger bundles up to 50 rods, with some coolant channels partly blocked to study the effect of flow redistribution.

4. Experimental Facilities, Computer Codes

The test rig is designed for the separate effect test program simulating the reflood phase of a LOCA in a PWR, excluding system effects.

The heat transfer analysis code is calculating the instantaneous values of the stored heat, surface temperature, heat flux and heat transfer coefficients. In a further option the simulation quality of the used heater rods can be calculated.

5. Progress to Date

Tests have been performed with the 5-rod row using grid plates as blockage simulators. To complete step 3.1, mentioned above, rods with sleeve blockages have been prepared. In preparation of the tests with the 25 rod bundle (step 3.2), heater rods and test section have been ordered.

The evaluation of the tests performed was continued and the additional option of this code allows the parallel calculation of a fuel rod and a heater rod with the assumption of identical cooling conditions. With this option the simulation quality of the used heater rods has been evaluated.

6. Results

The results obtained up to now by tests with a five rod row and plate blockages allow the following preliminary conclusions:

Compared are tests with identical subchannel mass flow.

Upstream of the blockage the heat transfer is unchanged compared with the unblocked case with the same local mass flow.

Downstream of the blockage the heat transfer is improved. Decreasing improvement was observed with increasing distance downsteam of the blockage.

At lower flooding rates or lower system pressure the heat transfer decreases in general.

The evaluation of the simulation quality has shown that during the reflood phase the fuel rods are well simulated by the heater rods used.

7. Next Steps

Two main problems are investigated with the next experiments:

- Influence of the shape of the blockage on local heat transfer, to be investigated in the 5-rod row and the 25-rod bundle, all coolant channels partly blocked.
- Influence of the flow distribution on the local heat transfer in bundles with some partly blocked coolant channels surrounded by undisturbed bundle geometry.

8. Relation to Other Projects

Low pressure tests performed by KWU.

9. References

PNS-Semi-Annual Reports (German, Engl. Abstract):
1976/1 KFK 2375, November 1976
1976/2 KFK 2435, April 1977
1977/1 KFK 2500, December 1977

P. Ihle, S. Malang, K. Rust, H. Schmidt,
"Der Einfluß von Kühlkanalblockaden auf den Wärmeübergang während der Flutphase eines Kühlmittelverluststörfalles", Reaktortagung Mannheim, 29. März - 10. April 1977, Deutsches Atomforum e. V., Kerntechnische Ges. im Dt. Atomforum e. V., Leopoldshafen 1977: ZAED pp. 201 - 204.

P. Ihle, S. Malang, K. Rust,
"Thermohydraulic Tests With Bundles of Ballooned Rods Simulating the Reflood Phase of a LOCA", ANS-Thermal Reaktor Safety Meeting, July/ Aug. 1977, Sun Valley, ID, USA.

P. Ihle S. Malang, K. Rust,
"Thermohydraulic Tests With Bundles of Ballooned Rods", I. Mech. E.-Conf. on Heat and Fluid Flow in Water Reactor Safety, Sept. 1977, Manchester, GB.

S. Malang, K. Rust,
"Heat Transfer Analysis of Experiments Simulating a Loss-of-Coolant Accident", I. Mech. E.-Conf. on Heat and Fluid Flow in Water Reactor Safety, Sept. 1977, Manchester, GB.

P. Ihle, S. Malang, K. Rust,
"Reflood Experiments with Bundles of Ballooned Rods", Symposium on the Thermal and Hydraulic Aspects of Nuclear Reactor Safety, 1977 Winter Annual ASME-Meeting, Atlanta, Ga., USA.

10. Degree of Availability of the Reports

Unrestricted Distribution.

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. 77 - 31. 12. 77 | 11.5 | RS 194 |
| Vorhaben/Project Title | | Land/Country |
| Blockierte Kühlkanäle<br><br>Blocked Cooling Channels | | FRG |
| | | Fördernde Institution/Sponsor<br>BMFT |
| | | Auftragnehmer/Contractor<br>KRAFTWERK UNION AG<br>Reaktortechnik<br><br>R 513, Erlangen |
| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
| 1. 7. 76 | 30. 6. 78 | D. Hein |
| Stand der Arbeiten/Status | Berichtsdatum/Last Updating | Bewilligte Mittel/Funds |
| Continuing | 31. 12. 77 | 1'048.800,-- DM |

1. General Aim

Investigation on the effect of ballooned fuel claddings during a LOCA on the emergency cooling of a BWR.

2. Particular Objectives

Based on the measured heat transfer rates in the blocked region, the influence of the flow distribution on the heat transfer will be evaluated upstream and downstream of the blocked region as a function of the geometry of the blockage.

3. Research Program

- Planning and preparation of blockage tests
- Modification of the test facility for BWR geometry blockage tests
- Assembly of the bundle in the rod bundle container
- Performance of tests with 37 % and 70 % blockage
- Evaluation of tests

4. Test Facility

The BWR tests will be performed in a double-bundle test facility which was installed within the scope of Task RS 36. During test performance, the cooling channels of one bundle will partly be blocked by sheets. The second, parallel bundle remains unblocked.

The following parameters will be investigated: initial temperature, pressure, power, coolant mass-flow and degree of blockage.

5. Progress to Date

The two test series comprised 24 spray- and flooding tests. The maximum temperatures of partially blocked bundles (37 % and 70 %) were compared with test runs of the project RS 36 C (blockage 0 %).

The parameters were:

| | |
|---|---|
| System pressure | 1, 5, 10 bar |
| Initial tube cladding temp. | 600, 800 °C |
| Flooding rate | 1,5; 2; 3,3 cm/s |
| Spray mass | 0,68; 1,02 $m^3/h$ |

The quench times and heat transfer coefficients were determined and compared with the results of lower blockage tests.

A 9 rod assembly was prepared for testing new measuring equipment in the bundle.

6. Results

The comparison of the maximum temperatures and quench times of the blockage tests and the non blocked tests showed that the influence on the cooling process is rather low with 37 % and 70 % blockage. The heat transfer in the blocked region is improved.

7. Next Steps

The 9 rod assembly will be set into operation. Special measuring equipments will be tested.

8. Relation with Other Projects

RS 36 C

9. References

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10. Degree of Availability

---

| Title 1 (Original language) | Classification |
|---|---|
| Effetti dello scoppio di tubo a pressione CIRENE. | 11.5 |
| **Title 2 (English)** | Country ITALY<br>Sponsor : CNEN<br>Organisation : CISE |
| CIRENE power channel pressure tube burst tests. | |
| Date initiated October 1970<br>Date completed 1978<br>Last updating May 1977 | Project Leader<br>G. Possa |

1. General aim: to investigate the consequences of the hypothetical explosion of a power channel pressure tube for the CIRENE reactor.

2. Particular objective: to measure the pressure waves produced by explosion in the $D_2O$ tank and to determine the associated stresses on major mechanical structural components.

3. Experimental facility and programme

3.1. Experimental facility.

- BETULLA: facility located in CCR Euratom at Ispra including a pressure vessel (with nearly CIRENE dimensions) for explosion containment.

3.2. Programme

3.2.1. Preliminary burst tests to individuate most relevant plant parameters.

3.2.2. Burst tests in full height scale with a dummy explosion tube (full length rupture).

3.2.3. Burst tests as in 3.2.2. with explosion tube and adjacent target channel simulating in full scale (dimensions and structural materials) the CIRENE power channel.

3.2.4. Numerical simulation with existing computer codes of the transient triggered by explosion.

Remark Burst tests for determination of the consequences of a hypothetical pressure tube rupture due to hot spot have been canceled.

4. Project status

4.1. Progress to date

- (3.2.1.): completed
- (3.2.2.): completed
- (3.2.3.): completed
- (3.2.4.): in the initial stage

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| Title 1 (Original language) | Classification |
|---|---|
| Effetti dello scoppio di tubo a pressione CIRENE. | 11.5 |

4.2. Essential results

- Basic understanding of the dynamics of underwater explosions in a confined volume.
- Understanding of the rupture process of a pressurized Zircaloy tube artificially defected, under typical conditions of a pressure tube reactor.
- Gathering information on explosion stresses in reactor structural components.

5. Next steps

Programme item 3.2.4.

6. Reference documents


1) M. Famiglietti, A. Parmeggiani, G. Possa, L. Galbiati: "Pressure Burst Due to Power Channel Explosion in a Pressure Tube Reactor", presented at the 3rd SMIRT Conference, London 1-5 Sept. 1975.


7. Degree of availability: to a limited extent.

# 12. QUALITY ASSURANCE

1285

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.10.77 - 31.12.77 | 12.1 | RS 317 |

| Vorhaben/Project Title | |
|---|---|
| Status analysis of quality assurance systems of the operators for construction and operation of nuclear power plants<br><br>Statusanalyse der Qualitätssicherungssysteme bei den Betreibern für Herstellung und Betrieb von Kernkraftwerken | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: VGB Essen<br><br>HA "Wärmekraftwerke"<br>AK "Statusbericht QS" |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.10.77 | 30.9.78 | Dipl.-Ing. Plate |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | 31.12.77 | 353.017,-- DM |

## 1. General Aim

Determination of the quality assurance (QA) over all phases of the nuclear power production (design, construction, operation). Deficiencies between individual QA-systems and/or insufficient single systems influence the entire system. Therefore also a description of the QA-systems of the power plant operators and following preparation of a general status report is necessary.

## 2. Particular Objectives

Survey on measures of the power plant operators. Detailed description of the existing condition, on activities of the operators in the scope of the QA.

## 3. Research Program

Elaboration to the following technical fields:

3.1 General viewpoints on the QA of nuclear power plant-primary circuit components at layout, manufacturing and construction, start-up and operation, including maintenance.

3.2 Performance of the QA measures

3.2.1 System of the QA of the operator at manufacturing, construction, start-up and operation of nuclear power plant facilities.

3.2.2 Quality organization
3.2.3 Ordering
3.2.4 Design principles
3.2.5 Construction and calculation
3.2.6 Fabrication and testing
3.2.7 Transportation, packing, preservation
3.2.8 Construction
3.2.9 Start-up
3.2.10 Operation
3.2.11 Measures in the cases of quality miss match
2.3.12 Maintenance
3.2.13 Alteration and extrusion of plant parts
3.2.14 Fuel elements
3.2.15 Repeating tests
3.2.16 Documentation
3.2.17 Control of the QA-systems of the operator and his deliverers
3.3 Valuation of the QA-measures, presently performed by the operator and within the scope of the licensing procedure

4. Experimental Facilities and Computer Codes

5. Progress to Date

Looking for existing documents, delegation of tasks to the members of the project-group, in accordance to program

6. Results

–

7. Next steps

to 3.- Preliminary elaboration for project, with detail division for all chapters, under coordination between the participating energy supply corporations, project aim will probably be obtained without having the printed final version.

8. Relation with other Projects

RS 124 KWU

as well as analogue orders of BMI

9. Literature

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10. Degree of availability of the Reports

–

| TITLE 1 (original language)<br>Programma di garanzia della qualità | Classification<br>12.1 |
|---|---|
| TITLE 2 (english)<br>Quality assurance program | Country: ITALY<br>Sponsor: Franco Tosi<br>Organisation: Franco Tosi |
| Date initiated January 1, 1972<br>Date completed June 6, 1975<br>Last updating | Project Leader<br>G. Attanasio |

General aim and particular objectives

The purpose of this project is to formulate a Quality Assurance System and to suit it to the organization and the procedures of the Sponsor, were possible; otherwise to modificate organization and procedures.

The outcome of this work is contained in a Quality Assurance Manual which contains a detailed description of the Quality Assurance System.

The Quality Assurance System is becoming working with the implementation of the orders for Enel V and VII nuclear power units.

Project status : completed

Degree of availability: for customers only

| PROJECT TITLE : RELIABILITY STUDIES | CLASSIFICATION 12.1 |
|---|---|
| SPONSORING COUNTRY : ITALY | ORGANISATION : CNEN |
| DATE INITIATED : October 1974<br>DATE COMPLETED : In progress | PROJECT LEADER : G. TOMASSETTI |

Description : A research has been initiated in the field of reliability studies focused on components (electronic and mechanical) of water reactors.

Particular attention will be given to failure mode analysis of mechanical components of experimental loops and to the methods for collecting failure data with the aim to evaluate the possibility of using, in the most suitable way, data banks.

30 juin 1978

| | |
|---|---|
| | Water reactors 12.3 |
| Caractérisation des équipements ultrasonores | Belgium |
| | |
| Characterization of ultrasonic equipments | Association Vinçotte |
| | P. CAUSSIN |
| 1975 | |
| In progress | |

## 1. General aim

Development of the measurement techniques of ultrasonic equipments characteristics for assessing the performances and the reproductibility of inspection:

## 2. Particular objectives

Development of accurate procedures in compliance with the US-NRC 10 CFR 50 and ASME Code. Analysis of the actual characteristics influence on the performances achieved.

## 3. Experimental facilities and program

Facilities :

- Acquisition of electronic measuring equipments (spectrum analyser, vect impedance meter, ..)
- Development of equipments for measuring the ultrasonic beam spread in steel for metal paths ranging between 0 and 350 mm, and refracted angle: between 0 and 80°
- Development of electronic equipments for recording the beam spread measurement results and for gating the echoes.

Program :

- Development of procedures for accurately checking the characteristics of ultrasonic equipments : apparatus, cables, probes.
- Establishment of a scheme for the periodic inspection of the equipments.
- Analysis of the influence of the equipment characteristics on the performances of the inspections.

## 4. Project status

The measuring equipments are available and the characterization scheme has been defined.
It was established that :
- the characteristics stongly affect the performances of the inspections
- commercially available ultrasonic equipments, which are nominally identical can have very different characteristics and performances.

## 5. Next steps

Quantitative analysis of the influence of different characteristics on the performances and the reproductibility of ultrasonic inspections.

## 6. References

Characterization scheme of an ultrasonic equipment for industrial application

P. CAUSSIN

Proceedings of the Ispra Courses on Characterization of Ultrasonic Equipment, CEC-JRC-Ispra, June 1-3, 1977, 37 p.

## 7. Availability

Details available at

ASSOCIATION VINCOTTE
Département Etudes
B 1640 RHODE-SAINT-GENESE (Belgium)

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 10.1.1977-31.12.1977 | 12.3 | RS 222 |

| Vorhaben/Project Title | |
|---|---|
| Analysis of possible improvements of objectivity and documentation of non-destructive-testing methods for reactor components of the primary circuit<br><br>Analyse bestehender Materialprüfverfahren auf Objektivierbarkeit für Reaktorbauteile des Primärkreislaufes | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Dornier System GmbH Friedrichshafen |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 10.1.1977 | 31.5.1978 | Dr. Sahm |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | 31.12.1977 | 739.426,- |

## 1. General Aims

Feasable methods of non-destructive-testing (NDT) during the work inspection of reactor components of the primary circuit are to be analysed either with respect to objectivity and automatic control of the measuring process either with respect to objective and automatic documentation of the test results. General aim of this analysis is to eliminate individual influences in the test process and to reduce time and cost consumption in NDT.

## 2. Particular Objectives

An extensive documentation of the actual NDT status is basically used to define the problems concerning objectivity and automation in NDT. A careful inventory of the applied inspection techniques will be especially important in this context. Solutions to the above problems will be given by a critical review of all feasable NDT methods. The usability of future inspection techniques not yet specified in NDT of reactor components, will be analysed with regard to physical limits of flaw detection and flaw diagnostic and to technical realization.

## 3. Research Program

Actual test situations are fundamental to the research program. Hence the first part deals with the definition of typical representative test problems. The analysis will then proceed in two steps. First the actual situation of NDT will be reviewed and secondary the future development will be analysed. The research program includes the following points:

2. Review of significant NDT procedures and their background during the work inspection of reactor components for different types of reactors
2.1 LWR
2.2 LMFBR
2.3 HTR
2.4 List of representative test problems
3. Definitions
3.2 Analysis of NDT methods
3.2.1 Objectivity of applied methods
3.2.2 Fundamentals and physical limits of NDT methods
3.2.3 Actual standard of NDT methods
3.3 Analysis of future development
3.4 Analysis of problems

## 4. Experimental Facilities, Computer Codes

## 5. Progress to Date

For the inventory of the actual test situation a prescribed form has been used in the first step to get an extensive information about any significant test problem during the production of reactor components of the primary circuit. Furthermore a flaw catalogue has been made up. In a second step the information has been concentrated in another prescribed form by summarizing similar test problems into typical representative test problems no more related to specific components.

## 6. Results

The prescribed form of the individual test situation contains information about components, drawings, materials, specimen geometry, test sequence, test methods, test specifications, extent of the test, automation and documentation. The relation between test method, test situation and magnitude of detected flaws has been extracted. The flaw catalogue relates the actual flaws in the specimens under test to the flaws detected by NDT methods. The information has been concentrated into typical representative test problems by comparison of the specific test problem with respect to test object, test sequence, material and geometry.

## 7. Next Steps

- Abstract of the test specifications which allows the minimum required flaw magnitude to be stated on the precribed forms of the representative test problems
- Determination of the actual status concerning objectivity and automation of the applied NDT methods by
  - a list of queries
  - interviews with NDT engineers
- Compilation of fundamentals and physical limits of applied and future NDT methods.

## 8. Relation with other Projects

- Qualitätssicherung - Darstellung des Istzustandes
- Studie über Personalqualifikation für zerstörungsfreie Prüfungen im Bereich Reaktorsicherheit (RS 216)

## 9. References

## 10. Degree of Availability of the Reports

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.1977 - 31.12.1977 | 12.3 | RS 102-17 |
| Vorhaben/Project Title<br>Automatisierung und EDV der US-Impulsecho-prüfung - Fehlerbildrekonstruktion -<br><br>Automation and electronic data processing of the testing with the US-pulse-echo method - reconstruction of flaw pictures - | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Fraunhofer-Ges., München<br>IzfP, Saarbrücken |
| Arbeitsbeginn/Initiated<br>15.9.1974 | Arbeitsende/Completed<br>31.12.1977 | Leiter des Vorhabens/Project Leader<br>O.A. Barbian |
| Stand der Arbeiten/Status<br>Completed | Berichtsdatum/Last Updating<br>December 1977 | Bewilligte Mittel/Funds<br>1.641.600,-- DM |

## 1. General Aim

During the production of reactor pressure vessels both, the basic materials and the single components undergo a nondestructive and fully volumetric test. After the completion of the reactor a basic test is made, and then, at certain intervals, fully automatic in-service inspections take place under more complicated conditions. This is to locate and evaluate defects. Evaluation is effected in a computer after the reconstruction of a three-dimensional flaw picture not dependent upon the different testing methods. These flaw pictures are compared with the specifications and standards.

## 2. Particular Objective

The aim of the investigations is the reconstruction of flaw pictures true as to locus and geometry. Only such reconstructions allow the application of fracture mechanical criteria or empirical rules for the evaluation of nondestructively detected flaws. To obtain such reconstructions the following steps have to be carried out:

- utilization of information included in echo-dynamics (local curce of transit time, local curve of amplitudes),

- utilization of information included in phase and amplitude of the signal,

1300

- adjustment of reconstruction models to natural defects must be better than the former one (for example finite cylinder, infinite cylinder, sphere, ellipse, circle),

- consideration of subliminal connexions of defects by means of neighbouring relations,

- consideration, respectively suppression of transfer fluctuations, coupling failures, form echoes, and redisturbances;

- the results are to be made available in a compressed manner and for all testing techniques simultaneously, i.e. as three-dimensional flaw picture, which allows sections in any direction.

## 3. Experimental Facilities, Computer Codes

For the realization of this concept it is necessary to work with probes, which have a high sensibility and dynamic range.
Moreover, the test systems have to work with different ultrasonic test methods as, i.e., tandem, pulse-echo, V-transmission, focus-tandem, focus-pulse-echo, etc.
An ultrasonic hardware has to be available, which allows the transfer of the necessary signals (for example, echo-amplitude + echo transit time, resp. the envelope of an echo-signal and, if necessary, the complete high-frequency signal) to the computer.
This work is done by IzfP as well as the experimental data recording at artificial and natural defects.
The software, necessary for data processing has to be developed by the associated institute IPW in Freiburg.

## 4. Project Status

The ultrasonic hardware components are completed and built up.
The function of the single components and the software program of the microprocessor are tested.
The IPW in Freiburg has completed the development of the software for the mentioned evaluation and reconstruction.

The following are the parts of the software program:

- determination of the local curves of amplitudes and transit time,
- compensation of the amplitude locus curves by parabolas,
- reconstruction of defect locus in B-picture,
- development and determination of signal locus curves of transit time,
- reconstruction of the boundary of the flaws by means of transit time locus curves.

For the reconstruction of the flaws in the far-field of piston diaphragms by models of reflectors and their sizes mathematical models have been developed which use signal locus curves for evaluation.
The following details are completed now:

- Development of a reconstruction model for voluminous reflectors (sphere and infinite cylinder) (pulse-echo and tandem),

- Development of a model for the classification of reflectors into the patterns sphere, cylinder, circle and strip. Two classification possibilities have been found:
  a) reduced locus curves
  b) distance-dependence of the different patterns.

- Calculation of amplitude-locus curves for all models (pulse-echo and tandem),

- Comparison of calculated and measured locus curves, which led to a good agreement.

- Development of a program in Fortran IV (Computer PDP 11) for the reconstruction of circle and strip (pulse-echo and tandem),

- Development of a program for the construction of reduced locus curves.

## 5. Next Steps

Final report of this project.

## 6. Relations with Other Projects

This program is partly based on the results of the reactor safety research project RS 27/1 and RS 27/2 and will be carried through in coordination with the still running projects RS 27/2 and RS 169.

## 7. References

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## 8. Degree of Availability of the Reports

-

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.1977 - 31.12.1977 | 12.3 | RS 169 |

| Vorhaben/Project Title | |
|---|---|
| Entwicklung und Bau einer Ultraschall - Prüf- und Auswerteelektronik einschließlich Vorverstärkerkasten.<br><br>Development and construction of an ultrasonic test- and evaluation electronic including preamplifier box | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Krautkrämer GmbH, Köln |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.7.1975 | 31.12.1978 | Ing.grad. G.Gutmann |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | December 1977 | DM 763,091.- |

## 1. GENERAL AIM

Developing an ultrasound electronic evaluation system to meet current safety requirements and being universally employable with all types of reactors.
On the basis of experiences gathered up to date and of the technical development foreseeable for the future this electronic evaluation system is to meet the following requirements:

## 2. PARTICULAR OBJECTIVES

Maximum number of channels: 60
Maximum travel rate of the manipulator mechanism: 100 mm/sec.
Scanning rate: one (1) shot per millimeter
Also the system shall continuously be monitored for performance and stability.

## 3. RESEARCH PROGRAM

- Development of the detailed performance scheme with network map.
- Development of a number of new analysing modules
  a) Min./max. value store
  b) Interference analyses
- Development and design of the electronic evaluation system in accordance with the stipulated requirements.
- Testing interaction of the developed modules.
- Planning work for prototype building.
- Building of the prototype.
- Investigations into bus debugging on the prototype.

4. PROGRESS TO DATE

The works of the research program as specified under 3. as well as the design of the testing and evaluation electronic system have been concluded.

5. RESULTS

All electronic modules especially required for this electronic evaluation system have been developed and tested as functional samples. The requirements relating to performance, accuracy, and stability of the modules were met.

Designing and development work for the complete task was continued, the following major requirements having been met:

a) maximum number of channels: 60
b) maximum travel rate of the manipulator mechanism: 100 mm/sec.
c) scanning rate of the testing system of one (1) shot per millimeter
d) fully-automatic and continuously effective performance and stability monitoring
e) maximum acceptable stability tolerance of ± 1.5 dB.

Designing work for the building of the prototype evaluation electronic was terminated.

Planning for pre-amplifier unit miniaturizing was terminated and comprised into a funds raise application.

6. NEXT STEPS

a) Pre-amplifier box
- changing the pre-amplifier box design after the funds raise application has been approved
- building the prototype

b) Electronic
- commencing prototype building
- testing the electronic system

c) Testing the complete system.

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 7/1/77 - 12/31/77 | 12.3 | R S 266 |

| Vorhaben/Project Title | Land/Country |
|---|---|
| Entwicklung und Bau einer Ultraschall-Prüfeinrichtung auf Laserbasis zur Prüfung von Reaktor-Komponenten<br><br>Development and construction of ultrasonic test-equipment based on lasers for testing reactor components | FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Krautkrämer GmbH, Köln |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 7/1/77 | 12/31/79 | Dipl.-Ing. W. Kaule |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | 12/31/77 | 2.413.106,-- DM |

## 1. General aim

With conventional ultrasonic reactor tests the probes used for transmitting and receiving sound have to be coupled to the material to be tested by means of a coupling agent. Lasers used as sound generator and receiver allow a non-contact coupling. Due to non-occurring variations in the coupling the reflectors are reliably detected with this method which can be used especially with high temperatures and complicated geometries as well as narrow gaps. The reflectors can be directly analyzed on account of the provided control of the lasers. The aim of the project is to construct the prototype of ultrasonic testing equipment, based on lasers for testing reactor components, which is planned to be tested on-site.

## 2. Particular objectives

For generating sound by means of laser pulses lasers are required with a high pulse power and a high pulse repetition rate. Such lasers are technically realizable but expensive. In order to optimize the available lasers for this application the relations between the individual parameters with the generation of sound shall be examined thoroughly so as to be able to produce the sound pulses required for a special test assignment with the least possible expenditure in view of the lasers.

## 3. Research programme

3.1 Determination of the optimal light wave length for the method. Study of literature and experimental investigations of the light wave length dependency of:

3.1.1 Reflection and absorption of the material surface
3.1.2 Generation of plasma
3.1.3 Transparency of or losses in the optical components
3.1.4 Selection of the suitable laser according to 3.1.1 to 3.1.3
3.2 Determination of the laser output power required for the method
3.2.1 Investigation of the connection between the optical power density on the material surface and the generated amplitude of sound
3.2.2 Determination of the necessary sound radiating surface by using the directional characteristic required by the test assignment
3.2.3 Examination of surface damage and structural changes depending on the power density on the material surface.
3.2.4 Examination of the dependency of the time constant of the plasma on the power density on the material surface.
3.2.5 Selection of the suitable laser according to 3.2.1 to 3.2.4.
3.3.1 Examination of the influence of the rising flank of the light pulse on the shape of the generated sound pulse.
3.3.2 Examination of the influence of the light pulse width on the shape of the generated sound pulse.

## 4. Experimental facilities, computer codes

To 3.1 and 3.2:

For carrying out the experiments optical components were procured and optical benches ordered for the alignment of the laser. Reference blocks were produced for measuring the generated sound intensity. They consist of a broad-band piezoelectric transducer which is coupled as a sound receiver to a cylindrical delay path. The generated sound pulse is delayed by the delay path so that it is routed free from interferences to the measurement after the laser pulse and electric interferences have faded. The reference blocks were calibrated by means of beaming ultrasound of a known amplitude from a conventional piezoelectric transducer into the delay path.
After the specifications had been established a 100 MW-laser system based on Nd glass was ordered. Subassemblies of this system were delivered, however the last extension for 100 MW

pulse power has not yet been delivered.

5. Progress to date

To 3.1.1:

Literature has been ordered. Rough preliminary examinations were carried out with the radiation of a ruby laser ( $\lambda$ = 694 nm), an Nd-YAG-laser ($\lambda$ = 1060 nm) and a $CO_2$-TEA-laser ($\lambda$ = 10 $\mu$m). The reference blocks were thereby tested for regular functioning. Then the specific sound amplitudes obtained with the three wave lengths and related to the laser power were measured with improved measuring accuracy of the light power.

To 3.1.2:

The plasma generation of the different light wave lengths was observed in preliminary examinations. Up to now generating plasma was only possible by means of focusing the laser beam in a small focal spot, as the final extension of the laser system for 1060 nm to the highest power stage is still outstanding.

To 3.1.3:

For the transmission of optical components, such as mirrors, lenses, Pockels cells, electro-optical light deflectors as well as light conductor cables, the relative data sheets were ordered and partly evaluated.

To 3.2.1:

As the laser system is not yet completely extended to the highest power in the 100 MW range the specific sound amplitude was preliminarily measured using different powers in the range of small light powers. Higher power densities were achieved by different degrees of focusing. It was tried to convert the sound radiation of different focal spot diameters to a sound radiating area of uniform size so as to achieve standardization.

To 3.2.2:

A theoretical preliminary clarification was carried out.

To 3.2.3:
In a series of experiments with different degrees of focusing aluminium and steel specimens were irradiated with different light power densities and microscopically examined for surface damage.

To 3.2.4:
With the power presently still limited, a small volume plasma was generated by focusing and its time behaviour examined.

To 3.3.1 and 3.3.2:
Contrary to the original work scheme these points have already been started on as due to the missing final extension of the laser system to 100 MW power some of the examinations of points 3.1 and 3.2 could not be carried out. The shape of the light pulses was measured with a rapid photo detector and the shape of the generated sound pulses with a broad-band sound receiver. For the present the examinations were carried out on a single shape of light pulse.

## 6. Results

To 3.1.1:
The absorption conditions in the classical metal optics cannot simply be applied to the light power range which is of interest for generating ultrasound. The optical absorption conditions were therefore not persued further; instead the specific sound amplitude related to the laser power was directly determined and used for evaluation. The range below the limit of the plasma generation showed that, with the wave lengths 694 nm, 1060 nm and 10 µm, there are no substantial differences with regard to the specific sound amplitude.

To 3.1.2:
With high power light pulses, as required for ultrasonic tests, it is possible to generate the disruption of the air, i.e. an air plasma, by means of focusing. When the material surfaces were irradiated then much smaller power densities led to plasma generation than with focusing in air. The plasma generation depends on the surface quality of the materials. It was estab-

lished that the generation of a plasma affects the sound amplitude. The results obtained by means of these small spherical plasmas are not yet sufficiently informative as they are not simply applicable to a more extensive (order of magnitude $cm^2$) flat plasma. The generation of plasma starts at 10 $\mu m$ wave length with lower light power densities than at shorter wave lengths.

To 3.1.3:
The first evaluations showed that these components are much easier to be used for the wave length range in the vicinity of visual light. They are obtainable especially for 1060 nm and the power range of interest. The attenuation in fiber optics has to be examined yet in detail.

To 3.1.4:
On account of studying the relative literature and by comparing the pulsed light power lasers available in the market the types of laser suitable for generating ultrasound were selected thus confining the possible wave length range by practical facts.
Two of the laser types which were taken into consideration - $N_2$ laser, ruby laser, Nd-Yag or Nd glass laser and $CO_2$ laser - are out of the question, namely the ruby laser because of the limitation of the pulse repetition rate due to physical properties and the $N_2$ laser because the pulse duration and pulse power are too low.
$CO_2$ lasers produce the necessary power with high efficiency. High pulse sequences are possible due to increased cooling by means of gas circulation. The pulses are probably too wide for generating ultrasound for material testing. The temporary decision is therefore in favour of Nd-glass or Nd-YAG lasers. This type of laser has a suitable pulse width and its wave length is within a range where the radiation can be easily focused and transmitted by light conductors. The necessary light power and pulse repetition rate can be realized with this type of laser according to the manufacturer's specifications.

To 3.2.1:
It was detected that with low light powers the increase of the sound amplitude was linear with the light power. With high light powers the sound amplitude first increases in a super-proportional manner and finally reaches saturation. Presently with this result of high powers there is still considerable uncertainty because the sound radiating area was different in size due to the differing degree of focusing and considerable corrections were necessary to the measuring results for comparing.

To 3.2.2:
The generated sound pulses have a shock wave characteristic. From the known acoustic conditions when testing with shock waves it follows that, for the generation of the desired directional characteristic, the suitable sound radiating area is appr. 80 $mm^2$.

To 3.2.3:
As the final extension of the laser system is still outstanding the results had to be exterpolated, for the time being, from focusing series. It was established that power densities of up to appr. 1 $MW/mm^2$ did not yet cause any damage to the surface. Beyond this limit slight material abrasions occurred with different power densities depending on the material and the surface quality.

To 3.2.4:
Time constants were found in the range of appr. 100 ns. This result obtained with small spherical plasmas is not easily applicable to the future conditions of larger areal plasmas because it is possible that the cooling down of the plasma proceeds differently.

To 3.3.1 and 3.3.2:
With the given pulse shape - a symmetric Gauss pulse - the shape of the sound pulse corresponds with the shape of the light pulse. The result does only apply for the light power range below the limit of the plasma generation.

## 7. Next steps

To 3.1.1 and 3.1.2:
The results shall be examined in the range of higher power densities (appr. 100 MW/cm$^2$) thereby including the plasma generation.

To 3.1.3:
The attenuation and the maximum load with fiber optics shall be examined in detail.

To 3.2.1, 3.2.3 and 3.2.4:
The examinations shall be completed by experiments and measurements in the power range above the limit of the plasma generation.

To 3.3.1 and 3.3.2:
The measurements shall be extended to the range of high light powers thereby including the plasma generation. In addition it is planned to tackle the following points:

3.3 Determination of the laser pulse shape most suitable for the method.
3.4 Determination of the highest pulse repetition rate that is possible with this method.
3.5 Determination of a suitable method of generating the directional characteristic of the sound beam required by the test assignment.

## 8. Relation with other projects

–

## 9. References

–

## 10. Degree of availability of the reports.

–

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 12.3 | RS 2703 |
| Vorhaben/Project Title<br>Weiterentwicklung zerstörungsfreier Prüfverfahren für wiederkehrende Prüfungen in Reaktoranlagen<br><br>Onward Development of Nondestructive Testing Methods for In-Service Inspection of Reactor Power Plants | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>BAM, Berlin<br>Krautkrämer, Köln<br><br>KWU, Erlangen<br>M.A.N., Nürnberg |
| Arbeitsbeginn/Initiated<br>1.2.1976 | Arbeitsende/Completed<br>30.9.1978 | Leiter des Vorhabens/Project Leader<br>Dipl.-Ing. J. Lindner/MAN |
| Stand der Arbeiten/Status<br>Continuing | Berichtsdatum/Last Updating<br>December 1977 | Bewilligte Mittel/Funds<br>4.685.000,-- |

## 1. General Aim

Basing on the work carried out under the Federal Research Programme RS 27, RS 2701 and RS 2702 the target of the present project is to achieve further progress in the development of inspection techniques and equipment for in-service inspection of pressure vessels and, on completion of the work described herein, to establish a means of remote-operated volumetric inspection of the entire reactor pressure vessel according to present-day technological standards. This development project relates to ultrasonic inspection techniques only. Volumetric inspection of the entire pressure vessel is a target which calls for all participant firms to make further progress in the development in inspection systems and inspection techniques, manipulators and electronic equipment. Participants in this programme are the Federal Institute for Materials Testing and the firms of Kraftwerk Union AG, Krautkrämer GmbH and Maschinenfabrik Augsburg-Nürnberg AG (M.A.N.).

## 2. Particular Objectives

The programme is sub-divided into the following areas:

1. Basic studies of ultrasonic inspection methods and optimization of detection and analysing techniques.

2. Conceptual design of inspection systems to improve the probability of fault detection in different inspection areas.

3. Development, fabrication and trials of internal and external inspection manipulators with a view to achieving the goal of complete volumetric inspection of the reactor pressure vessel by automated equipment.

4. Onward development of components and methods to improve data logging, data processing and display/printout systems.

## 3. Research Programme

The research programme embraces the following work:

### 3.1 General

The general aim of the research programme is to make further progress in the development of ultrasonic inspection techniques with a view to increasing the probability of fault detection and improving the interpretation of the inspection results. This includes research into techniques for the analysis of indications, e.g., focus technology or line holography, and preparations for practical application. The tasks call for extensive theoretical and experimental research.

### 3.2 Inspection Systems

Special inspection systems must be developed for the inspection of close-to-surface zones and of lugs and supports welded to the inside and outside of the reactor pressure vessel. The task involves testing a method of monitoring the stability of the probes during the course of inspection work. The task also includes the development of a probe with adjustable sound beam parameters for special inspection applications, e.g., for inspecting the ligaments in areas of penetration holes for control rods.

### 3.3 Manipulators

New manipulators are being developed and current manipulators are being improved for remote-controlled inspection of the following areas of reactor pressure vessels:

- Ligaments in areas of control rod penetration holes in the bottom closure and closure head
- External inspection of transition weld between cylindrical area of vessel and spherical bottom (BWR)
- Inspection of inner radii of small nozzles
- External inspection of flange at bottom of vessel and at closure head
- Extended external inspection of cylindrical area of vessel with automatic transfer of inspection car from one longitudinal rail to another. Motions under computer control.

3.4 Data Logging

For processing the data of automatic ultrasonic inspection the capacity of data logging equipment must be increased to accommodate the increased flow of data. Existing programmes are being increased to improve software and the interpretation quality of display and printouts.

## 4. Experimental Facilities, Computer Codes

The research programme envisages laboratory tests to establish the capability of inspection equipment for individual inspection techniques as well as the provision of working models of manipulator components. Various computer codes are used to establish the best parameters of inspection systems. The task includes the provision of further computer codes for evaluation and display/printouts of the data obtained during inspection.

## 5. Progress to Date

During the period under review, three-dimensional computer programmes were completed for echo dynamics for tandem and pulse echo methods with longitudinal waves taking into consideration possible inclinations of the reflector in two dimensions. The predicted values were compared with relevant observed results.

The influence of curved surfaces on the sound field was investigated and DGS charts determined for sensitivity settings in order to develop focussing technology.

Tests were made on a large vessel wall specimen to establish an appraisal algorithm for the flaw size of indications found during recurrent ultrasonic inspections.

A probe with adjustable sound beam parameters has been designed. Tests were carried out with the complex probe carrier in order to establish the variations of frequency spectrum and beam angle variations.

The experimental investigations aimed at the development of automatic stability control of ultrasonic probes have been completed.

Design work was continued on the following manipulators:

- Universal manipulator for outer surface inspection of the cylindrical parts of the reactor pressure vessel and the nozzles using an automatic transfer of the manipulator from one longitudinal track to the next
- Manipulator for extended ligament inspection on the bottom closure of the boiling water reactor
- Manipulator to inspect the transition weld between cylindrical vessel surface and spherical bottom from the outside
- Manipulator for inspecting the flange of the top head dome and the lower part of the pressure vessel.

Engineering solutions for an analysis manipulator and a suitable manipulator for the internal inspection of nozzle welds and nozzle radii on small nozzles were investigated.

With a view to improving the evaluation of tandem probe inspection, statistical studies were made of the transfer measurements on full size components, with allowance being made for the length and depth of the no-signal gaps.

To improve data presentation, theoretical studies were made on reflector locating based on pulse travel time.

A hardware concept was developed for the processing of 32 ultrasonic inspection functions and the implementation of a variable-volume integration.

## 6. Results

A comparison of observed and predicted values to establish the influence of the inclined position of flat bottom hole reflectors showed good agreement for X- and Y-echo dynamics. It was found that the theoretical model on the strength of the good agreement with experimental information is suitable to serve as a basis for inspection system planning and optimization.

$0.1^{o}$ increments of angle variations for beam and squint angles of the probe with variable sound parameters in order to be able to use this probe also as an analysis probe.

Suitable variables for the automatic stability checking of probes are provided by ultrasonic indications from the probe measured by means of the test crystal and/or additional control crystals.

For all manipulators referred to there are partly conceptual designs and detailed designs available.

The study of reflector locating by means of 3 probes has shown that even small tolerances in pulse range measurement are liable to cause unacceptable errors in locating the reflector.

## 7. Next Steps

Work on the current project is expected to be completed in September 1978. In the case of some points - especially the extended ligament inspection and the computer-assisted external surface inspection of the cylindrical reactor pressure vessel area, it will be necessary to continue the work beyond that date. In the case of bottom closure inspection, it will be necessary to carry out investigations into inspection technique and manipulator developments whereas in the case of the external surface inspection of the cylindrical reactor pressure vessel area further development will concern mainly the manipulators.

## 8. Relation with Other Projects

Reference is made to what has been said in Report A 74 with regard to RS 27o2.

## 9. References

During the period under review, the following summarizing reports were prepared in the German language:

| Author | Title | Date |
|---|---|---|
| BAM | Focussing probes | April 1977 |
| BAM | Possibilities and limits of flaw classifications and flaw size appraisal by evaluation of echo dynamics | April 1977 |
| BAM | The relative echo amplitude of obliquely positioned crack type flaws | June 1977 |

## 1o. Degree of Availability of the Reports

A relatively small number of reports have been issued. A limited number are still obtainable from the author or from GRS.

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 10.1.77 - 31.12.77 | 12.3 | RS 247 |

| Vorhaben/Project Title | |
|---|---|
| Non-destructive testing of three HSST-plates<br><br>Zerstörungsfreie Prüfung an drei HSST-Platten | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Fraunhofer-Gesellsch., München<br><br>IzfP, Saarbrücken |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 10.1.1977 | 31.12.1977 | Dr. G. Deuster |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| completed | December 1977 | 1.248.100 DM |

## 1. General Aim

It is to demonstrate for 3 plates out of the american HSST-program how the NDT-results according to a certain testing procedure can agree or disagree, if every participating country does use all instruments available and trained people.

## 2. Particular Objektives

2.1. Non-destructive analysis with standard methods of detected fault areas.

2.2. Description of faults concerning their shape, orientation, kind and structure.

2.3. Possibilities of detection and description of faults, application limits of the used testing methods.

## 3. Research Program

There were applicated all standard methods at the time available for the 3 HSST-plates as used during fabrication, basic and in-service inspection.
As for as these methods are automated a comparison was also made to hand testing. After detection of fault areas analytic methods were used for interpretation.
The following methods were applicated:

For inside faults:

3.1. Ultrasonic impulse-echo, focus
3.2. Ultrasonic-tandem
3.3. Evaluation, interpretation, reconstruction by special methods
3.4. US-holography
(optical reconstruction)
(numeric reconstruction)
3.5. Ultrasonic-high-frequency scattering and absorption
3.6. Radiography

For surface - faults:

3.7. Magnetic particle test
3.8. Multi-frequency-eddy current
3.9. Potential- and magnetic leakage flux-technique
3.10. Ultrasonic surface waves

## 4. Experimental Facilities

Manipulator, data collecting and evaluation systems from reactor-safety-programs RS 1o2-16, 1o2-17, 1o2-18, 1o2-2o, RS 27
NDT-standard equipment,
radiographic instruments

## 5. Progress to Date

For the 3 test pieces (see appendix) a total program was set up together with all different institutions.
At first there was made by RWTÜV and IABG a impulse-echo examination according to the prescribed test-procedure, also the plates were proved corresponding German guide lines.
For certain areas the impulse echo testing results were collected automatically by the IABG. In order to comprehend the surface faults RWTÜV made a magnetic particle test.
The IzfP performed US-impulse-echo, tandem and -pitch and catch measurements, also investigations with focus probes. All three

test specimens were investigated with acoustic holography and US-high-frequency scattering in order to determine grain size and inhomogenities of structure.
For surface inspection there were used eddy current-, potential-, magnetic leakage flux- and surface waves-techniques.
Also radiographic measurements were made by MPA-Stuttgart.
The BAM-Berlin performed also US-impulse-echo, Tandem and focus techniques, in fault-areas there was used the line-holography.

KWU-Erlangen used volumetric automatic inservice inspection methods an two plates especially in the area of welding seam, in addition to that a lot of hand tests were made.
From each institution reports were written.
The IzfP has made in the meantime a first evaluation (see VJ-Bericht 2. Quartal 1977).

## 6. Results

6.1. For the inside fault testing (see point 3.1. - 3.6. as an example the Testspecimen, 50-51) the results are shwon from the single institutions (fig. 4, 5, 6, 7). They show a lot of individual faults and a continuous fault area in the welding seam and the heat-affected-zone. For the discontinuities an exact coincidence is not always to discover.
The US-scattering techniques serves for detection of material-anomalies and for grain-size-determination in the regions of welding seam, heat-affected-zone and base material. The test-specimen-structures were of a very different quality, whereby the inhomogenities were found preferably in the base material especially in the centre of wall thickness.

The grain-size determination provided average values between ASTM 6 and 7.

6.2. Concerning the surface-testing (see point 3.7.-3.10.) a good conformity was noticed. For the cracks which were found by RWTÜV the IzfP executed crack-depth-measurement with the potential- and magnetic leakage flux-technique, whereby a maximum depth of 4 mm was detected.

## 7. Next Steps

The IzfP will write a HSST-final-report in the near future, whereby the single NDT-techniques are compared each other and their applicability with regard to fault-detection and interpretation is proved. Out of these results a recommendation is given for the destructive-testing in the fault areas. Then evaluation of accuracy, reliability and applicability of the single techniques is possible.

## 8. Relation with other Projects

For detection and interpretation NDT-methods of fault areas see RS-reports.
RS 1o2-16, 102-17, 102-18, 102-20, RS 27

## 9. References

See RS-reports mentioned under point 4 and 8

## 10. Degree of Availibility of the reports

GRS or contractors of the RS-projects.

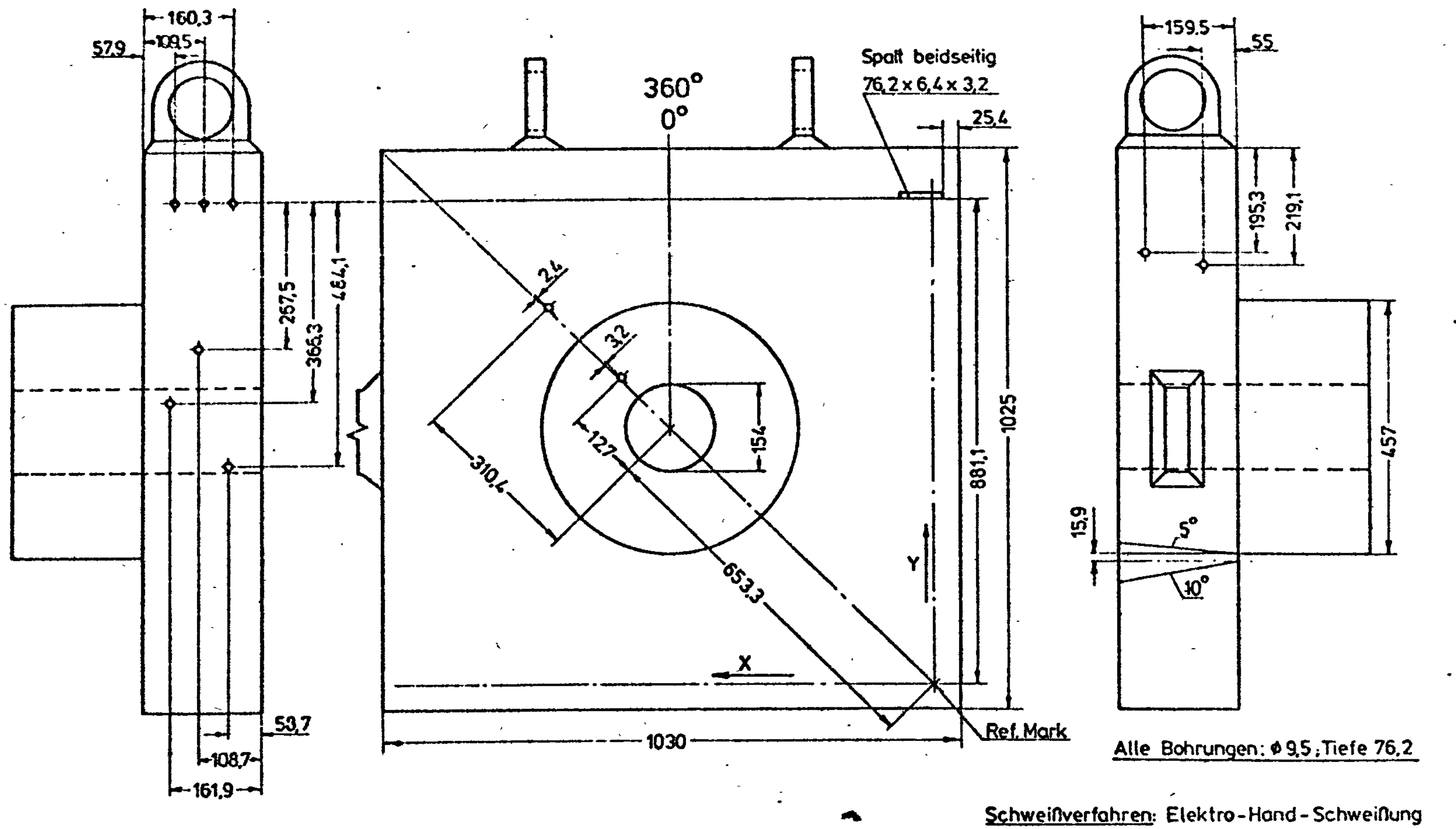


Fig. 1: Testspecimen 204

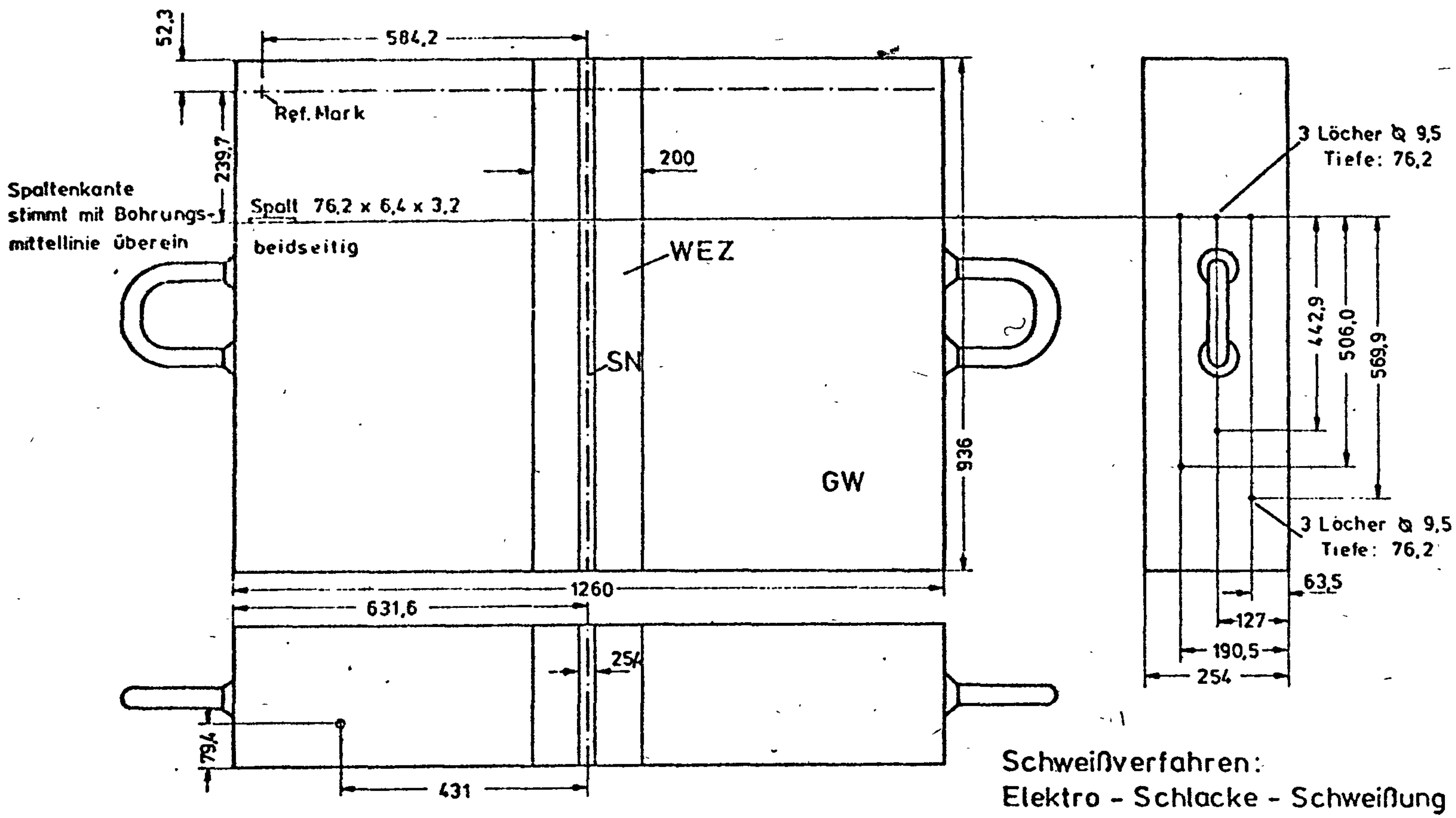


Fig. 2: Testspecimen 50-52

1324

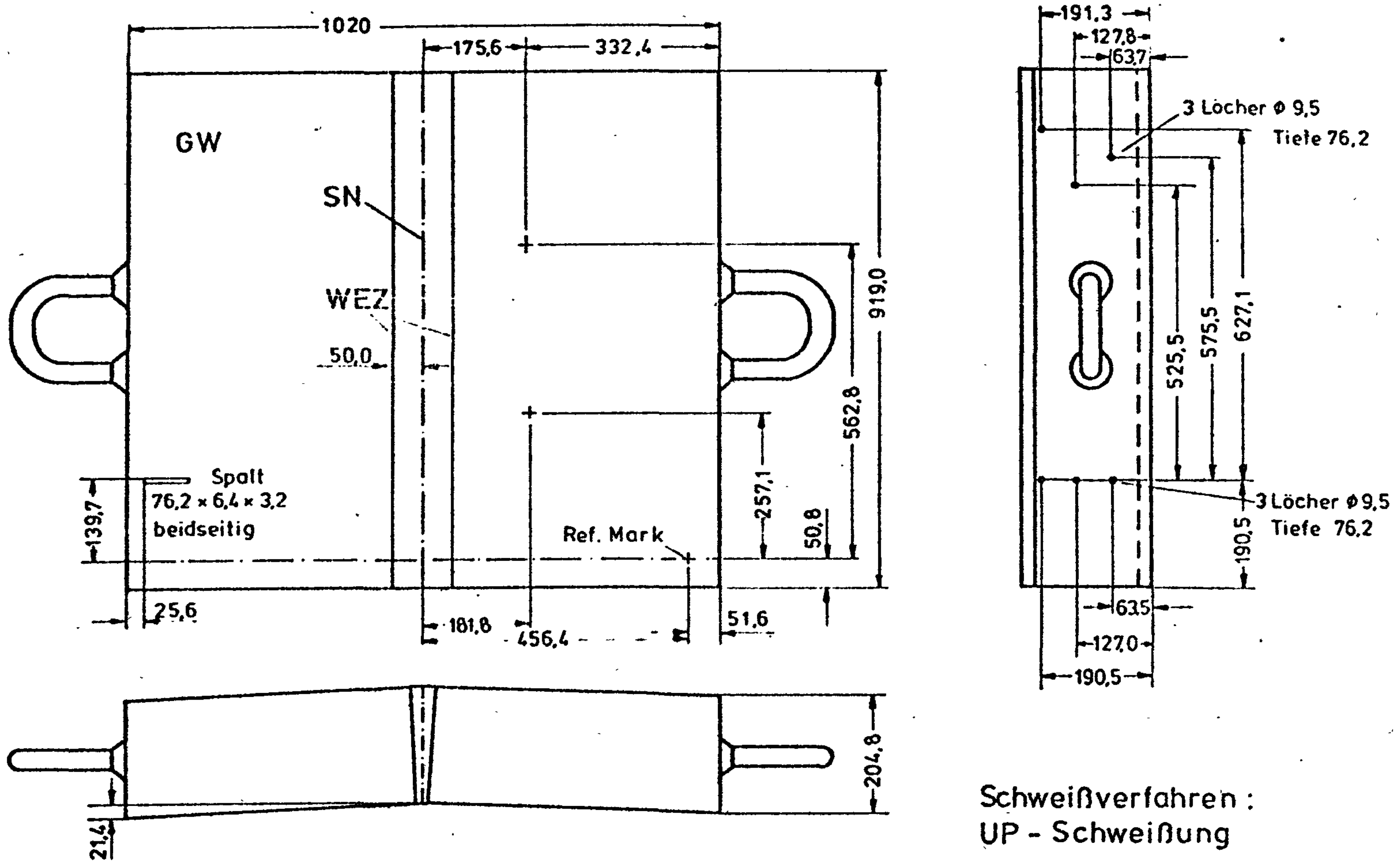


Fig. 3: Testspecimen 51-53

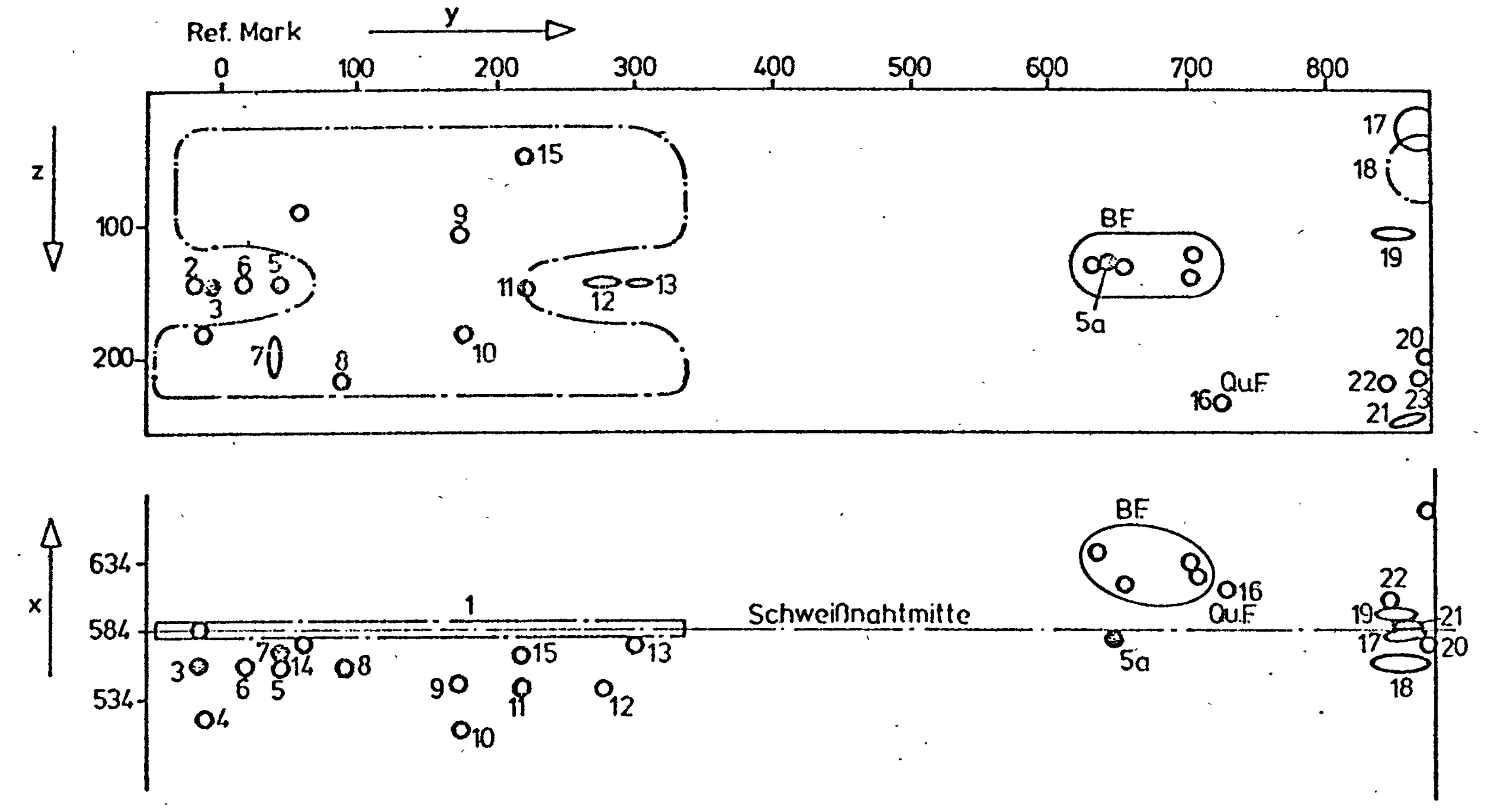


Fig. 4: Fault-Position-Plan of the Testspecimen 50-52
Excerpt BAM-Report

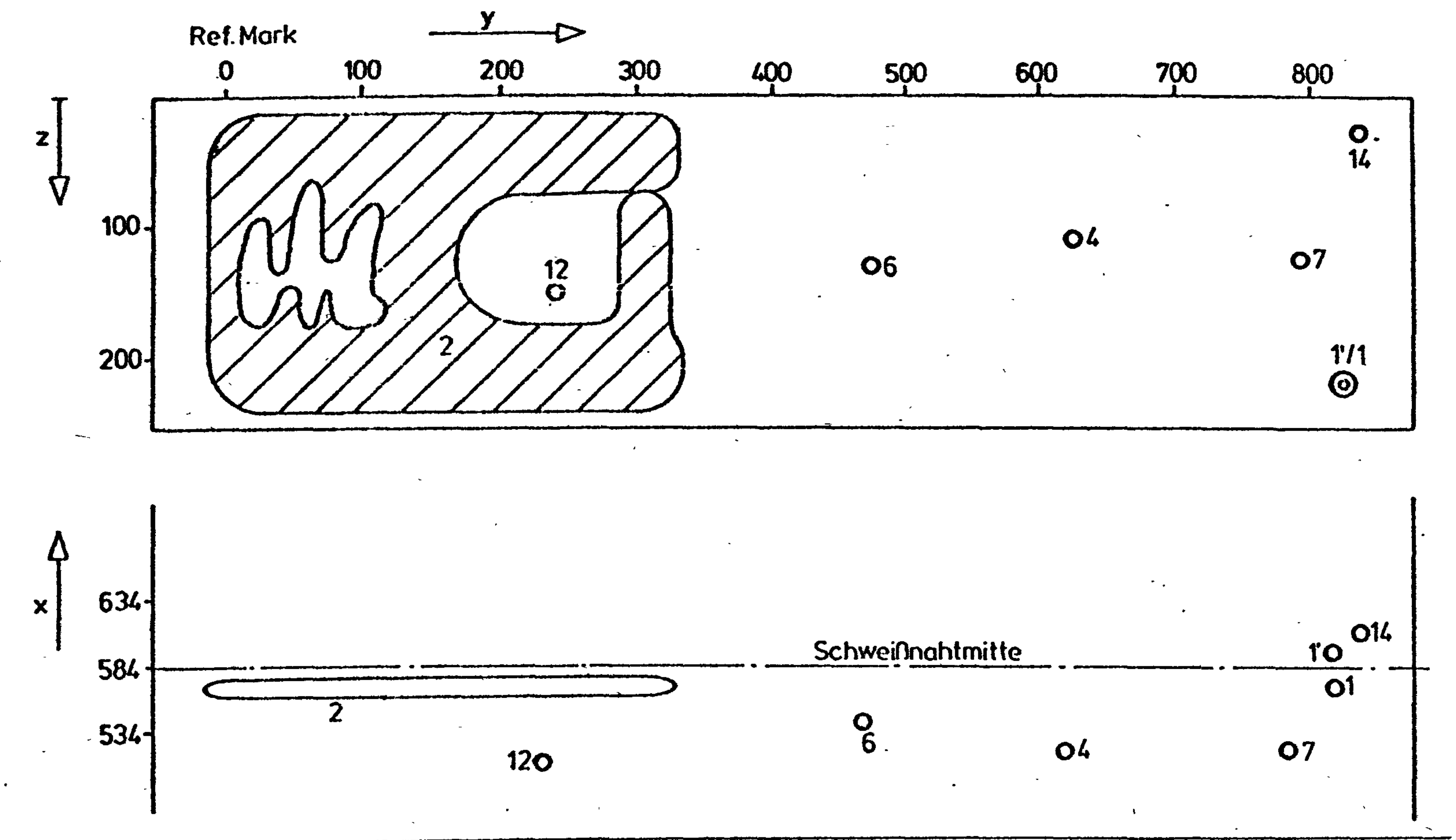


Fig. 5: Fault-Position-Plan of the Testspecimen 50-52
Excerpt KWU-Report

1327

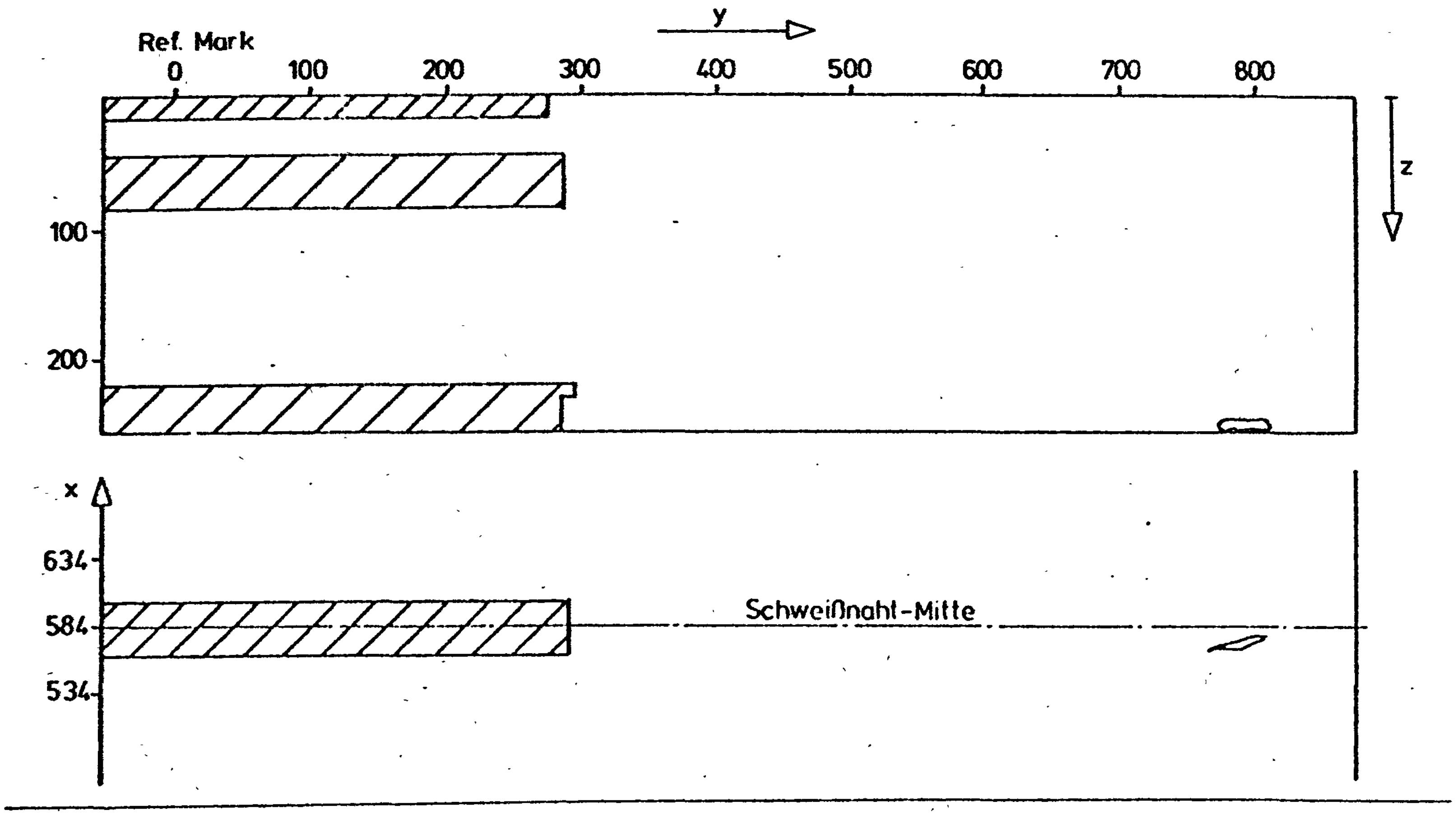


Fig. 6: Fault-Position-Plan of the Testspecimen 50-52
Excerpt RWTÜV-Report

1328

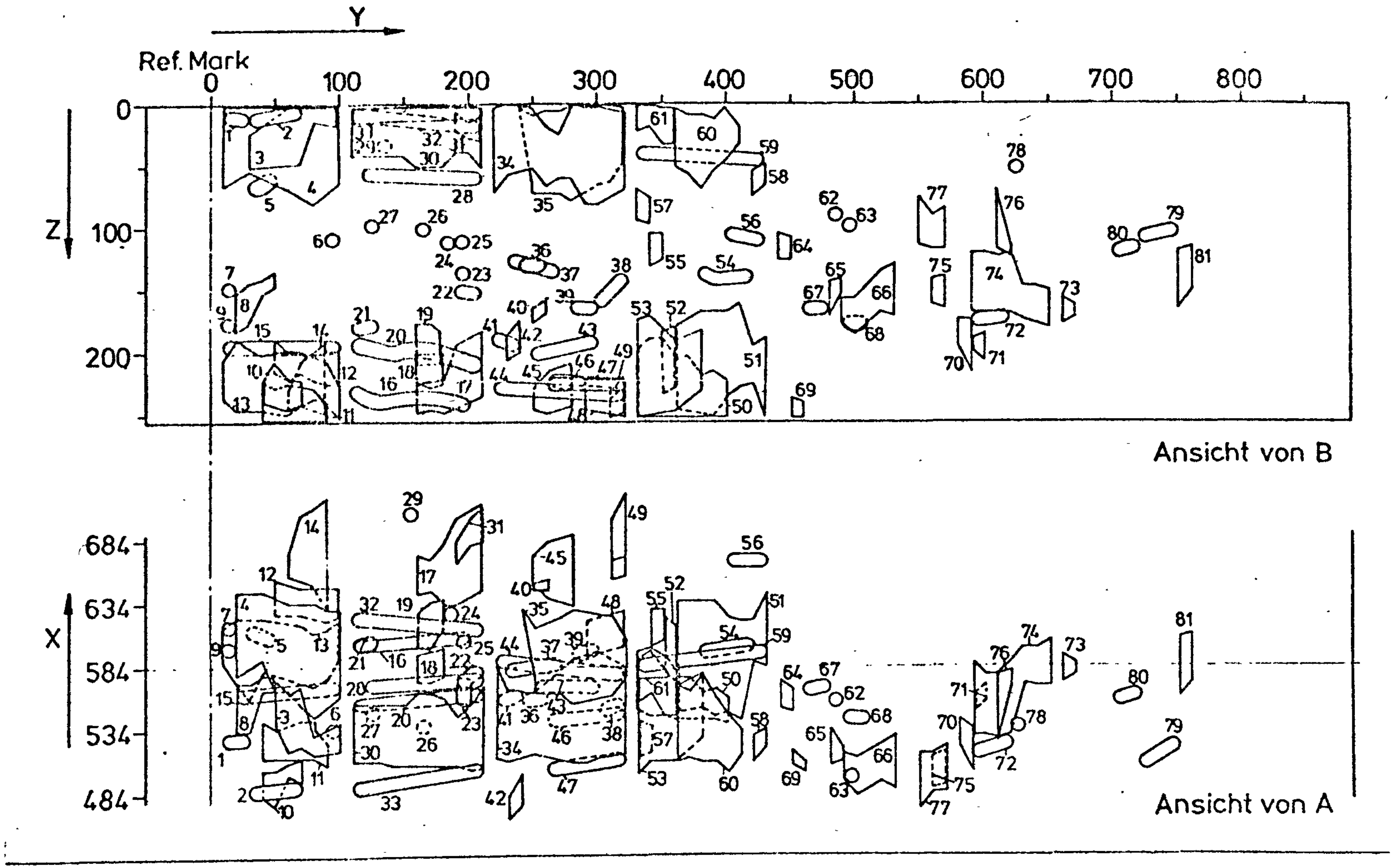


Fig. 7: Fault-Position-Plan of the Testspecimen 50-52
Excerpt IzfP-Report

1329

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 12.3 | RS 249 |
| Vorhaben/Project Title<br>Entwicklung eines elektronisch fokussier- und schwenkbaren Real-Time-Abbildungssystems für die Ultraschall-Werkstoffprüfung<br><br>Development of an electronically focussed and steered real time imaging system for non-destructive testing | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Fraunhofer-Ges., München<br><br>IzfP, Saarbrücken |
| Arbeitsbeginn/Initiated<br>1.1.1977 | Arbeitsende/Completed<br>31.12.1977 | Leiter des Vorhabens/Project Leader<br>Dr. W. Gebhardt |
| Stand der Arbeiten/Status<br>Continuing | Berichtsdatum/Last Updating<br>Dec. 1977 | Bewilligte Mittel/Funds<br>377.000,-- DM |

## 1. General Aim

The aim of project RS 249 is to explore the applicability of phased array techniques in non-destructive testing. Based upon these investigations a real-time imaging system for basic and repetitive inspections of reactor pressure vessels is to be realized.

## 2. Particular Objectives

While the field characteristics of "conventional" ultrasonic transducers are determined by the geometry of a single transducer, in the case of a phased array a more or less number of transducer elements contribute to the field structure. By variation of the amplitude and phase steering of the transducer elements the ultrasonic field can be influenced definitely: normal and angle probes as well as focussed probes can be substituted by a single array probe.
From the application of array techniques in non-destructive testing one expects the construction of an inspection system that is applicable both for flaw detection and for flaw analysis. The fast flaw detection could be done with wide opened beam in real-time B-scan, whereas the flaw analysis can be done with fixed and focused beam by A-scan.

## 3. Research Program

3.1. Market investigation.

3.2. Literature research
3.3. Physical basics
3.4. Measurement and computation of field characteristics of phased arrays
3.5. Conception and construction of an electronically steered and focussed pulse-echo system.

## 4. Experimental Facilities, Computer Codes

To explore the ultrasonic fields of array-probes an electronically steered and focussed transmitter system has been built. The phase steering was realized by means of a programmable word generator. The power amplifiers were developed and built. An 8 element array and a 16 element linear array served as probes. The measurements of the sound pressure distribution were performed at stepped and at cylindrical specimens with electro-magneto-acoustic and piezoelektric transducers.
A medical phased array was ordered and is installed in the meantime. An electronically steered and focussed pulse-echo system is completed so far that we can begin to experiment with various probes in the near future.

## 5. Progress to Date

The literature research and the market investigation are completed. Field characteristics of phased arrays were calculated theoretically. Thereby the following two cases were distinguished: array in a fluid and array on a solid.
The field characteristics of several array probes were measured and compared with theory.
An electronically steered and focussed transmitter system was built.
An electronically steered and focussed pulse-echo system is in construction. Three possibilities for signal delay were investigated: CCD's (charge coupled devices), tapped delay lines and voltage controlled phase shifters. The investigations brought a decision in favour of the CCD's (large bandwidth, large delay range). CCD's are digitally controlled analog shift

registers. The clock generation is made by 32 MHz frequency synthesizers. 25 W power amplifiers are used as transmitters.

## 6. Results

The literature research has shown, that phased arrays are applied almost exclusively for the medical diagnosis. The following companies offer complete systems: Diagnostic Electronic Corp., Grumman Health Systems, Varian, Oldelft. The importance of the phased arrays for non-destructive testing has been recognized. Firms and institutes all over the world are working in the development of such devices for non-destructive testing, but until now no system is available commercially. The development of a hybrid phase controlled pulse-echo system by Battelle-Northwest for the EPRI, for investigations at reactor pressure vessels illustrates the interest in such systems. The probe of this system is a 120 element array. Bulk waves in 9 bearings and shear waves in 15 bearings shall be excited. The operation is planned to be switchable from pulse-echo to holographic mode.
The Electroscan, a medical prototype developed by SOMER, was tested by the IzfP. Thereby measurements at steel and PVC-specimens with 2 mm, 4 mm and 6 mm bore holes in depths of 30 and 60 mm were conducted. Though the array probe was not adapted to these materials, the received B-scans demonstrate that phased arrays could be employed already today in non-destructive testing.
If a transducer of width $\lambda/2$ is radiating in a solid ($\lambda$ is the bulk wavelength), the farfield characteristic is cos-like. As a consequence of the directivity pattern of such transducer elements the intensity of the main beam decreases with increased bearing. Inside the sector of $-30^{o}$ to $+30^{o}$ the directivity pattern for the cases "array in fluid" and "array on solid" are almost identical, since in this region the pattern of the elements for both cases have no appreciable difference. At greater bearings however, the level of the main beam is reduced drastically. At $55^{o}$ this reduction is e.g. ca. 10 dB.

The ultrasonic field of several array probes and their elements were measured. The measured and computed bearings were in good agreement. A sector of 45° with 25 bearings could be realized whereby the side lobe level was always better than 10 dB. In the nearfield (nearfield length ca. 75 mm) various focus depths were realized by a quadratic phase steering of the transducer elements. So the half main beam width was decreased by a factor 3 in a distance of 40 mm from the probe by focussing at this plane. Deviations of the side lobe level and the directivity patterns of the elements can be used as a measure of the mechanical coupling.

## 7. Next Steps

- Completion of the pulse-echo system,
- development of various arrays for shear- and bulk waves, which are specialized for different inspection modes,
- conception of a smaller, bearable electronically steered pulse-echo system with minor flexibility for hand inspection and for application as stationary probe.

## 8. Relations with other projects: none

## 9. References

A survey of about 126 references is given in a technical report (IzfP No. 770117). Most of the experiments cited above are discussed in the technical report IzfP No. 770237.

## 10. Degree of Availability of the Reports

By GRS, Glockengasse 2, 5000 Köln 1.

1335

| Berichtszeitraum/Period<br>1.1.77 - 31.12.77 | Klassifikation/Classification<br>12.3 | Kennzeichen/Project Number<br>RS 102-16/I |
|---|---|---|
| Vorhaben/Project Title<br>Non-destructive structure evaluation by means of scattered ultrasound<br><br>Zerstörungsfreie Prüfung des Gefügezustandes mittels Ultraschallrückstreuung | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Frauenhofer-Ges., München<br>IzfP, Saarbrücken |
| Arbeitsbeginn/Initiated<br>1.5.1973 | Arbeitsende/Completed<br>31.12.1977 | Leiter des Vorhabens/Project Leader<br>Dr. K. Goebbels |
| Stand der Arbeiten/Status<br>Completed | Berichtsdatum/Last Updating<br>December | Bewilligte Mittel/Funds<br>DM 894.000,-- |

## 1. General aim

Among other things the material constants of steels are structure-dependent. Therefore, the determination of the structure of steels is an essential task and has to cover the whole volume, especially for big components like reactor pressure vessels. Under this category the structure of welds is to regard, too, and the evaluation has to be done in a ndt way. NDT methods for these requirements are not yet technical standard, but in an advanced stage of development.

Another important point of interest is the detection of defects - both real and latent - as flaws in the presence of a high coherent noise level, for example austenitic steel welds, the accumulation of inclusions in certain material areas and the proof of residual stress.

## 2. Particular objective

Measurements of scattered ultrasound can be used for qualitative and quantitative determination of metal structures (grain size, impurity distribution). Compared to attenuation measurements, they present e.g. the following advantages:

- no need of plane-parallel, polished test specimens,
- evaluation of the structure inside the material,
- quantitative separation of absorption and scattering coefficient from attenuation coefficient,

- quantitative grain-size determination, independent of other methods.

The included impurities, generally differing in size from the grains, become detectable when measuring scattered ultrasound at appropriate frequencies.

Inner tensions cause changes of the elastic constants of the material, which lead to changes of the velocity of elastic waves. For that reason, velocity measurements are done using the pulse-echo overlap method.

The reduction of the coherent noise level, e.g. at the ultrasonic examination of coarse-grained materials and cladded reactor pressure vessels, is possible when using signal averaging methods. In this respect several averaging methods (variation of probe position, excitation frequency or beam angle) are under consideration.

3. Research program

In 1977, there were the following points of special interest:

3.1. measurements on austenitic materials and welds for detecting the microstructures and to improve the signal-to-noise ratio by using the several averaging methods.

3.2. Scattering and attenuation measurements at ferritic steels for detecting the structure of bulk material and weld with regard to the inclusions.

3.3. The laboratory apparatus for scattering measurements was modified to enable investigations in nuclear power plants, for example.

## 4. Experimental facilities, computer codes

For very accurate measurements of ultrasound velocities and attenuation an experimental device (MATEC) is used.
Scattering measurements are made with an apparatus which was built in the institute, marked especially by an analog-digital converter (100 MHz sample rate) and the on-line evaluation of the scattering measurement in a computer.
For the averaging measurements the ultrasonic pulse generating system differs from that one of the apparatus (see above); the signal processing system is the same.

## 5. Progr[illegible] date

The applicability of the different averaging techniques for the examination of coarse-grained materials was shown.
The results concerning scattering measurements on steels with inclusions are summarized in a report.
The project itself was completed.

## 6. Results

- Several austenitic weld specimens with welding defects were examined. When using the averaging methods more defects could be detected than with usual ultrasonic testing.

- Inclusions in the bulk materials in several forms (microcracks, impurities whether metallic or not) could be detected. At the moment the discrimination between the different types of inclusions is impossible.

- Ferritic welds have been examined. Among others there was a narrow gap welded specimen. It showed the best results in regard to attenuation.

## 7. Next steps

The project was completed.

8. Relations with other projects

Work concerning austenitic steels in RS 102-16/1 and RS 143, and the signal averaging methods in RS 102-16/1 and RS 273 are tuned together.

9. References

K. Goebbels, S. Kraus, W. Zimmermann: "Die Beurteilung inhomogener Gefüge durch Ultraschallstreuungsmessungen". IzfP-Bericht Nr. 770517-TW

K. Goebbels, S. Kraus, O.A. Barbian: "Ultraschallprüfung grobkörniger Werkstoffe. Verbesserung des Signal-Rausch-Abstandes beim Vorliegen kohärenten Untergrundes". IzfP-Bericht Nr. 770916-TW

K. Goebbels, S. Kraus, W. Zimmermann: "Abschlußbericht zu RS 1o2-16/1". IzfP-Bericht Nr. 781001-TW.

10. Degree of availability of the reports

GRS, Köln, Glockengasse

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| Sept. 1 - Dec. 31, 1977 | 12.3 | RS 273 |

| Vorhaben/Project Title | |
|---|---|
| Erhöhung der Auffindwahrscheinlichkeit von Fehlern durch Verbesserung des Signal-Rausch-Verhältnisses bei der Ultraschallprüfung. Teil II<br><br>Enhancing the Detection Probability of Flaws by Improving the Signal-to-Noise Ratio During Ultrasonic Testing. Part II | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Battelle-Institut E.V. Frankfurt am Main<br>Kraftwerk Union AG Erlangen |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| Sept. 1, 1977 | Dec. 31, 1977 | Dr. R. von Klot |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | Dec. 1977 | DM 441,994.00 |

## 1. General Aim

The investigations conducted under the above project are aimed at determining the maximum possible signal-to-noise ratio by means of the signal averaging technique during the ultrasonic testing of reactor pressure vessels for flaws in the vicinity of claddings, and at increasing the testing velocity.

The effect of the various parameters on the signal-to-noise ratio is to be systematically investigated.

## 2. Particular Objectives

Since the coherent coarse-grain noise is disturbing in the ultrasonic testing of reactor pressure vessels for flaws in the vicinity of claddings, it is to be reduced relative to the flaw echo. For this purpose the exponential signal averaging technique is used, where the probes advance at a constant speed.

The signal processing speed of the signal averager is to be increased such that the speed of advance of the probes can be raised to about 50 mm/s. This must, however, also permit real-time processing of the echo signals, which requires a high repetition frequency.

The maximum possible signal-to-noise ratio is to be determined for different probe arrangements. The effect of the following parameters on the signal-to-noise ratio is to be investigated:

- Frequency and angle of incidence of the ultrasonic waves emitted by the probes

- Speed of advance of the probes
- Signal preprocessing (rectifying, smoothing, taking the logarithm)
- Weighting factor of the signal averager

3. Research Program

3.1. Purchase and modification of the signal averager

3.2. Extension of the ultrasonic flaw tester

3.3. Laboratory experiments on a clad specimen with artificial flaws; variation of signal preprocessing

3.4. Experiments on a test piece taken from the wall of a pressure vessel at KWU, Erlangen; variation of probe arrangement, frequency and angle of incidence of ultrasonic waves

4. Experimental Facilities, Computer Codes

Ad 3.1. Purchase, modification and trial of the signal averager; selectable number of digital points per signal, thus increase in signal repetition frequency.

Ad 3.2. Extension of the ultrasonic flaw tester, installation of several analog outputs for the HF signal.

Ad 3.3. Supply of a clad test piece of 22 NiMoCr 3.7 (Fig. 1) on loan by IzfP/Saarbrücken. Dimensions 290 mm x 270 mm x 140 mm, thickness of the austenitic strip cladding 7 to 8 mm. Artificial flaws: three flat-bottom boreholes of 3 mm diameter and different depths (flaws Nos. 1, 2 and 3) and one cylindrical borehole of 2 mm diameter parallel to the cladding (flaw No. 4).

5. Progress to Date

Recording of the echo amplitude of the four artificial flaws in the clad test piece (e.g. Fig. 2) and of the scattered reflections from the coarse grains of the cladding as a function of the probe path. Use of the angle probe WB45N4 supplied by Krautkrämer (4-MHz transverse waves, angle of incidence of ultrasonic waves 44°). Direction of incidence of ultrasonic waves and direction of advance parallel to cladding strips, constant speed of advance. The thickness of the test zone selected is 22 mm, a value usual in practice (Fig. 1). This results in a maximum number of 145 signals per second, which can be processed by the signal averager after exponential weighting. The following parameters were varied:

- Speed of advance v = 2.5/5/10/20/50 mm/s
- Signal preprocessing: a) High-frequency signal, b) Rectified signal, c) Rectified signal on a logarithmic scale
- Weighting factor G = $2^0$ to $2^7$

Determination of the signal-to-noise ratio from the measured data.

## 6. Results

Ad 3.3. Fig. 3 shows several examples of the signal-to-noise ratio as a function of the parameters wighting factor, speed of advance v and signal preprocessing. The following trends can be derived from the results so far available:

- At low speeds of probe advance the signal-to-noise ratio as a function of the weighting factor in general has a maximum.
- At higher speeds of probe advance, the signal-to-noise ratio in general shows a monotonic decrease with increasing weighting factor.
- The improvement in the signal-to-noise ratio for the high-frequency signal (Fig. 3a) is slightly larger than in the other two cases of signal preprocessing (Figs. 3b and 3c).
- The improvement in the signal-to-noise ratio is about the same for all the four artificial flaws.

These preliminary results have to be verified by further experiments. It is to be expected that it will be possible to achieve a further improvement in the signal-to-noise ratio by optimizing frequency, angle of incidence and spread angle of the ultrasonic waves, and by reducing the thickness of the test zone.

## 7. Next Steps

Ad 3.3. Further laboratory experiments on a clad test piece with artificial flaws.

Ad 3.4. First experiments with the test piece taken from the wall of a pressure vessel.

## 8. Relation with Other Projects

RS 190 Enhancing the Detection Probability of Flaws by Improving the Signal-to-Noise Ratio During Ultrasonic Testing
Battelle-Institut e. V., Frankfurt; January 1976 to August 1976

RS 2703 Onward Development of Nondestructive Testing Methods for In-Service Inspection of Reactor Power Plants
BAM/Berlin, Krautkrämer/Köln, KWU/Erlangen, M.A.N./Nürnberg; February 1976 to December 1978

RS 244 Ultrasonic Testing Techniques for Pre- and In-Service Inspections for Fast Breeder Reactors
BAM/Berlin; January 1977 to December 1979

RS102-16/1 Non-Destructive Structure Evaluation by Means of Scattered Ultrasound
IzfP/Saarbrücken, May 1973 to December 1977

## 9. References

./.

## 10. Degree of Availability of the Reports

./.

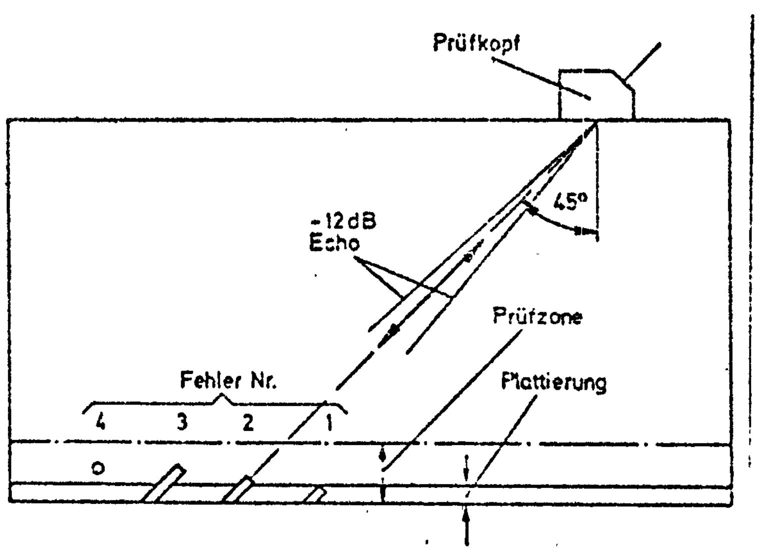


Fig. 1: Clad test piece with four artificial flaws; thickness of the cladding strip 7 to 8 mm; arrangement of the probe; thickness of the test zone 22 mm.

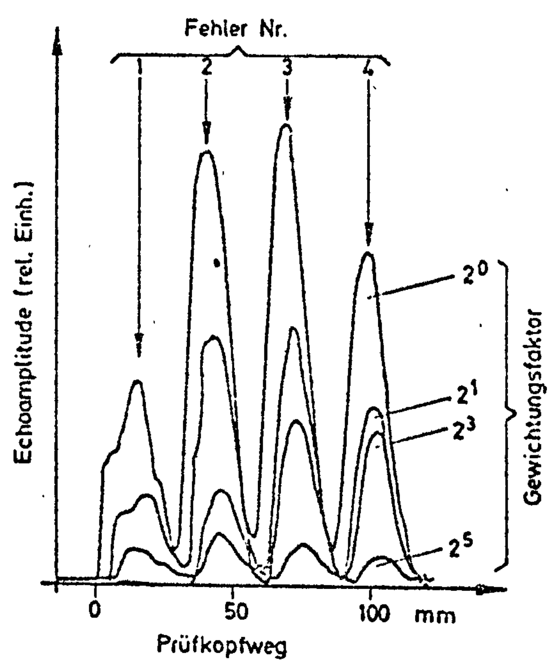


Fig. 2: Echo amplitude of the four artificial flaws as a function of the distance covered by the probe upon variation of the weighting factor. Speed of probe advance 20 mm/s; rectified signal.

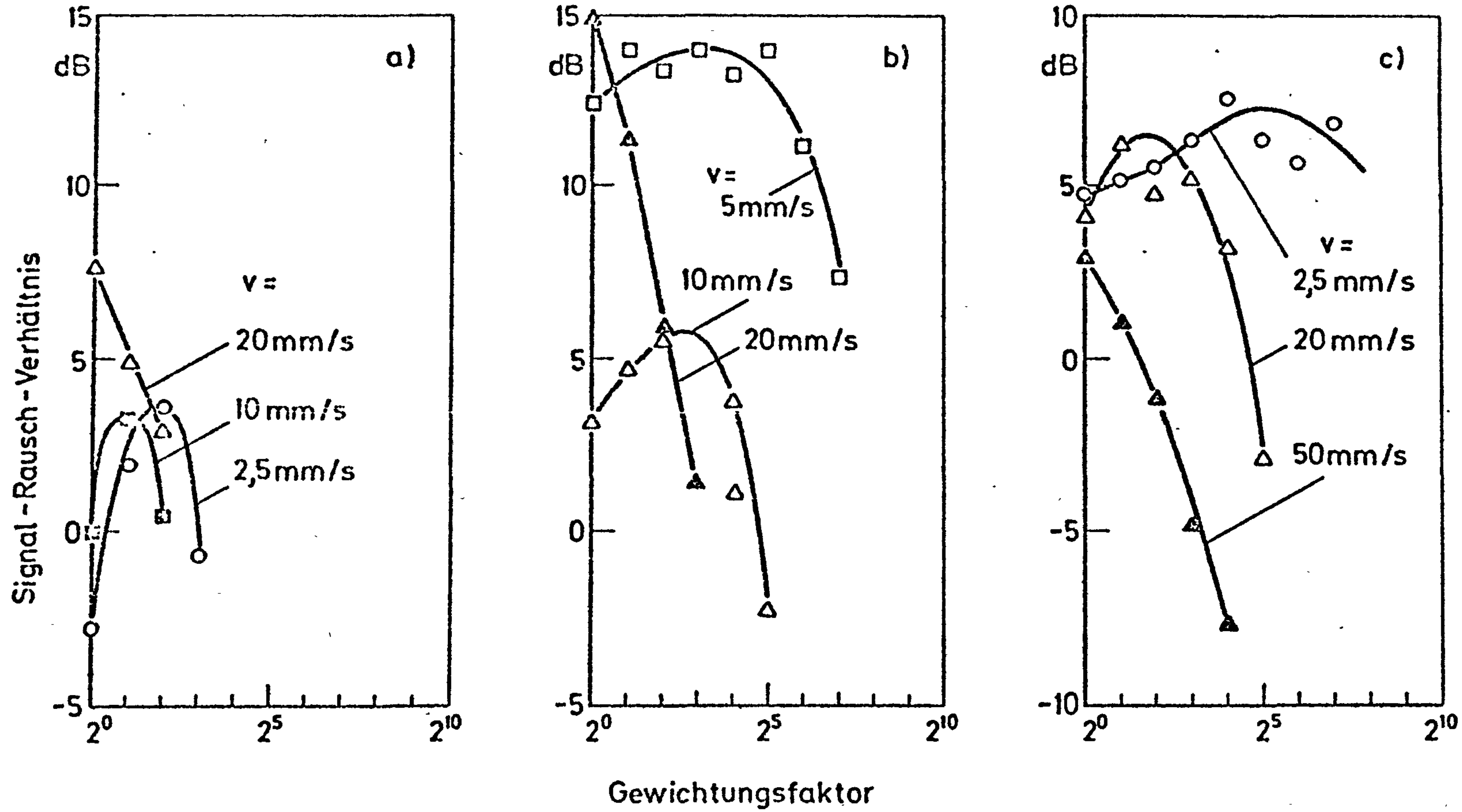


Fig. 3: Signal-to-noise ratio as a function of the weighting factor upon variation of the speed of probe advance v and the signal preprocessing

a) High-frequency tone burst, flaw No. 1

b) Rectified signal, flaw No. 3

c) Rectified signal on a logarithmic scale, flaw No. 3

3.2 Design of the data recording system

3.3 Investigations on fracture toughness specimens

3.4 Reduction of data and determination and optimisation of the required correlations between measured AE and fracture mechanical parameters

3.5 Investigation of acoustic wave propagation in thick-walled structural parts

## 4. Experimental Facilities, Computer Codes

Ad 3.2: To make frequency analysis of single signals up to about 800 kHz possible, a data recording system was been purchased, which comprises a two-channel transient recorder with display (paid from Battelle-Frankfurt funds), a rapid Fourier analyser, and a magnetic tape data store (paid from program funds) including a special interface.

Ad 3.3: A universal tensile testing machine (Instron) with hydromechanical control (25 Mp) is available for the static loading experiments with the single-edge notched (SEN) specimens made from the materials of the SBR. This machine has been adapted for AE measurements by special noise surpression measures.

For the experiments at temperatures up to 550°C an adequate furnace was purchased, special importance being attached to low-noise operation.

## 5. Progress to Date

Ad 3.1: Modified fracture toughness specimens (2", 3" and 4" CT specimens) were made from the ferritic steel 22NiMoCr3 7 (heat-treated in order to achieve 100-percent low-temperature bainite) and the austenitic steel X10CrNiNb18 9 in addition, tensile specimens were fabricated from broken CT specimens. Some SEN specimens were also made from the SBR materials, and six specimens of the X6CrNi18 11(1.4948) steel were heat-treated at 650°C for 1000 hours (at Interatom) in order to achieve an embrittled structure.

the crack is only a function of the depth of the crack while $U_o$ is the voltage in the undisturbed case. The theoretical formulation of the problem leads to a Neumann boundary value problem

1.3. Research Program

1.3.1. Numerical solution of two-dimensional electrical potential field problems, the specific optimation of the field sources and sinks as function of the geometry of the specimen.

1.3.2. Development of a three-dimensional Computer-Program.

1.4. Experimental Facilities, Computer Codes

1.4.1. Implementation of a finite-element FORTRAN IV program for numerical solution of Neumann-boundary value problems.

1.5. Progress Today

The Programm allows the solution for general two-dimensional geometries with arbitrary sources of the field.

1.6. Results

Computation of potential growth caused by crack-growth in a welded joint in flat specimen and in the nozzel of a reactor pressure vessel.

1.7. Next Steps

Publication of the results

1.8. Relation with other Projects: none

1.9. References: IzfP-Bericht 750102 TW
IzfP-Bericht 760214 TW

1.10. Degree of Availability of the Reports

Materialprüfung 18 (1976) 9, pp. 342-344

## 2. Magnetic Leakage Flux Method

### 2.1. Particular Objektive

For ferromagnetic materials which are flowed through by means of direct magnetic field producted by a yoke or by means of a magnetic field of a direct current, the magnetic leakage flux measured with Hall elements over and near a surface crack is a function of the permeability and of the geometry of the crack. The magnetography method allows documentation and signal processing of the leakage flux singals.

### 2.2. Research Program

2.2.1. The development of a magnetography signal measure and interpretation unit.

2.2.2. Theoretical examinations about the influence of surface crack parameters as crack-length, depth and width and of the permeability by solving numerically a Fredhom's integral equation.

### 2.3. Experimental Facilities, Computer Codes

2.3.1. Fundamental experiments with magnetography are made.

2.3.2. A FORTRAN IV program for iterative solution of integral equation is implemented.

### 2.4.Progress Today

Reconstruction of surface cracks form the measured magnetic leakage flux data.

### 2.5. Results

From the iterative solution of the Fredholm-integral equation it is possible to compute a calibration curve as function of crack depth and orientation.

A signal processing unit is building for evaluate magnetography signals.

## 2.6. Next Steps

Publication of the results

## 2.7. Relations With Other Projects: none

## 2.8. References:

Höller, P.:
Elektrische und magnetische Verfahren zur zerstörungsfreien Werkstoffprüfung.
In "Neuzeitliche Verfahren der Werkstoffprüfung", Verlag Stahleisen mbH, Düsseldorf 1973, Seite 139

## 3. Electromagneto-acoustic generation of free ultrasonic waves

### 3.1. Particular Objektive

Feasability-study of an ultrasonic angle probe for austenitic and ferritic steel on the basis of electromagneto -acoustic excitation.

### 3.2. Experimental Facilities and Computer Codes

The equipment has been in essential the same as in 1976 with the difference that a pulse magnetization of about the tenfold electric power was used.

### 3.3. Project status

3.3.1. Progress Today

The investigations concerning the principles of an ultrasonic angle probe were extended to non-ferritic specimens, in particular to aluminium and austenitic steel (DIN 1.4948). Measurements concerning the dynamic range, the detection of test flaws and the electronic variation of the angle of incidence were carried out. Besides a E.M.A. receiving coil with a one-sided directivity was tested. Finally some experiments concerning the perpendicular incidence of wide-band ultrasonic pulses were carried out.

### 3.3.2. Essential results

Due to optimization of the geometry of the transducer and the electric matching a dynamic range of 66 dB for ferritic specimens, of 60 dB for aluminium and of 55 dB for austenitic steel could be realized.
Using a magnetic field directed perpendicular to the surface of the specimen the angle of incidence could only be varied in the range from about 20 $^{o}$ to 40 $^{o}$. The generation of ultrasonic waves with an angle of incidence less than 20 $^{o}$ was not possible.
In austenitic steel a flat bottom hole with a diameter of 3 mm being situated in a distance of 90 mm to the transducer, oriented parallel to the wave-front, was detected at an incidence angle of 35 $^{o}$ with a signal to noise ratio of 30 dB.
The receiving transducer with one-sided directivity consisted of two windings displaced about $\frac{\lambda}{4}$ one from the other.
The phase of the receiving voltage of one of these windings was shifted about 90 $^{o}$ in relation to the phase of the receiving voltage of the second winding. After this phase-shifting the voltages were added. A forward-backward ratio of more than 20 dB was obtained.
The experiments concerning the vertical incidence of wide-band ultrasonic pulses generated by pancake coils (for the excitation of radially polarized transverse waves) and by rectangular coils (for the excitation of linearly polarized tranverse waves) had for result signal-to-noise-ratios of more than 30 dB for ferritic specimens although simply ultrasonic flaw detectors customary in the trade were used.

Dependent on the electric specification of the transducer the center frequency varied from 1,5 to 5 MHZ. The length of the ultrasonic pulse was less than 1,5 $\lambda$.

3.4. Next Steps

The project is finished.

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1. - 31.12.77 | 12.3 | RS 229 |

| Vorhaben/Project Title | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>BAM Berlin<br><br>Labor 6.22 |
|---|---|
| Studie über die Anwendung des Wirbelstromimpulsverfahrens zur Qualitätssicherung von Kernkraftwerkskomponenten<br><br>Study on the application of pulsed eddy current methods to quality control of nuclear power plant components | |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.10.1976 | 31.12.1977 | Dr. G. Wittig |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Completed | 31.12.1977 | 50.3oo,-- DM |

1. General Aim

Study of the possibilities of applications of the pulsed eddy current method to nondestructive testing of nuclear reactor components. The evaluation of the signals generated by the test system has to deliver informations about present defects with a high degree of statement power in a simple manner. The obtained results shall permit a critical evaluation if experimental investigations will result in successful solutions of detailed testing problems.

2. Particular Objectives

Ascertainment of the state of knowledge about the theoretical foundations and the foundations in the field of technical applications. Critical examination of the evaluation of test and investigation results from practical use of the method. Obtaining of statements about possibilities of application of the pulsed eddy current method for important test problems in the field of nuclear reactor components.

3. Research Program

3.1 Compiling of informations

3.2 Working of statements about possibilities of application of the method

3.3 Estimations of further developments

4. Experimental Facilities, Computer Codes

Programs for numerical calculations of time and local distributions of pulsed magnetic fields and eddy currents in flat test specimens.

5. Progress to Date

Ad 3.1 Evaluation of literature about applications and foundations of the pulsed eddy current method. Discussion of problems in non-destructive testing of nuclear reactor components in the frame of collaboration in the working-group LA RS 255 ff.

Ad 3.2 In the case of a plane exciting magnetic field with a pulsed shape on the surface of test specimen the following problems are numerical calculated:

1. Distribution of the magnetic field strength and the eddy current density within a conducting half-space.
2. The time-derived signal of transmitted field strength through a conducting sheet of finite thickness.
3. Reflected signals from a coated test specimen (model of an austenitic cladding on a ferritic base material).

Ad 3.3 Estimation of possible further development of the pulsed eddy current method.

6. Results

Ad 3.1 Applications of the pulsed eddy current method are found nearly exclusive in the field of testing of thin-walled fuel tubes - irradiated and nonirradiated. By special masking of the coil systems a higher defect resolution was possible compared with the frequency method. By sampling the signal at various points in time and application of a multichannel processing technique a statement about the defect position of depth could be obtained. Signal analysis and processing with digital methods are discussed but not yet state of engineering.

Ad 3.2 The estimations and experimental preliminary trials showed that there are no hints to fundamental difficulties for applications of the pulsed eddy current method to testobjects with low conductivities and defect depth ranges until about 15 mm. Possible applications are in the field of austenitic claddings, thick-walled tubes and other components. Concerning the influence of noise no experience is yet made.

Ad 3.3 For present and future testing problems in the constructional configuration of the coil systems are to find development-potentialities. By application of digital methods for signal processing and analysis an improvement in information yield about test results could be expected.

7. Next steps

The project is completed. Investigations for applications are provided in the frame of the project RS 299.

8. Relations with other Projects

Collaboration in the working group LA RS 255 ff for the projects RS 255, RS 256, RS 257 and RS 258.

9. References

10. Degree of Availability of the Reports

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 12.3 | RS 231 |

| Vorhaben/Project Title | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor |
|---|---|
| Mehrfrequenz-Wirbelstromprüfung<br>Phase 1: Aufbau eines Mehrfrequenz-Geräteprototyps<br><br>Multifrequency - eddy current testing<br>Phase 1: Construction of a multifrequency prototype | Fraunhofer-Gesellsch., München<br><br>IzfP, Saarbrücken |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.9.1976 | 31.7.1977 | Dipl.-Phys. R. Becker |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| completed | December 1977 | 91.700,-- DM |

## 1. General Aim

Development and improvement of a multifrequency eddy current test equipment (for details see last annual report).

## 2. Particular Objective

Construction of the electronic device (for details see last annual report).

## 3. Research Programme

Development of the electronic components and integration to the final system.

## 4. Progress to Date

Continuation of the investigations on the electronic circuits and their improvement, construction of coil drivers in the absolute and differential mode.

## 5. Results

The construction of the device is finished.

## 6. Next Steps

none

## 7. Relations with Other Projects

The prototype will be used for investigations on the suitability of the multifrequency method (RS 255, 256, 257, 258).

## 8. References

none.

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. 77 - 31. 12. 77 | 1 2.3 | RS 255 |

| Vorhaben/Project Title | |
|---|---|
| Mehrfrequenz-Wirbelstromprüfung<br><br>Multi-Frequency Eddy Current Testing | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik<br>R 214, Erlangen |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1. 5. 77 | 31. 3. 80 | B. Devrient |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | 31. 12. 77 | 961.100,-- DM |

1. General Aim

To design, construct and test a multi-frequency eddy current test unit (prototype) based on the currently available test results. This test unit is to meet the requirements of the automatic in-service inspection test of the pressure-containing reactor component walls (plated, unplated). For this reason it must produce an on-line record while being used in the pertinent control area. It is the aim of the venture for the unit to be as broadly applicable as possible; it is to be universally used for both LWR types, i.e. BWR and PWR as well as for LMFBR.

2. Particular Objectives

To extend the test findings for S.G. tube testing, in addition to the free tube area, especially for the rolled-in sections of the tube sheet, spacers and flow distribution sheet as well as the necking down on both sides of the tube bend.

Test of ferritic tubes used in the intermediate heat exchangers of Na-cooled reactors.

Comparison of results obtained from external tests of straight tubes and internal tests of U-tubes in order to find a relationship between manufacturing test and in-service test.

Test of the multi-frequency method as applied to surface testing in order to improve failure detection by better sub-surface resolution.

To design, construct and test a data acquisition system for eddy current testing for master data storage, on-line evaluation and test data recording.

3. Research Program

3.1 Coordination of developmental work

3.2 Tube test
- to erect a S.G. model
- to apply multi-frequency eddy current method to the model
- to improve current manipulator system
- to apply method to PWR S.G.
- system analysis

3.3 Surface testing
- prepare basic test data
- optimization of characteristic data for test scans
- applicability of current manipulating systems
- multi-frequency eddy current test method applied for HDR

3.4 Data acquisition

Acquisition, combining and recording of test results.

4. Test Facilities

Measurement equipment, provided by KWU, will be used. This operates on 4 frequencies from 1 kHz to 1 MHz, which will be simultaneously applied to the test coil.

The preparation of the pertinent algorithms as well as the required soft ware package (master data storage, on-line evaluation, recording) as well as specification, concept and technical arrangements of the data processing plant will be performed under the scope of this task.

5. Progress to Date

Available S.G. tubes were evaluated and a concept prepared for the S.G. model.
The following work was performed:

- supply of S.G. tubes to JFR for testing purposes
- " " " " " IzfP for the S.G. model
- recording of all spacer types currently used in nuclear power plants.

Evaluation of available test components as well as a preparation of a multi-frequency device specification. Basic research on eddy current testing of ferromagnetic materials was initiated.

Considerations of the hard- and software concepts and an interface for connection of each of two eddy current signals and for max. 6 channel couples have been made.

For further improvement of error detection a new connecting algorithm was tested on the basis of the cross correlation function.

A detailed specification of multi-frequency eddy current device was prepared.

The hardware-interface was defined with regard to transfer rate as well as positioning data and eddy current signal level.

For determination of the scanner rate, an analysis of test error signals was made referring to frequency, amplitude and phase position.

6. Results

The present status of development has not yet produced any results.

7. Next Steps

To: 3.1 Signing of the co-operation contract

To: 3.2 - Assembly of the S.G. model at IzfP
- Specification of the test task

To: 3.3 - Selection of suitable test- and adjusting devices for the measurement program
- Determination of test traces by means of instrument parameters
- Fabrication of test devices for hidden cracks
- Specification of test matrix.

To: 3.4 - Experimental checking of the theoretically obtained scanning frequencies
- optimization of the calculation algorithm with respect to the highest possible data transfer rate.
- Investigation of the possibilities of a prompt cross correlation as a means for providing a yes/no answer on error detection and the thus resulting data reduction.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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1361

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 12.3 | RS 257 |
| Vorhaben/Project Title: Anwendungstechnische Versuche zur automatischen Prüfung der Einbauten von Reaktorkomponenten und deren Wandungen mit Mehrfrequenzwirbelstromverfahren mit ON-LINE-Dokumentation<br><br>Investigations on the application of the multifrequency eddy current method on the automatic testing of installed reactor components and their walls with on line document. | | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Fraunhofer-Ges., München<br>IzfP, Saarbrücken |
| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
| 1.3.1977 | 29.2.1980 | Dipl.-Phys. R. Becker |
| Stand der Arbeiten/Status | Berichtsdatum/Last Updating | Bewilligte Mittel/Funds |
| Continuing | December 1977 | 1.071.500,-- DM |

## 1. General Aim

The aim of the project is the application of the results obtained from the preceeding projects (RS 231, RS 102-18) for the development and improvement of a multifrequency eddy current test equipment (prototype). This test device has to meet the requirements of the automatic recurrent testing of the installed components of reactors and their walls. For this purpose an on-line documentation as well as the employment in a control range is necessary. The application field of the test device is aspired to be as large as possible.

## 2. Particular objective

Realization of the test method, development of the needed software, adaptation of the test device to the different test situations, improvement of the test method at models and real reactor components, application of numerical methods to optimize the test frequencies and the construction of the test coils.

## 3. Research Programme

3.1. Evaluation of the optimal test frequencies.

3.2. Determination of the electric conductivity and the magnetic permeability of a material in an absolute manner.

3.3. Application of the multifrequency method to the testing of the RS 27 wall.

## 4. Experimental Facilities, Computer Codes

4.1. An existing computer programme has been modified to determine the coil impedance as a function of the frequency, the material characteristics and the coil dimensions.

4.2. A second programme computes the angle between the readout vector and the defect vector for all interesting frequency combinations /1/.

4.3. A third programme varies in the vector space the direction of the readout and produces the projections of the defect and disturbing vectors to the readout vector with the aim to optimize the ratio between defect indication peak and disturbing underground.

4.4. The multifrequency test equipment built up in the scope of RS 231.

## 5. Progress to Date

Relation to 3.1:
For a given set of frequencies the coil impedance is computed as a function of the relevant disturbing parameters. The angle between the readout vector and the defect vector is a measure for the suitability of a frequency combination. The optimal combination has an angle which is as small as possible.
But the results can even be more improved by means of a small rotation of the readout vector. Then the projection of the measuring vector on the new readout vector can yield a greater difference between defect indication peak and remaining disturbing underground.

Relation to 3.2:
The impedance of a pick up coil, which is in contact with the material to be tested, is measured at 50 Hz. For this reason the

measurement depends only on the permeability of the material; the influence of the conductivity can be neglected. Applying the computer programme described in 4.1 a calibration curve is produced, which gives the correlation between the permeability and the indication peak at given coil dimensions.
Knowing the permeability, now the conductivity is obtained by a measurement of the coil impedance at a higher frequency (≈ 10 kHz). This can be realized by means of a second calibration curve, which is computed by the same programme in 4.1 and gives the correlation between indication peak and conductivity at given frequency and coil dimensions.

Relation to 3.3:
The multifrequency prototype is outfitted for the applications of testing the RS 27-wall. The coils are constructed. The test system is optimized and the test is made by means of two frequencies.

## 6. Results

Relation to 3.1:
The determination of suited combinations of frequencies was possible for several pick up coils and austenitic specimens. Although the indication peaks of the disturbing parameters were not totally suppressed, the improvement of the signal-to-noise ratio could be realized in some cases.

Relation to 3.2:
The permeability and the conductivity could be measured in an absolute and independent manner. So the suppositions for the calibration of the test device are accomplished. The variations of permeability and conductivity in any combination can be suppressed.

Relation to 3.3:
After the application of the multifrequency algorithm a distbur-bing underground remained (except for the region of the horizon-tal weld), which is smaller than the indication peak of an

1 mm deep saw cut. In the weld sporadic signals were found with an indication peak equivalent to 1.5 mm deep saw cuts. The greater disturbing underground can be explained by variations of thickness of the cladding caused by the grinding of the surface, and can be reduced by the application of the 3-frequency algorithm.

## 7. Next Steps

- Expansion of the theory, especially to compute eddy current distributions in the presence of defects.
- Construction of a heat echanger model.
- Theoretical and experimental investigations on covered defects.

## 8. Relation with Other Projects

RS 231, 255, 256, 258.

## 9. References

/1/ R. Becker:
Zur Wirbelstromprüfung dünnwandiger Rohre nach einem Mehrfrequenzverfahren.
Materialprüfung 17 (1975) 7, S. 238-239

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| July 1,77 to Dec.31,77 | 12.3 | RS 258 |

| | |
|---|---|
| Vorhaben/Project Title Experiments regarding automatic inspection of the built-in parts of reactor components and their walls by means of multi-frequency eddy-current method with on-line documentation for repetitive inspection<br><br>Anwendungstechnische Versuche zur automatischen Prüfung der Einbauten von Reaktorkomponenten und deren Wandung mit Mehrfrequenz-Wirbelstromverfahren mit On-line Dokumentation für wiederkehrende Prüfung | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>INSTITUT DR. FÖRSTER<br>7410 Reutlingen<br><br>Development department Industrial Processing Research |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| March 1, 77 | February 29, 80 | Dr. W. Stumm |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | December 31, 1977 | 516.337,-- DM |

## 1. General Aim

It is aimed at the application of the knowledge gained in previous projects for design, construction and testing of an eddy-current test installation using the multi-frequency method, considering available test experience. This test installation shall meet the requirements of the repetitive inspection of the built-in parts of reactor components and their walls (plated and unplated). For this purpose is necessary an on-line documentation as well as an in-situ operation of the test installation. A maximum range of application of the test installation is aimed at; it ought to be universally applicable for light water reactors of the boiling water and pressurized water design as well as for sodium-cooled fast breeders.

## 2. Particular Objectives

Inspection of austenitic plated surfaces on cracks originating from the surface as well as from the joint face to the basic material. Application of the multi-frequency method for improvement of the signal-to-noise ratio by a - compared to the single-frequency method - better suppression of characteristic disturbing influences as distance variations of the probe and variations in ferrite content. Possibility to classify depths of defects by means of eddy-current signal.

Application of multi-frequency method for inspection of steam generator tubes over the whole tube range, especially at the rollings at the bottom of the steam generator tube, the support grids, the flow distributor plate as well as the whole tube bend.

3. Research Program

3.1 Electric adaption of the probe TL 10 according to /1/, /2/ to the two-frequency measuring device.

3.2 Foundations of the eddy-current harmonic method for defect inspection of ferromagnetic materials.

3.3 Linearization of the signals of eddy-current probes.

3.4 Determination of a fit measuring method to find out the probe space independent of the properties of the material under test.

3.5 Design of transmission between probe and test instrument

3.6 Optimization of pick-up probe setup

3.7 Discussion of theoretical and experimental findings regarding present day stage of tube inspection with KWU.

4. Experimental Facilities

re. 3.1 Eddy-current measuring device DEFECTOMAT S 2.801 in two-frequency version, 1 kc to 1 Mc

re.3.2 Measuring device for evaluation of harmonics, consisting of DEFECTOMAT S, various pick-up coils, ferromagnetic plates with saw cuts, constant field magnetizing yoke, filter amplifier, field strength measuring device, XY-recorder

re.3.3 Measuring device for linearization of probe signals, consisting of DEFECTOMAT S, phase shifter, power amplifier, pick-up coil TLK 1 and internal probe IK1

re.3.4 Capacitive distance measuring system from Disa

re. 3.6 DEFECTOMAT S, three-coordinate positioning device

5. Progress to Date

re.3.1 Computation of excitation and measuring windings. Mechanical completion of scanner. Chance control of consistency of test results according to /2/

re.3.2 Theoretical explanation of the origin of harmonics in ferromagnetic materials. Finding of a fit layout of equipment. Basic investigation and determination of the influence of specific parameters.

re.3.3 Design, computation and completion of the probes TLK 1 and IK 1. Determination of a fit test setup. Performance of measurements and records with variation of parameter, excitation current intensity and phase shiftings between both excitation currents.

re.3.4 Construction of a commercial quality capacitive distance measuring system. Hereby, a metal electrode at the probe face forms together with the reactor wall a condenser which represents the frequency-determining component of an oscillator.

re.3.5 Design of transmission between probe and test instrument. Start of the development of an amplifier integrated into the probe, investigations regarding cable splices, especially cable make-up, shielding, insulating materials, radiation resistance, evaluation of the influence of cables on the eddy-current signals.

re.3.6 Collection of literature on pick-up coil design. Layout and completion of various pick-up coil configurations. Survey of field of the field generating coils at various frequencies and at different materials. Measurement of eddy-current field disturbances caused by material defects, case in point the boring.

Installation of a laboratory measuring device for exact measurement of the influences of the specific parameters (defects, distance, ferrite content etc.) on the various pick-up coil assemblies.

re.3.4 Discussion of experience with Kraftwerk Union on present day stage of tube inspection by eddy-current methods, adaption of the probes used at present to the test device DEFECTOMAT S2.801.

## 6. Results

re.3.1 The results obtained under /2/ are reproducible by means of probe TL 10 with the DEFECTOMAT S.

re.3.2 According to /3/, under superposition of an alternating and a constant field, magnetization of a ferromagnetic test object shows hysteresis loops of the shape of a lancet. Therefore the eddy-current signal obtained by the tests consisted to about 10 % of the 3rd harmonic. The investigations proved the dependence of the harmonic signals on the various parameters of the test. Evaluation of the 3rd harmonic renders possible a suppression of the lift-off effect with absolute coils because with an appropriate constant-field premagnetization harmonics are generated only in the neighbourhood of the defect and not in flawless material. Up to now only tests with unplated ferromagnetic materials.

re.3.3 According to /4/, the eddy-current signal can be linearized by combination of the signals of two or more geometricly different measuring coils. According to the reciprocity principle and because of easier realization two field coils have been used. A high linearization of the lift-off effect could be achieved with both tests carried out on iron and nonferrous material.

Tests by means of the internal coil IK 1 proved the applicability of this method to signals of support grids with tube inspection, too.

re.3.4 The influence of temperature on the value of capacitance was measured on condition of distilled water in order to eliminate the influence of foreign ions. Hereby it was observed that temperature influences significantly the dielectric constant of water. Water temperature ought to be constant in order to obtain exact distance measurement. In case of strong variations of water temperature during test, temperature ought to be measured separately. Exact informations on temperature distribution in the pressure vessel are required. Investigations on the influence of foreign ions dissolved in the water proved a high dependence on the concentration of the ions (a 3-per cent boric acid was used).

re.3.5 Concrete results have not yet been obtained. Preamplifiers as used in the IFR have been checked regarding the possibility to miniaturize them.

re.3.6 Measured field distributions are available. They can be taken into consideration for a final design of complete pick-up probes.

re. 3.7 Modus of application and limits of defect resolution of the eddy current method for inspection of tubing as applied at present can be considered for judging the tube test methods that are to be extracted from this project.

## 7. Next Steps

re.3.1 Decision on the application of the capacitive distance measurement method according to definition of neighbouring conditions.

re.3.2 Miniaturization of a preamplifier, integration into probes, determination of radiation resistance of the preamplifier, selection and procurement of appropriate cables, measurement of the whole transmission part.

re. 3.3 Evaluation of the eddy current signals of the specific parameters with the designed pick-up coil assemblies and decision on the appropriate geometries.
Manufacturing of coils for the inspection of the straight part of the tube and of the tube bend. Investigation of the differences in test sensibility between tube bend and straight tube.

## 8. Relation with Other Projects

RS 255; RS 256; RS 257

## 9. References

/1/: Nondestructive material testing by means of eddy-current methods for repetitive inspection. Final Report RS 89, Subproject 1.

/2/: Nondestructive repetitive inspection at the RDB by means of eddy-current methods. Subproject 3. Institut Dr. Förster (1976).

/3/: R. Feldtkeller: Theory of coils and transmitters (1971).

/4/: H.L. Libby: Introduction to Electromagnetic Nondestructive Test Methods (1971).

## 10. Degree of Availability of the Reports

/1/, /2/: GRS

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.1977 - 31.12.1977 | 12.3 | RS 230 |
| **Vorhaben/Project Title** | | **Land/Country** |
| Untersuchungen zur Leistungsfähigkeit der akustischen Holographie, vor allem im Vergleich zu fokussierenden Prüfköpfen bei der ZfP<br><br>Investigations of the efficiency of acoustical holography especially in comparison to focussed beams in NDT | | FRG<br>**Fördernde Institution/Sponsor**<br>BMFT<br>**Auftragnehmer/Contractor**<br>Bundesanstalt für Materialprüfung (BAM)<br><br>Berlin<br>Laboratorium 6.21 |
| **Arbeitsbeginn/Initiated** | **Arbeitsende/Completed** | **Leiter des Vorhabens/Project Leader** |
| 1.10.1976 | 31.3.1979 | Dr.Kutzner,Dr.Wüstenberg |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | 31.12.1977 | 496.500,-- DM |

1. General Aim

Development of simplified methods of
- acoustical holography with numerical reconstruction
- flaw sizing by scanning with focused beams.

Application of the developed methods to the analysis of ultrasonic indications under practical circumstances.

2. Particular Objectives

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3. Research Program

3.1 Construction of probe systems

Determination of the decisive criterions (aperture length, resolution etc.) to construct optimal probe systems.
Application of the developed probe systems on test-blocks.

3.2 Software-Development

Elaboration of theoretical foundations (normal incidence, oblique incidence, tandem-technique, distance of scanning points), development of the reconstruction- and interface-software (AD-converter, plotter) for scanning in one- and two-dimensions.

3.3 Holography in tandem-technique

Investigations of the efficiency of this method, experiments on test-blocks.

3.4 Construction and testing of focusing probes

Checking of the available construction method, manipulator for focusing probes, adaption of the focusing probes on the curvature of the surface.

3.5 Experiments with the probe-systems developed for holography- and focusing systems
Testing of the developed systems on the test-blocks, which were produced in other research programs of reactor security, estimation of the efficiency of the developed techniques.

4. Experimental Facilities, Computer Codes

To 3.1 Manipulator system, electronic equipment for acoustical holography

To 3.2 Reconstruction software (FORTRAN), driver-software for AD-converter, CRT-terminal and plotter (Assembler)

To 3.3 Tandem-manipulator, electronic-equipment

To 3.4 Device to the production of acoustical lenses, digital equipment for positionning.

To 3.5 Whole equipment

5. Progress to Date

To 3.1 In the year 1977 some new probes for holography especially for oblique incidence were constructed. The field of this probes should be as wide as possible. Therefore probes with a small transducer width and a special choice of wedge angle were used. Moreover a focusing probe in contact technique for two-dimensional holography was constructed.

To 3.2 Theoretical investigations for the resolution of acoustical holography were made. Especially the limited apertures and the real probe fields were considered. Moreover the choice of other important parameters for acoustical holography, e.g. number and distance of sample points, was discussed. Finally we start with the development of software for two-dimensional holography.

To 3.3 The linear acoustical holography in tandem-technique was used to analyse artificial defects, which were orientated perpendicularly to the surface of the test-blocks. Further some investigations in connection with the HSST research program were made.

To 3.4 An electronic device to produce B- and P-images for the use of focusing probes was developed. Several investigations to check the available construction concept for focusing probes were performed.

To 3.5 In the connection with the HSST research program several experimental investigations for acoustical linear holography in single probe technique and tandem technique were made. Moreover the method of flaw sizing by scanning with focused beams was also used to compare both results.

## 6. Results

To 3.1 The voluminous investigations concerning probe development have shown that for linear holography a probe with a small transducer width (about 3 mm) and a special choice of wedge angle (about 52°) give the best compromise. The maximum of lateral resolution is given by the value of wave length.

To 3.2 In the 1. technical report the whole equipment for linear acoustical holography is described in detail. Further we have found some criterions for the choice of probes as well as numbers and distances of sample points (a detailed discussion of these results will be given in the 2. technical report).

To 3.3 The attempt to analyse artificial defects with linear acoustical holography in tandem-technique was successful. Therefore we also could apply this technique in connection with HSST research program (for results see the special report).

To 3.4 The quality of the B- and P-images produced by the developed electronic device was sufficient. Moreover the available construction concept for focusing probes are well proven.

To 3.5 The results of the measurements in the HSST research program are published in a special report. Especially we have found that the lateral resolution of the holography was better than the resolution of the flaw sizing by scanning with focused beams.

## 7. Next Steps

To 3.1 Development and construction of probes and a manipulator for the two-dimensional holography.

To 3.2 Software development for two-dimensional holography (normal-, oblique- and tandem incidence).

To 3.3 See 3.2

To 3.4 Measurements concerning the comparison between acoustical holography and flaw sizing with focused beams.

To 3.5 Experiments on test-blocks which were produced in other research programs of reactor security.

8. Relation with other Projects

Real-time ultrasonic holography (IZfP-Saarbrücken) (RS 102)
reactor security research program FB 2703

9. References

1. technical report

10. Degree of Availability of the Reports

GRS, Köln

1375

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77-31.3.77 | 12.3 | RS 193 |

| Vorhaben/Project Title | |
|---|---|
| Vorversuche zur Leckageüberwachung mit Hilfe der Schallemissionsanalyse<br><br>Preliminary Investigations on On-Line Leak Detection by Acoustic Emission | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Battelle-Institut e.V. Frankfurt<br>Abt. Betriebsverhalten von Werkstoffen |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| January 1, 1976 | March 31, 1977 | Dr. P. Jax |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Completed | March 31, 1977 | DM 314.800,-- |

## 1. General Aim

The general aim of the program is the development of a continuously operating surveillance system for leaks in the primary system of a nuclear power plant with the aid of acoustic emission. The system should be capable of constantly inspecting all components of the primary system of the plant during operation and locating the position of the leak with sufficient accuracy.

## 2. Particular Objectives

Under this research program it was to be investigated on a test loop whether acoustic emission is suited to detect leaks in the presence of background noise similar to that occurring under actual operating conditions. In particular, it was to be investigated what factors of influence on acoustic emission have to be taken into account (e.g. pressure, temperature, position), whether there is a relationship between leak size and leak type, and what possibilities of locating leaks exist in principle.

## 3. Research Program

Various defined leaks were produced on a test loop of KWU, Erlangen, and the acoustic emission involved was analyzed. The following experimental parameters were varied:

- Enthalpy of the water (pressure and temperature)
- Leak type (crack-type leak, flange-type leak and valve leak)
- Leak size
- Background noise; by variation of the speed of the circulating pump.

4. Experimental Facilities, Computer Codes

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5. Progress to Date

Continuation of the frequency analyses, writing the final report. The research program has thus been completed.

6. Results

Frequency Spectrum

The frequency spectrum of the leakage noises shows a complex fine structure with many maxima and minima. In the stationary experimental phases it remains constant, but changes upon minor variation of the experimental parameters (temperature, pressure, leak size). Systematic relationships between the frequency spectrum and the experimental parameters could not be established, except in one case: in agreement with theoretical considerations, it was found for the borehole-type leak that at constant pressure and temperature conditions the higher frequency components become more intensive with decreasing diameter of the borehole. A similar relationship for the flange-type leak does not exist.

7. Next Steps

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8. Relation with Other Projects

Parallel to the AE measurements in the ultrasonic frequency range (50 to 800kHz) described in the present report, the Allianz-Zentrum für Technik, Ismaning, measured the frequency components of leakage noise below 30 kHz (project RS 97). It was to be clarified in this way whether the audible or the ultrasonic frequency range is better suited for leak detection.

The work reported here was continued with measurements of the background noise on nuclear reactors during operation under projects RS 289 to RS 292.

1377

9. References

P. Jax,
"Vorversuche zur Leckageüberwachung mit Hilfe der Schallemissionsanalyse"
Report BF-R-62.944-1 (March 1977)

10. Degree of Availability of the Reports

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1379

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77-31.12.77 | 12 3 | RS 191 |
| Vorhaben/Project Title<br>Schallemissionsmessungen an bruchmechanischen Proben (Ermüdungsrissen)<br><br>Acoustic Emission Measurements on Fracture Toughness Specimens (Fatigue Cracks) | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Battelle-Institut e.V. Frankfurt<br>Abt. Betriebsverhalten von Werkstoffen |
| Arbeitsbeginn/Initiated<br>Jan. 1, 1976 | Arbeitsende/Completed<br>Sep. 30, 1977 | Leiter des Vorhabens/Project Leader<br>Eisenblätter/Jöst |
| Stand der Arbeiten/Status<br>Completed | Berichtsdatum/Last Updating<br>Dec. 31, 1977 | Bewilligte Mittel/Funds<br>1.236.970,-- |

## 1. General Aim

Correlations are to be established between the measured values of AE and the type and size of and stresses on various defects with a view to using acoustic emission (AE) measurements for the inspection of pressure vessels.

## 2. Particular Objectives

At first all the investigations will be conducted on one type of defect (fatigue crack). The examinations will cover the materials used for the pressure vessels of light-water reactors, i.e. the ferritic steel 22NiMoCr3 7 and the austenitic cladding material X10CrNiNb18 9, as well as materials used for the sodium-cooled fast breeder reactor (SBR), i.e. the austenitic steels X6CrNi18 11 and X6CrNiMo17 13, the ferritic steels 8CrMoNiNb9 10 and X12CrMo9 1, and "Incoloy 800" (X10NiCrAlTi32 20).

To be able to determine the types of acoustic waves which are excited at the surface of thick-walled structural parts when the AE source is located at the surface opposite the transducer or within the wall, the propagation of AE signals will be measured.

## 3. Research Program

### 3.1 Preparation of specimens and construction of the experimental setup

3.2 Design of the data recording system

3.3 Investigations on fracture toughness specimens

3.4 Reduction of data and determination and optimisation of the required correlations between measured AE and fracture mechanical parameters

3.5 Investigation of acoustic wave propagation in thick-walled structural parts

4. Experimental Facilities, Computer Codes

Ad 3.2: To make frequency analysis of single signals up to about 800 kHz possible, a data recording system was been purchased, which comprises a two-channel transient recorder with display (paid from Battelle-Frankfurt funds), a rapid Fourier analyser, and a magnetic tape data store (paid from program funds) including a special interface.

Ad 3.3: A universal tensile testing machine (Instron) with hydromechanical control (25 Mp) is available for the static loading experiments with the single-edge notched (SEN) specimens made from the materials of the SBR. This machine has been adapted for AE measurements by special noise surpression measures.

For the experiments at temperatures up to 550°C an adequate furnace was purchased, special importance being attached to low-noise operation.

5. Progress to Date

Ad 3.1: Modified fracture toughness specimens (2", 3" and 4" CT specimens) were made from the ferritic steel 22NiMoCr3 7 (heat-treated in order to achieve 100-percent low-temperature bainite) and the austenitic steel X10CrNiNb18 9 in addition, tensile specimens were fabricated from broken CT specimens. Some SEN specimens were also made from the SBR materials, and six specimens of the X6CrNi18 11(1.4948) steel were heat-treated at 650°C for 1000 hours (at Interatom) in order to achieve an embrittled structure.

Special chuckings without pin joints were designed and constructed to be able to treat the 3" and 4" CT specimens without interfering friction noise from the pins.

Ad 3.2: The acoustic signals recorded with a broad-band transducer during some experiments on SEN specimens were fed to the frequency analyser identified in Section 4. Single AE signals were analysed for their frequency content.

In the experiments with SEN specimens and the tensile specimens, the RMS value of the AE signals was recorded in addition to the normal AE measurements.

Ad 3.3: During the cyclic loading experiments both the load applied and the loading frequency were repeatedly varied so as to permit the resulting effects on AE to be studied. In the static loading experiments with CT and SEN specimens, the load was applied in steps in order to be able to determine the various causes of acoustic emission as far as possible separately. Fracture toughness speci= mens (modified 2", 3" and 4" CT specimens) from the materials 22NiMoCr3 7, X10CrNiNb18 9(1.4550), and X6CrNi18 11(1.4948) as well as SEN specimens from the materials of the SBR were loaded until frac= ture occurred.

Ad 3.5: The acoustic wave field was recorded at the ground surface of a steel plate (83 mm thick) and on a petrochemical vessel (length 15 m, diameter 2.4 m, wall thickness 22 mm) by time delay and ampli= tude measurements, using approximately point-shaped ultrasonic sources. The parameters used included the distance of the AE source to the surface (full and half plate thickness; source at the sur= face), the frequency and the directional characteristics of the AE source. The ultrasonic sources on the steel plate and on the vessel were located and the measuring results were evaluated.

## 6. Results

Ad 3.3: The results of the fracture mechanical experiments can be summarised as follows:

Taking the AE pulse rate and the amplitude distribution of the signals as a basis, three different mechanisms are identified as the major causes of AE in all the materials investigated:

(a) Friction between the fracture surfaces.
In this case the AE intensity depends strongly on the roughness of the fracture surface and the loading rate.

(b) Processes taking place in the plastic zone at the crack tip, such as fracture of brittle microstructural constituents or delamination of the matrix material from inclusions. The AE intensity here depends on the size of the plastic deformation zone, the ductility of the material and the stress intensity factor. This kind of AE occurs after the initial load, e.g. the upper fatigue load, has been exceeded, but prior to the onset of macroscopic propagation.

(c) Macroscopic crack propagation.
Also in ductile materials is stable crack growth accompanied by AE signals with relatively high amplitudes. These signals are probably due to the formation of small cracks and cavities and their passing into one another. Because of the larger proportion of high AE signal amplitudes at all experimental temperatures, this process can be clearly distinguished from the processes described under (a) and (b), particularly in the ferritic steel 22NiMoCr3 7 and, less well, in the austenitic steels.

Ad 3.5: The most important result of the investigations into the propagation of simulated acoustic wave pulses was that even deep-seated sound sources (full or half plate thickness) are accurately located because the first major signal maximum propagates at the surface of the specimen at the speed of the Rayleigh wave. This was found even for source transducer distances corresponding to one plate thickness only.

The shape of the signals depends critically on the transducer bandwidth: when narrow-band transducers are used, the body waves emitted from the source obviously can change into Rayleigh waves through interference because of their long duration. When recorded by broadband transducers the individual body wave pulses are still resolved, but the envelope of the signal strongly resembles the signal recorded by a narrow-band transducer.

## 7. Next Steps

A report on the work performed so far will be prepared.

## 8. Relation to Other Projects

The work performed under projects RS 0031 and RS 0031 D is continued by the present project, which also covers close cooperation with IzfP, Saarbrücken, and IfaM, Bremen (RS 196).

## 9. References

./.

## 10. Degree of Availability of the Reports

./.

1385

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 12.3 | RS 196 |

| Vorhaben/Project Title | |
|---|---|
| Verbesserte phänomenologische Beschreibung von Schallemissionssignalen und ihre Analyse auf eine bessere Fehlerbewertung<br><br>Improved phenomenological description of acoustic emission signals and their analysis with respect to a better evaluation of defects | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Fraunhofer-Ges., München<br><br>IzfP, Saarbrücken<br>IfaM, IFKM |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.2.1976 | 30.4.1979 | Dr. J. Lottermoser |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | 31.12.77 | 2.574.700,-- DM |

## 1. General Aim

- Development of an acoustic emission testing method for the application on pressure vessel tests on nuclear reactor power plants.
- Detection and evaluation of acoustic emission signals.

## 2. Particular Objectives

- Development of models for the evaluation of acoustic emission signals, firstly for laboratory experiments on CT specimens with fatigue crack on ferritic and austenitic steels.
- Determination of the laws for propagation of acoustic emission signals in thick-walled work pieces (plates).

## 3. Research Program

- Measurements of acoustic emission signals on modified CT specimens.
- Signal analysis and determination of the correlation between signal parameters and the physical phenomena.
- Development and experimental verifications of models.
- Theoretical and experimental investigations on the propagation of acoustic emission signals in thick plates.

## 4. Experimental Facilities, Computer Codes

Electronic system for the determination of the acoustic emission

energy (100 dB range) for single events, event rate, acoustic emission energy rate, distribution functions with respect to acoustic emission energy, computer system for the determination of frequency spectra.
Computer programs for spectral analysis, multiplexing of up to 3 ADC's (Biomation 8100), simultaneous acception of signals in up to 6 channels.

## 5. Progress to Date

5.1. Acoustic emission measurements on CT specimens (50 mm) and SEN-specimens (7 mm thick) of the ferritic steel 22 NiMoCr 3.7 and the austenitic steel 1.4948 at room temperature and 200 and 550 $^{o}$C and metallurgical analysis.

5.2. Development of models for the correlation between acoustic emission energy and stress intensity factor.

## 6. Results

6.1. On a typical tension test on 50 mm CT specimen we have the following acoustic emission results (Fig. 1):

- In general we have first a peak which is due to friction of crack flanks,
- acoustic emission which is due to plastic deformation (after depassing maximum load of fatigue),
- we are observing the Kaiser effect (depending on the material properties there may be acoustic emission due to friction or not),
- single acoustic emission pulses with high energy due to macroscopic crack propagation.

Macroscopic crack growth is correlated with high-energetic AE-signals (energy $\geq 10^{-11}$ J as a first approximation for the steel 22 NiMoCr 3.7). For a demonstration we stopped one tension test just after observing an acoustic emission signal with high energy (approximately $10^{-11}$ J) and we found by metallographic analysis a crack propagation of 6/10 mm in the middle of the specimen and a new crack surface of approximately 10 to 20 $mm^2$ (Fig. 2 is the crack tip).

For the plastic deformation we found a correlation between acoustic emission energy and the fourth power of the stress intensity factor K (Fig. 3).
In fatigue we may have a large amount of AE from friction of the fatigue crack, as may be seen in Fig. 4: for the medium load-region we found at the beginning of fatigue (no crack) nearly no AE, but later (having already a fatigue crack) there is a large amount of AE in this load-region. For a small maximum load corresponding to $K \approx 600\ N/mm^{3/2}$ we observed no friction.

6.2. On the base of the McClintock model we found for the volume of the plastically deformed material in front of the crack a correlation with fourth power of K and sixth power of K (regarding plane stress and plane strain conditions), but for the here mentioned materials and specimen geometries we have only a correlation to $K^4$. This is in accordance with the experimental results (see 6.1.).

Considering microcrack formation and hardening processes we found a correlation between acoustic emission energy E and stress intensity factor K:

$$E = c_1 \cdot K^4 + c_2 \cdot K^{4+2\alpha}.$$

The parameter $\alpha$ is dependent on material properties; an exact definition on the base of material properties has to be given. This is in agreement with experimental results (Fig. 5), where we observed laws with 7th and 9th power.

## 7. Next Steps

- Acoustic emission measurements on specimens with natural defects.
- Theoretical work and experiments on the problems of propagation of acoustic emission signals.
- Interpretation of measurements of the directivity pattern of

natural acoustic emission sources (cracks).

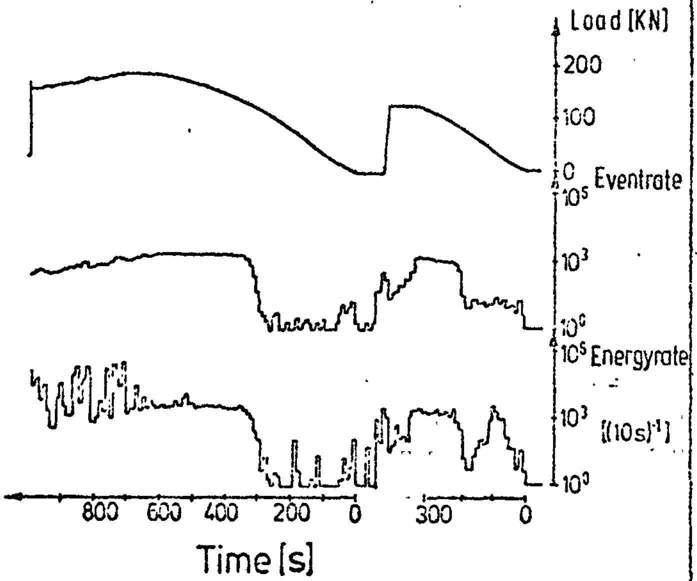
- Calibration of acoustic emission transducers by the reciprocity method.
- Interpretation of acoustic emission measurements.

8. Relations with Other Projects

RS 241.

9. References

10. Degree of Availability of the Reports




| | |
|---|---|
| Tension test on a 50mm CT-specimen of 22NiMoCr37, stroke control with fatigue crack (fatigue from 8 to 80 KN). | IzfP |
| | Fig. 1 |

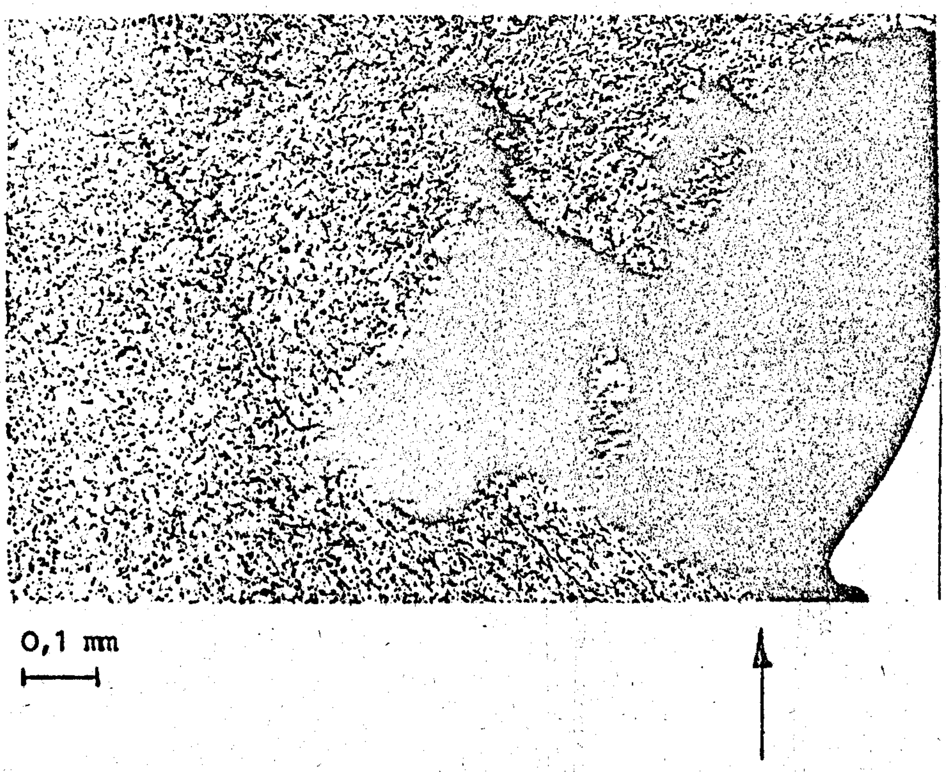
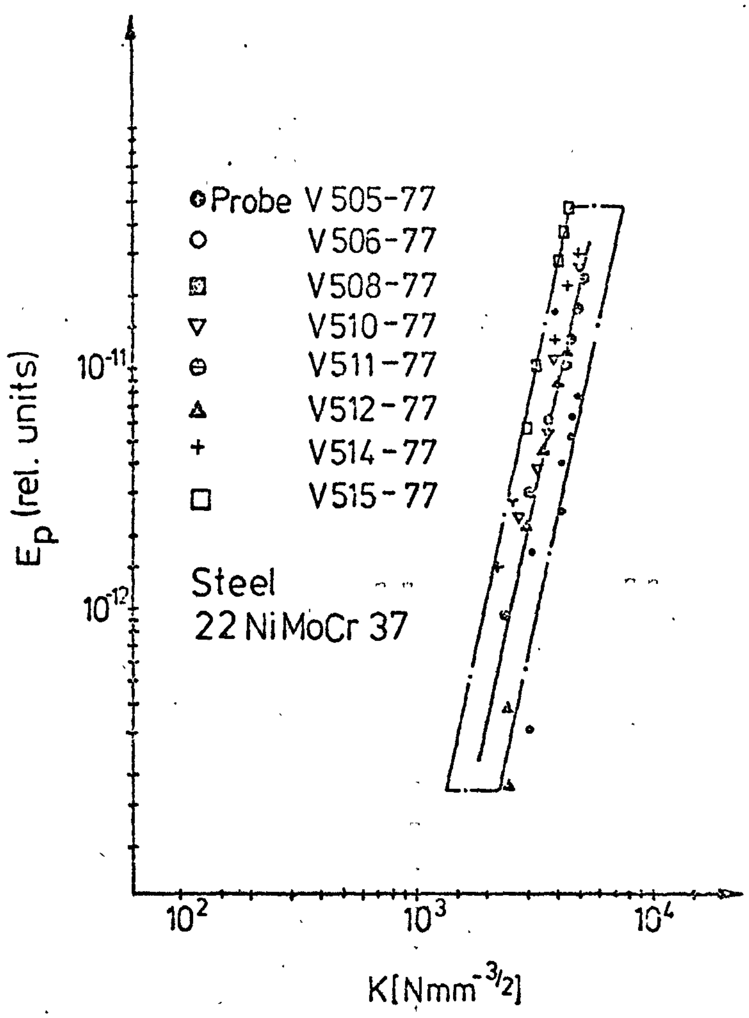
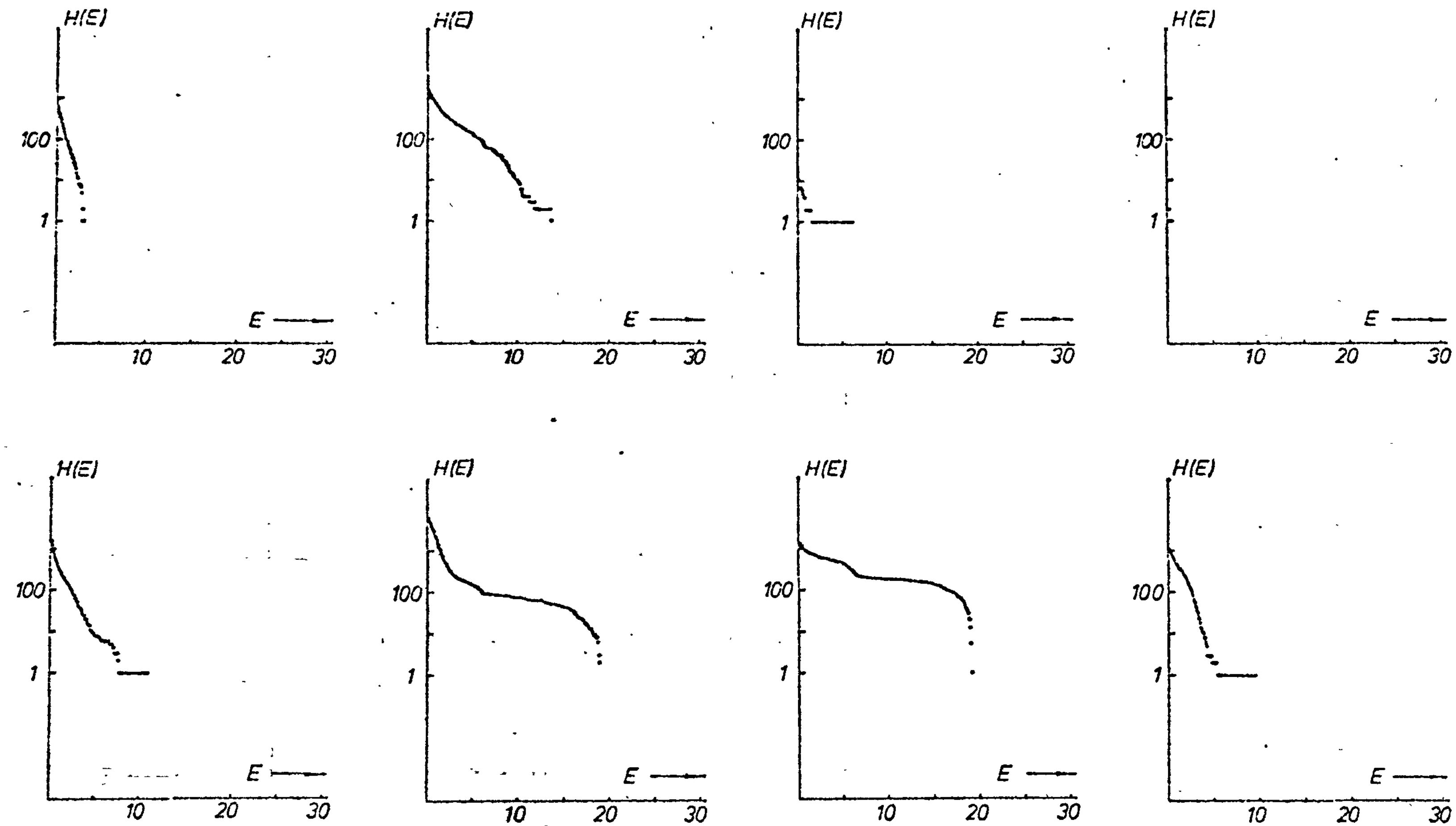



Fig. 2: macroscopip crack propagation on a CT-specimen, detected by AE (arrow indicates the fatigue crack tip)




1 Energyunit ≈ 1 Joule

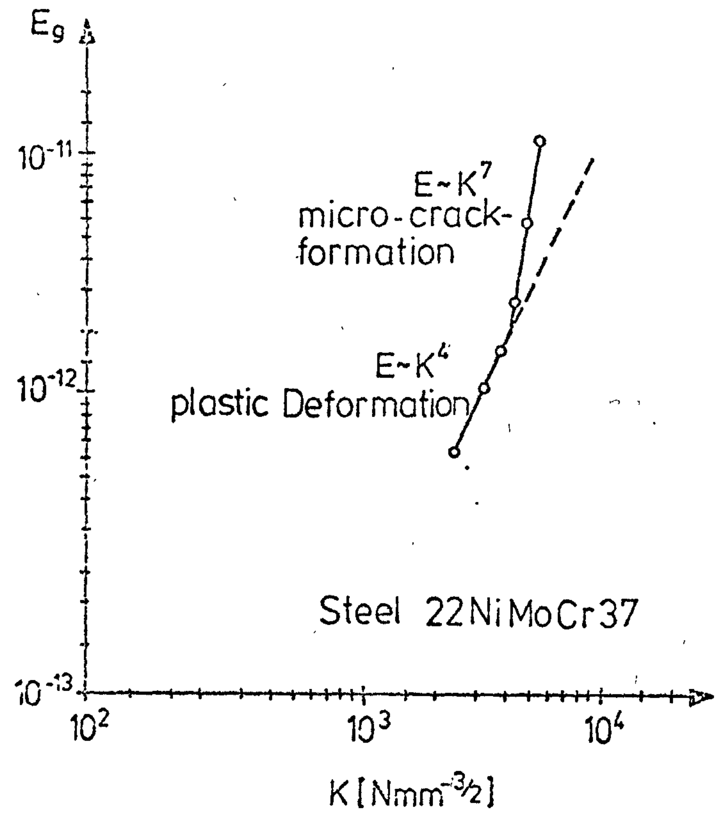
| Energy as a function of the stress-intensity-factor K for plastic deformation. | IzfP |
|---|---|
| | Fig. 3 |




IzfP

Distribution function H of Energy E for fatigue. First row (1-4) for cycles 0-5000, second row (5-8) for cycles 20 000-25000. Load regions: fig. 1 and 5 for 20-70KN, fig. 2 and 6 for 70→80 70KN, fig. 3 and 7 for 70→20KN, fig. 4 and 8 for 20→8→20KN.

Fig. 4




Energy as a function of the stress-intensity-factor K for a steel with micro-crack-formation.

IzfP

Fig. 5

1395

| | | |
|---|---|---|
| Berichtszeitraum/Period<br>1.1.77-31.12.77 | Klassifikation/Classification<br>12.3 | Kennzeichen/Project Number<br>RS 241 |
| Vorhaben/Project Title<br>Verbesserung der Meßtechnik der Schallemissionsanalyse (SEA)<br><br>Improvement of the Instrumentation Technology of Acoustic Emission | | Land/Country<br>FRG<br><br>Fördernde Institution/Sponsor<br>BMFT<br><br>Auftragnehmer/Contractor<br>Battelle-Institut e.V.<br>Frankfurt<br><br>Abt. Betriebsverhalten von Werkstoffen |
| Arbeitsbeginn/Initiated<br>January 1, 1977 | Arbeitsende/Completed<br>December 31, 1977 | Leiter des Vorhabens/Project Leader<br>Dr. P. Jax |
| Stand der Arbeiten/Status<br>Completed | Berichtsdatum/Last Updating<br>December 31, 1977 | Bewilligte Mittel/Funds<br>DM 149.540,-- |

1. General Aim

Extension and improvement of the technique of acoustic emission measurement with respect to the application on large vessels and large-sized specimens under the "HDR" program and the "component safety" program.

2. Particular Objectives

Measurement of another AE parameter, i.e. the peak amplitude of the located AE signal, and automatic statistical evaluation of all individual data by means of the multichannel location system.

3. Research Program

3.1 Installation of a disk system for the computer of the existing multichannel location system for intermediate storage of measured data and programs

3.2 Extension of the measuring system to permit AE measurements at an arbitrary arrangement of the four transducers belonging to one location array

3.3 Installation of amplitude modules for measuring the peak amplitude of an AE signal

3.4 Completion of a FORTRAN program for the statistical evaluation of the measured data and output of the results by teletype

3.5 Mounting of casings and installation of air filters and fans to protect the location system from dust and mechanical damage

3.6 Preparation of a documentation on the software programs and of a final report

4. Experimental Facilities, Computer Codes

Ad 3.1 As mass storage unit, a Floppy Disk system (Dual Floppy Disk Calcomp Model 142 M) was attached to the computer of the location system.

Ad 3.2 Development of the FORTRAN program "AKORT", which determines the location of AE sources by means of three or four transducers per location array, which may be arbitrarily distributed on the structure to be examined. Localization is effected with the aid of an iterative method, the initial solution being the location of that transducer where the AE signal is received first.

Ad 3.3 Six of the eight location arrays (with four transducers each) were provided with amplitude modules to determine the peak amplitude of the AE signal. Furthermore, one module was added which permits measurement of the rise time of the signal and its duration in all location arrays.

Ad 3.4 A previously developed evaluation program was completed and extended. This can be used for the quantitative evaluation of the signal data (transit time differences, coordinates, energy, and AE count) that have been derived on-line during a measurement and stored in digital form on tape after the experiment. To this end, each location array, which in general is surveyed by four transducers in the case of planar location and by two transducers in the case of linear location, is divided into 50x50 zones in the case of planar location and into 50 zones in the case of linear location; the number of signals, their energy and their total AE count in these zones are printed by teletype in the form of matrices. In addition, the number of signals emitted from a specific sector of the location array, their energy, and their AE count are represented as a function of pressure in the form of histograms.

Ad 3.5 Casings were installed to protect the multichannel location system from dust, and in addition air filters and fans were installed.

5. Progress to Date

The work under this program has been completed. The final report is being prepared.

6. Results

Ad 3.1 The attached mass storage unit now permits a quantitative evaluation of the raw data (time differences, locations, etc.) in the course of the measurement, while at the same time signals are accepted and localized qualitatively as points on the CRT screen of the system.

Ad 3.2 In contrast to the previous program, which, for planar location, required three transducers at the corners and one transducer in the center of gravity of an equilateral triangle, the newly developed program permits arbitrary transducer arrangements. A comparison of the location accuracies showed that both programs generally yield the same accuracy, whereas with the new program it may be markedly better for specific areas (around the corner transducers).

Ad 3.3 Results of measurements made after installation of the amplitude modules are not yet available.

Ad 3.4 The evaluation programs have been debugged and successfully used in many cases for the evaluation of measurements on pressure vessels.

7. Next Steps

Completion of the final report.

8. Relation with Other Projects

The measuring technique which has been improved compared with that used in the previous projects RS 31 to RS 31 C will be applied to measurements on vessels and large-sized specimens under the research programs in the field of AE, e.g. in RS 191 A, in the "HDR" program and in the "component safety" program.

9. References

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10. Degree of Availability of the Reports

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| 141-3 -01/4111-10 | | 12.3 |
|---|---|---|
| Titre<br>Programme européen de contrôle de tôles fortes par ultrasons. | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DgCs/DSN |
| Titre (anglais)<br>European Program of ultrasonic testing of heavy section steel plates. | | Organisme exécuteur<br>CEA/DTECH/STA<br><br>Responsable<br>A.C.PROT (DSN-SETSSR) Fontenay |
| Date de démarrage 01/03/77 | Etat actuel<br>en cours | Scientifiques<br>R. SAGLIO (STA) |
| Date prévue d'achèvement 01/07/78 | Dernière mise à jour 8/1/78 | |

1 - Objectif général :

Participation à un programme européen de contrôle de tôles de forte épaisseur comportant des soudures avec défaut volontairement introduits. Action entreprise dans le cadre des communautés européennes avec des tôles mises à disposition des laboratoires européens par les USA dans le cadre du programme HSST.

- Objectifs particuliers :

1) Réalisation d'une cuve d'immersion capable de recevoir les tôles à contrôler. Cette cuve est équipée d'un dispositif d'enregistrement cartographique "en tout ou rien " disponible au laboratoire de la STA et susceptible d'effectuer des cartographies "C-SCAN" à l'échelle 1.

2) Réalisation de traducteurs focalisés.
La participation du CEA a été dès l'origine limitée à l'utilisation des traducteurs focalisés développés à la STA.
Au total 11 traducteurs ont été utilisés.
à 2 MHz : 4 en ondes transversales
3 en ondes longitudinales ) Ø tache focale 5 mm environ.

à 4 MHz : 2 en ondes transversales
2 en ondes longitudinales ) Ø tache focale 2,5 mm environ.

Plusieurs autres traducteurs avec correction de courbure ont été réalisés pour le contrôle de la soudure du piquage.

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3 - Installations expérimentales et programmes :

L'installation comporte la cuve réalisée conformément au § précédent, l'ensemble des traducteurs et l'équipement ultrasonore du laboratoire de la STA.
Le programme comporte la détection des défauts dans la soudure de trois tôles dont une soudure de piquage, leur localisation et leur caractérisation, notamment leur dimensionnement. Pour ce faire la méthode mise au point au CEA en parallèle avec le développement des traducteurs focalisés est utilisée. Elle permet le dimensionnement avec une précision égale au diamètre du faisceau ultrasonore au foyer.
A l'issue des essais non destructifs réalisés tant à Saclay que dans les autres laboratoires français et européens, le programme prévoit la découpe des tôles et l'examen macro ou micrographique des défauts identifiés, afin de déterminer le degré de corrélation
- dans la détection des divers types de défaut
- dans leur localisation
- dans leur caractérisation, notamment leur dimensionnement.
Il est à noter que le programme standard comporte normalement un contrôle en manuel suivant une procédure voisine de celle du code ASME section XI et proposée par le PISC (Plate Inspection Steering Committee) organisme mis sur pied dans le cadre des communautés européennes et responsable du programme européen.

4 - Etat de l'étude :

En ce qui concerne l'étude objet de cette fiche (CEA) l'état d'avancement est le suivant :
- La totalité des contrôles non destructifs a été réalisée. Des cartographies "C-SCAN" à l'échelle 1 sont disponibles ; le contrôle ayant été effectué dans 5 directions :
  - normalement à la surface en ondes longitudinales
  - en ondes transversales obliques à 45° dans deux directionsdans le plan de la soudure et deux dans le plan de section droite.
Le signal ultrasonore a été enregisteé sur bandes magnétiques. Une restitution à grande vitesse (76cm/s) permettra l'analyse de chaque défaut en vue de son dimensionnement suivant la méthode CEA. Cette analyse est actuellement en cours.

5 - Prochaines étapes :

- fin de l'analyse des résultats CEA
- Rédaction du rapport d'essais en vue de la présentation au PISC.
- Participation aux réunions relatives au programme d'essais destructifs.
Il est à noter que l'acheminement des tôles a pris du retard dans plusieurs pays. Deux pays ( l'Italie et l'Espagne) restent encore en compétition avant que la phase 2 (découpe) puisse être entreprise.

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6 - Relations avec d'autres études :

Ce programme entre dans le cadre des études de justification de la méthode utilisant les traducteurs focalisés et de leur aptitude au dimensionnement des défauts décelés.

7 - Documents de référence :

PISC / UK (76) P1
Fiche réf: / HS / GR SR / 100 12 13
" Procedure for ultrasonic examination of P.V.R.C. welded test blocks",
Révision 2 Nov 1976

8 - Disponibilité :

Documents disponibles sous réserve des accords habituels.

1403

| 141-3 -02/4111-10 | | 12.3 |
|---|---|---|
| Titre<br>Détectabilité par ultrasons des défauts en compression. | | Pays<br>FRANCE<br>Organisme directeur<br>CEA/DgCS/DSN |
| Titre (anglais)<br>Ultrasonic detectability of flaws under compression | | Organisme exécuteur<br>CEA/DTECH/STA<br>Responsable<br>DSN-SETSSR)<br>Fontenay |
| ate de démarrage 01/07/77 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 31/12/78 | Dernière mise à jour 12/77 | |

1 - Objectif général :

Etude de l'influence du champ de contraintes sur la détectabilité des défauts lors d'un contrôle par ultrasons.

2 - Objectifs particuliers :

1) Réalisation des éprouvettes
   Elles sont réalisées dans le cadre d'une autre étude.
   Il s'agit d'éprouvettes de fatigue en traction compression, comportant en leur milieu une soudure réalisée volontairement avec un défaut de type déterminé.
   Les défauts sont décelés au stade de la fabrication et dimensionnés tant par radiographie que par ultrasons.
2) Réalisation des traducteurs focalisés.
   Les 4 traducteurs focalisés (Ø tache focale = 2 mm) ont été réalisés :
   2 à 45° en ondes T ; 1 à 60° en ondes T et 1 en ondes L. Les défauts doivent se situer à mi-épaisseur soit 80 mm.
3) Le personnel du STCAN (Service Technique des Constructions et Armes Navales ) qui doit se charger des contrôles en cours d'opération est en formation au laboratoire de Saclay pour s'initier à l'utilisation des traducteurs focalisés.

3 - Installation expérimentale :

Voir à ce sujet la fiche "Propagation par fatigue des défauts naturels des soudures "

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4 - Etat de l'étude :

1) Avancement à ce jour :
Seuls les contrôles en usine des défauts sur éprouvettes ont été réalisés par Creusot-Loire. Les résultats ne sont toutefois pas encore entièrement disponibles.

5 - Prochaines étapes :

Il s'agit essentiellement des essais de fatigue dans les laboratoire de la marine : Chaque période sera précédée et suivie d'un dimensionnement des défauts. La partie de ce travail intéressant cette étude sera la comparaison du dimensionnement en l'absence et en présence d'un champ de contraintes de compression.

6 - Relations avec d'autres études :

L'essentiel des essais de fatigue sera effectué dans le cadre d'une autre étude.

1405

| 141-3 -03/4111-10<br>151-3 -03 | | 12.3 |
|---|---|---|
| Titre<br>Etude des filtrages optiques de films radiographiques | | Pays<br>FRANCE<br>Organisme directeur<br>CEA/DgCS/DSN |
| Titre (anglais)<br>Study of optical filtering of radiographic films | | Organisme exécuteur<br>CETIM<br>Responsable<br>(DSN-SETSSR)<br>~~Fontenay~~ |
| Date de démarrage 19/5/76 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 19/12/78 | Dernière mise à jour 12/77 | |

1 - Objectif général :

Il s'agit de rechercher un procédé, notamment par filtrage optique, permettant d'augmenter le contenu d'informations délivrées par les films radiographiques, notamment en présence d'un rayonnement $\gamma$ (cas des inspections périodiques de circuits primaires)

2 - Objectifs particuliers :

1) Etude bibliographique.
   - Cette étude est maintenant quasiment terminée et a permis de faire une revue exhaustive des procédés envisageables.
2) Etude et réalisation d'étalons de fissures.
   - Elle a pour but la réalisation de fissures de profondeur et d'épaisseur étalonnées, permettant une approche du problème libérée des incertitudes relatives à la dimension des défauts. Elle a comporté notamment la réalisation d'éprouvettes comportant des fissures calibrées, de profondeur et d'épaisseur croissantes, situées à des profondeurs différentes, mais également des fissures de profondeur variable . Le procédé utilisé est celui mis au point au CETIM, qui fait appel à un usinage précis du défaut étalon lequel est ensuite inclus dans l'éprouvette par soudage par diffusion.
3) Radiographie des éprouvettes. Elles ont été effectuées en utilisant les procédures courantes suivies en fabrication, et en respectant au mieux les consignes permettant un bonne reproductibilité.
4) Dépouillement densitométrique.
   Une étude systématique du dépouillement densitométrique est entreprise pour tenter de rechercher les paramètres caractéristiques de la détection des fissures.

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## 3 - Installations expérimentales et programme :

Les installations sont celles des laboratoires de contrôle non destructifs (M. FLAMBARD) et d'optique (M. PARASKEVAS) du CETIM.
Le programme prévoit l'étude systématique des étalons définis en 2, puis celle de défauts réels radiographiés dans diverses conditions.

## 4 - Etat de l'étude :

1) Avancement à ce jour :
   - bibliographie : terminée, rapport en cours
   - réalisation des étalons : terminée, rapport en cours
   - radiographie des éprouvettes : en cours
   - dépouillement densitométrique : en cours
2) Résultats essentiels :
   - il semble qu'existe une taille critique pour la pupille d'examen du densitomètre.
   - Un examen direct au densitomètre ne révèle pas les fissures très fines même visibles à l'oeil
   - Un traitement du signal est nécessaire pour les mettre en évidence (dérivation, intégration...)

   Compte tenu de l'ensemble de ces résultats un effort de réflexion reste nécessaire avant de poursuivre plus avant.

## 5 - Prochaines étapes :

A l'issue de cette période de réflexion et compte tenu du caractère encore fragmentaire des résultats acquis à ce jour, la poursuite du programme ne peut être totalement définie. Une partie essentielle consistera à étudier le comportement de fissures réelles (donc de géométrie non nécessairement rectiligne) afin d'évaluer la différence de comportement avec les étalons.

1407

| | | |
|---|---|---|
| 141-3 -05/4111-6-2 | | 12.3 |
| Titre<br><br>Emission acoustique.<br>Développement des équipements et des méthodes. | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DgCS |
| Titre (anglais)<br><br>Acoustic Emission.<br>Development of equipment and methods. | | Organisme exécuteur<br>DTech/STA/CNP<br><br>Responsable<br>STA - Saclay |
| Date de démarrage 1977 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 1980 | Dernière mise à jour 1/78 | |

1 - Objectif général :

Mise au point d'équipements et méthodes propres à rendre possible l'écoute de l'émission acoustique des matériaux soumis à contrainte en tant que moyen de détection des évolutions dangereuses des défauts, (en particulier, suivi en continu des circuits primaires de réacteurs).

2 - Objectifs particuliers :

Amélioration du traitement des données recueillies lors des essais pour en faciliter l'exploitation et améliorer leur contenu en informations.

3 - Installations expérimentales et programme :

Les installations sont essentiellement celles développées au CEA depuis plusieurs années.

- capteurs piézoélectriques avec et sans guides d'ondes.
- préamplificateurs.
- amplificateurs.
- dispositif de mesures des différences de temps de parcours.

Les programmes en cours sont essentiellement dirigés vers la surveillance des installations en fonctionnement, notamment sur Fessenheim 1, le pressuriseur.

4 - Etat de l'étude :

- Les premiers essais ont eu lieu en 1977 sur le pressuriseur de Fessenheim 1. L'installation comportait 4 capteurs sur 2 tubulures du dôme.
- 2 équipés de guide d'ondes.
- 2 capteurs chauds ENDEVCO.

Les signaux étaient envoyés en salle d'instrumentation.

- Le dispositif d'acquisition de données comportait 4 voies avec microprocesseur permettant de prédigérer les informations, (amplitude et nombre des salves, niveau du bruit de fond) afin de diminuer le volume des informations acquises en sortie sur ruban perforé.

5 - Prochaines étapes :

Il s'agit essentiellement de confirmer les résultats obtenus et de compléter l'étude par la constitution d'un programme de traitement automatique compatible avec les installations à grande capacité de la CISI, (Compagnie Internationale de Services en Informatique, filiale du CEA).

6 - Relations avec d'autres études :

Dans le cadre du contrat passé avec Creusot-Loire et destiné à vérifier les propriétés des traducteurs focalisés, un essai d'écoute de bruits pendant la phase de soudage avec introduction de défauts réels a été réalisé et exploité. Les conditions de cet essai n'ont pas permis de tirer toutes les informations nécessaires. Il a néanmoins montré que la plupart des défauts réalisés émettaient et que cette émission se prolongeait suffisamment pour être dissociée des bruits provoqués par le refroidissement du laitier.

7 - Documents de référence :

Rapport à paraître.

1409

| 141-3 -07/4111-6-21 | 12.3 |
|---|---|

| | |
|---|---|
| Titre<br><br>Détection et dimensionnement par courants de Foucault des fissures sous revêtement (PWR) | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA - DgCS |
| Titre (anglais)<br><br>Under cladding cracks detection and sizing by eddy currents method. | Organisme exécuteur<br>CEA-DTECH-STA-CND<br><br>Responsable<br>STA - Saclay |

| | | |
|---|---|---|
| Date de démarrage 1978 | Etat actuel à lancer | Scientifiques - |
| Date prévue d'achèvement 79 | Dernière mise à jour 1.1.78 | |

1 - Objectif général :

Mise au point d'une méthode de détection d'éventuelles fissures se développant sous le revêtement en acier inoxydable des cuves.

2 - Objectifs particuliers :

Les méthodes de détection actuellement envisagées par ultrasons ne sont pas entièrement satisfaisantes. L'utilisation des courants de Foucault peut constituer une solution plus satisfaisante à condition de travailler à basse fréquence.

3 - Installations expérimentales et programme :

L'étude comporte :

- la réalisation de maquettes représentatives.
- l'étude et la réalisation de sondes spéciales.
- le choix du mode d'excitation.

.../

- la recherche des conditions d'essais optimales en particulier le choix d'un dispositif mono ou multifréquence.
- l'adaptation du dispositif et de la méthode aux cuves de réacteur PWR.

4 - Etat d'avancement :

A lancer en 1978, en collaboration éventuelle avec la RFA dans le cadre de l'accord de coopération.

1411

| 141-3 -08/4111-6-7 | | 12.3 |
|---|---|---|
| Titre<br><br>Contrôle par ultrasons (et radiographie) des soudures mixtes.<br>(collaboration avec RFA) | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA - DgCS |
| Titre (anglais)<br><br>Ultrasonic (and radiographic) testing of dissimilar metal welds.<br>(Cooperation with FRG) | | Organisme exécuteur<br>CEA<br>DTech/STA/CND<br><br>Responsable<br>STA - Saclay |
| Date de démarrage 1978 | Etat actuel<br>en cours de lancement | Scientifiques |
| Date prévue d'achèvement 1980 | Dernière mise à jour 1.1.78 | |

1 - Objectif général :

Dans le cadre de l'accord qui doit être passé avec la R.F.A., mise en commun des expériences dans le domaine du contrôle par ultrasons (et éventuellement par radiographie) des soudures d'acier inoxydable austénitique.

2 - Objectifs particuliers :

Recueillir le maximum d'informations sur les paramètres intervenant dans ce type de contrôle afin de mieux les maîtriser et d'optimiser les procédés.

3 - Installations expérimentales et programme :

Il s'agit essentiellement sur le plan français d'utiliser les traducteurs focalisés développés au CEA, notamment en ondes obliques longitudinales.

.../

Deux étalons seront utilisés :

- Bloc étalon N° 1 représentatif d'une soudure mixte de tubulure de générateur de vapeur.
- Bloc étalon N° 2 représentatif d'une soudure de safe-end (embout de sécurité) d'une tubulure de cuve.

On utilisera essentiellement des ondes L et T en incidence normale et oblique avec des traducteurs focalisés, en émetteur récepteur ou à fonctions séparées, ainsi que des traducteurs du type VINÇOTTE.

Les présentations type A et C seront normalement retenues.

La présentation type B sera limitée aux zones intéressantes.

Pour chaque type de défaut rencontré on étudiera l'influence :

- du type d'ondes.
- de l'incidence.
- de la forme du faisceau ultrasonore.
- de la fréquence.
- du type de méthode.

en relation avec les caractéristiques des soudures.

Des essais du même type seront effectués par nos équipes sur des blocs en provenance de RFA.

De même les équipes RFA effectueront sur les blocs français des essais dont le type reste encore à préciser.

- Etat de l'étude :

Les programmes prévisionnels ci-dessus sont actuellement en cours d'étude et/ou d'approbation au sein du groupe de travail qui a été spécialement mis sur place pour cette collaboration.

- Prochaines étapes :

- Définition des capteurs à utiliser et réalisation.
- Réalisation du programme.
- Exploitation.
- Eventuellement étude et rédaction de nouvelles spécifications.

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| 141-3 -09 / 151-3 -04 /4112-60 | | 12.3 ★ 5.1.3 |
|---|---|---|

| | | |
|---|---|---|
| Titre<br><br>Utilisation de la neutronographie pour l'inspection en service des réacteurs nucléaires et le contrôle non destructif des composants du coeur. | | Pays<br>FRANCE |
| | | Organisme directeur<br>CEA |
| Titre (anglais)<br><br>Use of neutro radiography for in-service inspection of nuclear reactor and for non destructive testing of core components. | | Organisme exécuteur<br>CEA/DSN/SEESNC VALDUC |
| | | Responsable<br>DSN/SEESNC/Valduc |
| Date de démarrage 01/01/73 | Etat actuel<br>étude en cours | Scientifiques |
| Date prévue d'achèvement 31/12/78 | Dernière mise à jour 15/12/77 | |

1-Objectif général

Le but de ce programme est d'effectuer des essais de contrôle par neutronographie de composants de réacteurs. Le mini-réacteur pour neutronographie MIRENE, mis en service en 1976 permettra de mener à bien cette action. L'installation est équipée de 2 faisceaux collimatés de neutrons qui servent à effectuer les essais d'examen de pièces.

2-Objectifs particuliers

Valeur du procédé pour :

1 - Examen de pièces épaisses en acier

2 - Examen d'éléments combustibles de réacteurs à eau

3 - Examen de matériels divers (ensembles collés)

3-Installations expérimentales et programmes

Réacteur pour neutronographie MIRENE

Essais et calibration du réacteur et des faisceaux de neutrons

4-Etat de l'étude

Après divergence du réacteur en Novembre 1976, les essais de mise en service du réacteur ont été effectués. Des neutronographies d'objets et matériels divers ont été réalisées qui ont montré que l'installation fonctionnait avec satisfaction et que la qualité des clichés de neutronographie était bonne. Les documents de sûreté du réacteur (CR des essais, RGE) ont été rédigés ou mis à jour.

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5 -Prochaines étapes

Mise à jour du rapport de sûreté de MIRENE suite aux essais.

Essais d'examens de pièces test en acier et pièces diverses de réacteur électrogène susceptibles d'être démontées et transportées pour le contrôle dans une installation de type MIRENE.

6 -Relations avec d'autres études

Comparaison avec les autres méthodes de contrôle non destructif en ce qui concerne l'examen de composants de réacteur et l'examen d'objets et de matériels divers.

7 -Documents de référence disponibles :


1 - "Source utilisant la bouffée de neutrons produite par un saut de réactivité dans un réacteur" - M. HOUELLE - supplément au Bulletin d'Information de l'ATEN N° 90 (Juillet - Août 1971).

2 - "Radiography with Neutrons" - M.HOUELLE, C.MERCIER, H.REVOL - British Nuclear Energy Society - Conference at the University of Birmingham - 10/11 september 1973

| PROJECT TITLE : RELIABILITY OF CLASSICAL NDT | 12.3<br>11.2.3 LWR<br>11.1 |
|---|---|
| SPONSORING COUNTRY : Commission of the European Communities | ORGANISATION : JRC Ispra |
| DATE INITIATED : 1977<br>DATE COMPLETED : 1980 | PROJECT LEADER : S.J. CRUTZEN |

Description :

1. GENERAL AIM

Among the actual non-destructive testing techniques on nuclear reactors components, ultrasounds and radiography are principally used. The experience however teaches that often dubious results are obtained when NDT techniques are applied. Development and assesment of ultrasonic inspection methods require more precision, resolution and reproductibility. Besides global detection and localization methods, sizing techniques are required. Classical ultrasonic testing could give better reliability if known correctly in its parameters. Therefore, characterisation studies leading to simple methods to be used or on the spot or in maintenance laboratories or in research laboratory are proposed.

2. PARTICULAR OBJECTIVES

1. Critical analysis of the transfer function of standard ultrasonic testing chain in order to assess the influence of the various components and of the operational conditions, on the overall reliability.

2. Study of the correlations between U.S. signals and reference artificial defects. This study will involve the use of the NDT Techniques, destructive techniques and the fabrication of well defined reproducible artificial defects.

3. Practical applications
   a) characterisation of ultrasonic apparatus (in particular transducers) to be used for quality control or in service inspection, Up to now, requests were received from : ENEL (Lab.Centrale,Piacenza) CNEN,CEA-Cadarache, IRSID, NERATOM, BREDA, DOEL Reactor, NUKEM. Collaboration is foreseen with : AERE-Harwell, CEA(Saclay and Fontenay-aux-Roses) NUKEM, DANISH WELDING INSTITUTE, ASSOCIATION VINCOTTE and transducer manufacturers (up to now: AEROTECH, BALTEAU, NUKEM, CEA, ISTITUTO CORBINO)

   b) HSSTP (PISC).Ispra will receive the test pieces in Summer 1977 and will try to answer the specifications but also take the opportunity to demonstrate the necessity of characterisation techniques for ultrasonic transducers, for reproducibility in defects detection and localisation A collaboration will be set-up between ENEL(Lab.Centrale di Piacenza) and EURATOM (NDT Section) as the final report has to be unique for Italy. Ispra will take care mainly of what concerns characterisation and calibration as well as original methods other than the one's

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specified by the ASME XI (focal probes).

c) Identification of the failed fuel pins in an irradiated BWR fuel bundle using ultrasonic without any dismantling of the bundle. This activity was initiated in the frame of the Technical Support to Power Stations Programme and will be performed in collabroration with CNEN-ENEL on the practical case of the GARIGLIANO-CAORSO fuel bundles (with the already received agreement from the ACPM).

d) In the frame of a collaboration contract (EUR-CEA-CNEN-203/75/PIHOIF) included in the Safety Programme, some work will be persued for the quantitative direction of cracks (in quantity and in depth), in sodium ducts submitted to thermal shocks. The feasibility study demonstrated the reliability of the X-ray techniques combined with the use of well defined reference defects and of a densitometer.

3. PROJECT STATUS

This study is in fact the partial continuation of a study initiated under the name "Quality Control" in the frame of the "Technical Support to Power Stations" Programme of the CEC. Laboratories are well equiped for:

- rather fundamental studies and results are available for what concern characterisation of transducers and materials
- systematic characterisation of ultrasonic equipment
  - Schlieren bench (high resolution)
  - point by point precise examination
  - correction of transducers
  - liquid crystal characterisation bench
- fabrication of well shaped and reproducible reference defects using
  - punch method
  - electric discharge method.

4. NEXT STEPS

4.1. Rather fundamental studies concerning the characterisation of the transducer : intrinseque characteristics;

4.2. Characterisation of some materials using
- echo phase shift analyses
- spectrum analyses

4.3. Adaptation of the knowhow of the punch and electric discharge techniques for the fabrication of reference defects in the frame of Inservice Inspection of reactor pressure Vessels.
Utilisation of forming and brazing techniques for fabrication of reference defects in reference blocks.

4.4. Characterisation of the Ultrasonic equipments of ENEL (Laboratorio Centrale di Piacenza)
Characterisation of some probes for NERATOOM.

4.5. Participation at the HSST (Pisc) programme (Sptember-October 77

4.6. Definition and fabrication of the prototype apparatus for identification of the failed fuel pins in the GARIGLIANO BWR irradiated fuel bundles (CNEN-ENEL-ALCI)

4.7. Definition of the resolution of the "X-rays-densitometry" method for the quantitative (depth) detection of cracks in sodium tubing submitted to thermal shocks.

5. REFERENCE DOCUMENTS


(1) Proceedings of "Information Meeting on Characterisation of Ultrasonic Equipment" S.Crutzen and all, Ispra, Novembre 25th and 26th, 1976
Euratom J.R.C.-Ispra NDT Section (to be issued)

(2) E.E.BORLOO, P. JEHENSON
Caractérisation des transducteurs ultrasonores. St.Etienne (France) 30 Novembre et 1er Décembre, 1976
Société Française de Métallurgie. Colloque sur les progrès dans les méthodes d'investigations des métaux.

(3) R. DENIS
Characterisation of ultrasonic transducers using Cholesteric Liquid Crystals. J.R.C.-Ispra, Materials Division NDT Section EUR 5710.e, 1976

(4) S.CRUTZEN, R.DEBEIR, B.JEHENSON, and F. LUCHTMANS
JRC-Ispra, Italy
A. BAROSI, L. ROSAI, T.A. GIORGI, Saes-Getters, Milan, Italy
Use of neutron Radiography for quantitative measurement of sorbed Hydrogen in Getters and quality control of nuclear fuel pins. - IAEA/SR-7-21, Oslo, May 1976.

(5) J.R.C. -ISPRA, Biannual Report 1974-1975 EUR 5550.e


6. BUDGET

6.1. Total investments on the past programme actions to be continued (Technical Support to Nuclear Power Stations")
nearly 100.000 u.a.

6.2. Total investment for 1977-1980 / ≃ 200.000 u.a.

7. PERSONNEL

1977-1980 : 6 men/year

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.9.1976 - 31.10.1977 | 12.4 | RS 216 |

| Vorhaben/Project Title | |
|---|---|
| Studie über Personalqualifikation für zerstörungsfreie Prüfungen im Bereich Reaktorsicherheit<br><br>Study on Non-Destructive Testing Personnel Qualification for Reactor Safety | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Deutsche Gesellschaft für Zerstörungsfreie Prüfung eV. (DGZfP) |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.9.1976 | 31.10.1977 | Ing.(grad.) Trusch, Dr. Wüstenberg |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| completed | December 1977 | 100.000,-- DM |

1. General Aim

Inventory, demand analysis and proposals for NDT personnel qualification for reactor safety.

2. Particular Objectives

–

3. Research Program

Collection of empirical data by special questionnaires and interviews with experts in selected companies and institutions. 242 questionnaires sent out, 103 returned (≙43 %), 84 completed. - 18 interviews with experts in the field of reactor safety or in institutions with NDT training.

4. Experimental Facilities

–

5. Progress to Date

see 6

6. Results

6.1 Present State:

Complete training system only through DGZfP but not tailor made for reactor safety. 1964 - 1970 11.000 persons trained in companies, 1970 - 1976 4.000 persons trained through DGZfP. Strong growth tendency in DGZfP. Equal figures for in-plant training and DGZfP training expected for 1977. In-plant training often modeled after U.S. pattern, particularly abroad. Foreign systems comparable only in limited way.

Estimated total personnel in general NDT 7.000 - 8.000. Cir. 1.700 persons in companies who also conduct tests in reactor installations.

6.2 Personnel Demand

Estimated increase until 1985 70 %: + 5.000 - 5.500 persons in general NDT, with replacement of retiring personnel cir. 7.000 to be newly qualified. Expected growth rate in nuclear technology until 1985 around 100 %, i.e. cir. 2.000 persons, with replacements and fluctuations estimated qualification demand 3.000 persons. Bottle necks in training facilities only avoidable through active promotion.

6.3 Required Type of Training

International comparability also with American system desirable, greater offer of courses for automatic testing.
Development of a formal job profile would also bring undesirable requirements. Necessary enhancement of extent and division of former training: expansion to lower and higher levels of training, adjustment to company requirements.

6.4 Conclusions

Expansion of training offer in quality and quantity, specifically for reactor technology: thorough and specialised training with specific requirements for supervisory personnel.

6.5 Proposals

Specific job analysis in testing reactor components, development of a specialised qualification system, preparation of a sufficient number of lecturers who themselves analyse and develop learning steps and syllabi for nuclear technology. Availability of training facilities, and teaching aids. Organic integration with present system (DGZfP). Coordination of university lectures.

7. Next Steps

This project is already completed.

8. Relation with other projects

—

9. References

Final report October 1977.

10. Degree of Availability of the Report

BMFT, Bonn / GRS, Köln.

in a final report.

6. Results

—

7. Next steps

—

8. Relation with Other Projects

—

9. References

Final report BF-R-63.119-1

1o. Degree of availability of the Reports

Report may be obtained from GRS-FB.

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 13 | RS 219 |

| Vorhaben/Project Title | Land/Country |
|---|---|
| Vergleich verschiedener Druckbehälterkonzepte für Leichtwasserreaktoren<br><br>Comparison of Different Pressure Vessel Concepts for Light Water Reactors | FRG |
| | Fördernde Institution/Sponsor<br>BMFT |
| | Auftragnehmer/Contractor<br>BATTELLE-INSTITUT E.V.<br>Frankfurt am Main<br>Abt. Energietechnik |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.9.1976 | 30.11.77 | G. Langer |
| Stand der Arbeiten/Status | Berichtsdatum/Last Updating | Bewilligte Mittel/Funds |
| Completed | 31.12.77 | 311.105,-- DM |

1. General Aim

The aim of the study is the clear presentation of the characteristic properties of various pressure vessels and a comparison of these pressure vessels on the basis of general comparative criteria. With the aid of a technological assessment, the most suitable pressure vessel concept for a possible back-up solution for the thick-walled steel pressure vessel is to be ascertained.

2. Particular Objectives

3. Research Program

The study has been carried out in two steps.

3.1 Collection of the Necessary Data

The information necessary for comparison has been collected by a literature survey. It has been supplemented by data obtained in interviews and discussions with manufacturers and consultants.

3.2 Technological Assessment of the Concepts

The individual concepts have been assessed by utility value analysis.

4. Experimental Facilities, Computer Codes

5. Progress to Date

The work has been completed and the results are documented

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in a final report.

6. Results

–

7. Next steps

–

8. Relation with Other Projects

–

9. References

Final report BF-R-63.119-1

10. Degree of availability of the Reports

Report may be obtained from GRS-FB.

# 14. PROBABILISTIC METHODS OF SAFETY ANALYSIS

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 01.01.77-31.12.77 | 14 | RS-189 |

| Vorhaben/Project Title | |
|---|---|
| Rechenprogramm für Zuverlässigkeit und Verfügbarkeit von Gesamtkernkraftanlagen mit Hilfe der Zustandsanalyse<br><br>Computer Program for Reliability and Availability of Nuclear Power Plants Using the Method of State Analysis | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: INTERATOM<br>5060 Berg. Gladbach 1<br><br>Abt. Zuverlässigkeit und Strukturdynamik |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 01.01.76 | 31.12.78 | Dr. Zeibig |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | December 1977 | 506.081,-- DM |

1 General Aim

Development of quantitative methods for the assessment of safety, availability and risk of nuclear power plants.

2 Particular Objectives

It is planned to use the method of state analysis as an aid in estimating the risk of nuclear power plants. For this purpose the relevant states of the plant have to be found. Possible transitions between these states have to be identified. The mathematical methods for calculating the state probabilities have to be developed. The chief objects of the project are:

- a computer program for this purpose
- the performance of an exemplary state analysis for a specific plant
- documentation in order to permit general use of the method

3 Research Program

The project consists of the following tasks:

3.1 Definition of states
3.2 First calculations
3.3 Extension of methods
3.4 Revision of computer programs
3.5 Improvement of details
3.6 Test runs, variation of parameters

3.7 Documentation

4 Experimental Facilities, Computer Codes

(3.1) A complete set of relevant exclusive states of the plant has to be found. The possible transitions between these states have to be identified and transitions rates have to be estimated taking into account failures, inspections, and repair of components or subsystems.

(3.2) First calculations for the state model can be performed using the INTERATOM-program MARKOV which calculates the solution of medium sized Markov processes analytically.

(3.3) The problems expected in performing 3.1 and 3.2 have to be solved by improving the methods of system analysis and calculation.

(3.4) Results of 3.3 have to be integrated into the code MARKOV or other codes that have to be written in addition.

(3.5) A more detailed set of states for a specific plant has to be found on the basis of the more general model 3.1.

(3.6) Test runs have to be performed. The influence of uncertainties of the input data and the potential of the method to cover these uncertainties has to be demonstrated by variations of input parameters.

(3.7) The code to be developed and the method of state analysis is planned to be made available for general use. Therefore, much care has to be devoted to the task of documentation.

5 Progress to Date

(3.1) The state model of the reactor plant was extended by taking into account the storage for fuel elements (sodium- and gas-storage). The model is developed for the fast breeder reactor plant KKW-Kalkar.

Work is started to create a state model on the subsystem-level in order to get the failure combinations of - for example - one power supply redundancy with decay heat removal systems of the reactor or the fuel element storage. Going down to the component level one increases the number of states to such a degree, that the handling of the model is unpracticable. To determine the large number of transition rates for subsystems analytical approximation formulas were developed to avoid separate fault tree analysis.

(3.4) A computer code ZUSTA was developed and tested, which allows to handle state models in which the Markov characteristic (independency of history) is not valid. By using variable dimensions it was achieved that the number of states is not limited except by the storage capacity of the computer used.

6 Results

—

7 Next Steps

(3.1) The elaboration of the state model for KKW-Kalkar will be continued.

(3.2) A further development of the computer code ZUSTA is necessary expecially with regard to reduce the number of states (registrating the time history for non-Markovian transitions etc.)

8 Relation with other Projects

—

9 References

—

10 Degree of Availability of the Reports

—

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1. 77-31.12.77 | 14 | RS-201 |
| Vorhaben/Project Title: Zuverlässigkeitsbeurteilung für den Sicherheitseinschluß (SE) am Beispiel des Druckwasserreaktors<br><br>Reliability Assessment of the Secondary Containment of a PWR | | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Techn. Univ. München<br><br>Inst.Bauingw.III,GRS, Abt. f. Werkstoffphysik |
| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
| 1.6.1976 | 31.5.1978 | Prof.H.Kupfer,Dr.G.I.Schuëller |
| Stand der Arbeiten/Status | Berichtsdatum/Last Updating | Bewilligte Mittel/Funds |
| continuing | December 1977 | 492.700,-- DM |

1. General Aim

The general aim of the project is the assessment of the structural reliability of a PWR which may be subjected to internal and external load conditions, i.e. hazards. The structural reliability, as defined here, is the probability of survival of the structure within its design life.

2. Particular Objectives

Based on the probability of occurrence of various load conditions and the utilization of stochastic processes the reliability for each particular external load case i.e. earthquake, airplane crash, external pressure wave will be determined on one hand and on the other hand the reliability, given an internal load condition, is to be sought.

Particular emphasis will be given the determination of the joint probability of occurrence of the earthquake together with other failure events. Stochastic models for the prediction of these joint probabilities will be developed. The scale parameter for the combinations of the particular load intensities is the reliability which needs not to be higher than that for each separate load case. Furthermore the problem of the stochastic treatment of the fracture of shell structures, with reference the probabilities of failure of those structures, is investigated. Load effects are determined by static and dynamic structural analysis. The concept to be developed during course of this research project will be applied to the containment of the powerplant BIBLIS B.

3. Research Program

Essentially the investigation consists of seven problem areas:

3.1 Determination of the various failure events or failure paths of the system which are to be taken into account, considering the originating events inside and outside of the plant. This, of course, with reference to the containment.

3.2 Determination of the types of loads to be expected acting on the containment and their characteristic values based on the statistically distributed failure events as a result of 3.1.

3.3 Calculation of the relevant load spectra which result from the various types of loads with reference to the reliability assessment.

3.4 Evaluation of the materials properties under load effects which are generated by loads as described above. Consideration of fracture mechanical methods for impact loads.

3.5 Evaluation of the load effects using stochastic theory of plasticity.

3.6 Calculation of the structural failure probabilities under single and combined loading.

3.7 Global reliability assessment of the containment.

4. Experimental Facilities, Computer Codes

(With reference to item 3.: Research Program)

3.1 Given a LOCA the ZOCO-Program is used to evaluate the load conditions due to the resulting pressure- and temperature increase.

3.2 The mass- and energy-flow of water and steam out of the primary circuit into the containment during a loss of coolant accident is calculated with the code BRUCH at blow-down phase and with the code WAK at refill phase. The built-up of pressure and temperature in the steel hull is calculated with the code ZOCO.

3.4 An experimental set-up to simulate impact load conditions has been developed. Circular mortar discs can be tested. The force spectrum as well as the resulting strains can be recorded simultaneously. Parallel runs under quasi-static loading conditions can be carried out in a controlled way.

3.5 For the calculation of the load effects under static and dynamic load action in the linear and non-linear range, the SAP and NONSAP-program systems are used respectively.

5. <u>Progress to Date</u>
(With reference to item 3.: Research Program)

3.1 According to the working schedule the evaluation of the internal loads resulting from the LOCA "Large Leak" was analysed as the first case. For this purpose it was assumed that a leak with a cross-sectional area of >1000 $cm^2$ occuring between the main pump of the coolant and the pressure vessel is the worst case. According to WASH·1400 this failure event has an estimated probability of occurrence of $10^{-4}$/a.

Following this major failure event at the end of blow-down and during refill and low pressure recirculation phase maximums of pressure and temperature are possible. The different accident sequencies during refill phase result from function or failure of the systems reactor protection, high pressure injection and low pressure injection. Four accident sequencies of refill phase have been computed to get a first glance at the different possibilities of the entire spectrum of accident sequencies. During blow-down and low-pressure recirculation phase only one accident sequence had to be analysed.

For calculating the pressure values with the Program ZOCO the following input data have been evaluated:

- the simplified one-room model of the reactor building with internal walls
- the mass and energy flow from the primary system into the containment during blow-down phase resulting from the code BRUCH
- the mass and energy flow from the primary system into the containment during refill phase resulting from the code WAK
- material properties.

The influence of 18 parameters on the maximum overpressure at the end of blow-down was checked by a sensitivity study. By means of this study the first terms of the taylor series could be calculated. After evaluating the statistical values of the parameters, the standard deviation of the resulting overpressure at the end of blow-down was determined. Assuming normal distribution for the parameters and taking in account only the first terms of the taylor series, the maximum overpressure at blow-down is also normally distributed.

Besides the integral loading of the steel hull due to internal pressure following a loss of coolant accident, the steel hull can be also locally constrained by missiles as a result of a steam line break. Up till now a double-ended rupture before the last bending inside the steel hull was considered. In such a case, because of the assumed elastic connection (bellow) to the steel hull, the pipe would wing in the form of a circular pendulum till it strikes the containment after $90^{o}$.

3.4 An experimental set-up to test dynamic loading of circular mortar discs has been designed. In this way we can measure simultaneously the force function generated in the missile and the strains at different places on the disc. The time dependence of the implied force and the resulting strains can be observed with a sampling rate of up to $10^{6}$ points/sec. The transient-recorder is connected with a table computer for further processing of the data. With the help of the computer a Fourier analysis of the strain spectrum is calculated. The damping coefficient is also obtained. Parameters of dynamic fracture mechanics are calculated. All calculated data as well as the recorded force and strain functions are finally plotted or printed.

Up to now 20 mortar discs have been tested. The impact load conditions have been varied.

One part of this investigation deals with the dynamic excitation of a mortar disc under impact load. In this way we try to simulate the conditions in a conrete containment. The aim of the second part is primarely to evaluate materials properties.

3.5 The steel hull of the containment under the load case large LOCA was analysed. The structure was modeled by 250 shell elements. The material and the shell locks have been considered as penetrations. The area in the vicinity of the material lock was

was analysed taking into account geometric nonlinearities. In addition the problem of the steel hull under pipe whipping has been treated. The dynamic analysis of the containment under earthquake loading has been carried out. The FE model includes the possibility of consideration of the soil properties at the site. Failure probabilities have been calculated utilizing the ElCentro Earthquake record. The load cases "external explosion" and air plane crash have also been analysed. For this purpose the modes of the reinforced concrete dome have been determined.

3.6 Structural failure probabilities of the steel hull have been calculated for the following load conditions: earthquake, internal pressure and temperature build up due to LOCA and pipe whipping. The failure probability of the concrete dome under external pressure wave has been determined as well.

6. Results

3.1 The numerical analysis of four failure event sequencies of the refill and low-pressure phase showed that the maximum overpressure at the refill phase is lower than that of the end of blow-down (Fig. 1). Furthermore an estimation for the low pressure recirculation phase with only one coolant pump working showed again a clearly lower relative maximum /2/.

The maximum overpressure in the steel hull and its distribution is therefore - core melting excluded - determined by the blow-down phase, irrespective of the number of functioning colling units.

The results of the sensitivity study, the standard deviation of the parameters and the corresponding change of the blow-down pressure are listed in table 1. Table 1 shows that one can judge the influence of one parameter only, if one knows both the derivation per parameter and its expected deviation.

The standard deviation of the maximum overpressure in the containment of the nuclear powerplant Biblis B after a large LOCA is the square root of the sum of the quadratic single deviations. It is 0,27 bar corresponding to a mean value of 3,77 bar.

3.4 The Fourier analysis of the dynamically excited strains indicates that several different vibrational modes are present.

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Flexural vibrations with the natural and higher modes are generally excited. In addition pulsation of the mortar disc is recorded. Under high impact rate a considerable amount of the energy is transferred to higher mode vibrations.
The dynamic excitation of a mortar disc has been studied in detail. Different transfer processes such as travelling hinge and flexural strains can be clearly separated.
It has been shown that the force function of a given missile is severly influenced by the vibrating structure. It could be shown experimentally that the duration of the implied impact is approximately doubled (in comparison with an impact on a rigid wall) if the duration of the vibration of an essential mode is comparable to the duration of the impact. The peak load, however, is reduced by a factor of two.
An extensive literature survey of materials properties under high rate of loading has been carried out. Results have been presented at the SMiRT'77 /2/. From our experimental results we can conclude that failure strain can be increased up to 6-times the quasi-static value at strain rates of $\dot{\varepsilon} = 50s^{-1}$. The ultimate load, however, is only doubled under the same loading conditions. With increasing rate of loading the failure mode changes from normal flexural to local shear failure.
The maximum crack propagation in the mortar disc has been estimated to be 250 m/s.

3.5 and 3.6 Assuming linear elastic structural behavior, a stress concentration factor of 1.7 has been determined for the boundaries of the penetrations of the steel hull, while the Weibull distribution shows a good fit to the yield data of the containment steel, the Gumbel distribution reveals an excellent fit to the reference stresses (v. Mises) resulting from the large LOCA. Failure probabilities for the fracture mode have been compared with the yield failure condition. Utilizing the Griffith equation a total failure probability of the hull of $6,4\cdot10^{-8}$ has been obtained for an undetected crack of 2 cm length. The respective partial failure probability is $2.5\mathrm{x}10^{-3}$. The respective values for the yield failure conditions are $9.3\cdot10^{-14}$ and $9.8\cdot10^{-4}$. For the El Centro earthquake loading a failure probability of $3.5\cdot10^{-3}$

has been calculated. For this case a relative displacement between the steel hull and the reinforced concrete dome of 12 cm - at the height of the material lock - was obtained. Taking into account the damping properties of the soil, the failure probability of the hull reduces to $2.7 \cdot 10^{-3}$. The analysis of the load case of external pressure wave resulted in a failure probability of $10^{-18}$.

7. Next Steps
(With reference to item 3.: Research Program)

3.1 The time dependent pressure- and temperature values for small LOCA are calculated using ZOCO. The frequencies of their occurrence and their distributions are calculated using failure event sequence analysis with accompanying failure tree computations and probabilistic considerations for single failure sequences.

3.4 Further experiments with circular mortar discs are planed. The dynamic excitation under various impact load conditions shall be studied. Both theoretically and experimentally we try to define worst cases.
Additional tests using mortar bars shall be carried out. In these experiments failure can be more precisely determined. Therefore materials properties are measured in a more reliable way. Special emphasis shall be placed on the scatter of materials properties under impact load.

3.5 The calculation of the load effects of the steel hull and the concrete dome will be continued utilizing German earthquake record. Load effect calculation will be continued for the load case of the plane crash as well.

3.6 Failure probability calculations will be carried out for the various load cases mentioned in item 3.5. Combined load cases will also be considered.

8. Relation with Other Projects

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## 9. References

[1] Schuëller, G.I.: A Concept for the Reliability Assessment of the Containment of a PWR, Proc. Int. Conf. Nucl. Syst. Rel. Engr. Risk Assess., Univ. Tennessee, Gatlinburg, June 20-24, 1977.

[2] Augustin et al: A Complex Study on the Reliability Assessment of the Containment of a PWR, Proc. 4th Int. Conf. Struct. Mech. in Reactor Techn., San Francisco, Aug. 15 - 19, 1977.

[3] Schuëller, G.I. and J. Bauer: Reliability-Based Design of the Containment of a PWR, Proc. 2nd Int. Conf. Struct. Safety and Reliability (ICOSSAR'77), Werner-Verlag, Düsseldorf, 1977.

## 10. Degree of Availability of the Reports

Table 1

Derivatives ($a_i$) of the overpressure with respect to the parameters $x_i$, standard deviations ( $\sigma_{x_i}$ ) of the parameters and difference ($\Delta p_i$) to the expected value of the pressure ( $\Delta p_i = a_i \cdot \sigma_{x_i}$ )

| Parameter | $a_i$ [bar/%] | $\sigma_{x_i}$ [%] | $\Delta p_i$ [bar] |
|---|---|---|---|
| volume of the containment | $-3,2 \cdot 10^{-2}$ | 1 | 0,03 |
| mass of primary coolant | $3,0 \cdot 10^{-2}$ | 1 | 0,03 |
| specific energy of the primary coolant | $5,0 \cdot 10^{-2}$ | 4,5 | 0,22 |
| duration of blow-down | $-1,7 \cdot 10^{-3}$ | 13 | 0,02 |
| heat- transfer-coefficient | $-1,8 \cdot 10^{-3}$ | - | - |
| thickness of the varnish layer | $5,0 \cdot 10^{-3}$ | 30 | 0,15 |
| heat conductivity of varnish | $-6,1 \cdot 10^{-4}$ | 20 | 0,01 |
| heat capacity of varnish | $-1,5 \cdot 10^{-5}$ | 16 | 0,00 |
| heat conductivity of concrete | $-2,9 \cdot 10^{-4}$ | 15 | 0,00 |
| heat capacity of concrete | $-2,7 \cdot 10^{-4}$ | 22 | 0,01 |
| heat conductivity of steel | $-6,7 \cdot 10^{-5}$ | 10 | 0,00 |
| heat capacity of steel | $-1,1 \cdot 10^{-3}$ | 3 | 0,00 |

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| Parameter | $a_i$ [bar/%] | $\sigma_{x_i}$ [%] | $\Delta p_i$ [bar] |
|---|---|---|---|
| surface of heat-absorbing structures | $-3,9\cdot10^{-3}$ | 6,2 | 0,02 |
| thickness of steel walls | $-2,0\cdot10^{-2}$ bar/mm | 0,5 mm | 0,01 |
| initial temperature in the containment | $7,4\cdot10^{-4}$ | 5 | 0,00 |
| initial humidity of the air in the containment | $-6,5\cdot10^{-5}$ | 20 | 0,00 |
| gas constant, heat capacity of air | - | - | 0,00 |
| gas constant, heat capacity of steam | - | - | 0,00 |
| mathematical model | $3,7\cdot10^{-2}$ | 2,5 | 0,09 |

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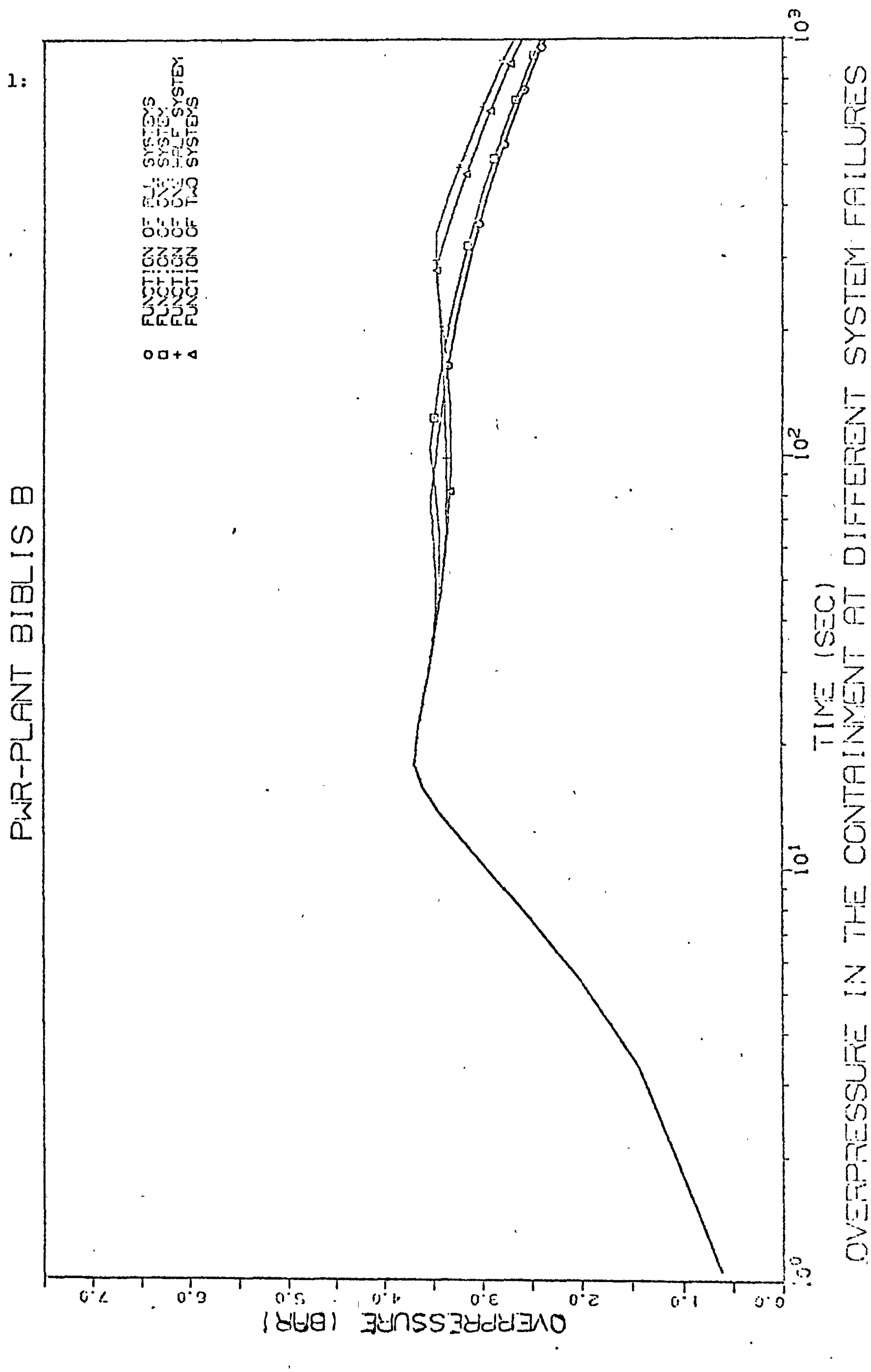
Fig. 1:




OVERPRESSURE IN THE CONTAINMENT AT DIFFERENT SYSTEM FAILURES

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| Berichtszeitraum/Period 1.1. - 31.12.1977 | Klassifikation/Classification 14 | Kennzeichen/Project Number RS 217 |
|---|---|---|
| Vorhaben/Project Title Risikostudie zur Sicherheitsbeurteilung von Kernkraftwerken mit Druckwasserreaktoren für einen Standort in der Bundesrepublik Deutschland<br><br>An Assessment of Risk for Nuclear Power Plants PWR on German Siting Condition | | Land/Country FRG<br><br>Fördernde Institution/Sponsor BMFT<br><br>Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH Glockengasse 2<br>5000 Köln 1 |
| Arbeitsbeginn/Initiated 1.6.1976 | Arbeitsende/Completed 31.12.1978 | Leiter des Vorhabens/Project Leader Prof.Birkhofer/Dr. Heuser |
| Stand der Arbeiten/Status continuing | Berichtsdatum/Last Updating December 1977 | Bewilligte Mittel/Funds DM 7.157.300,-- |

## 1. General Aim

The principal objective of the study is an overall risk analysis for nuclear power plants with reference to German siting conditions. As has been done in the American Reactor Safety Study WASH 1400, /1/, a quantitative assessment of risks will be made, which are due to potential accidents in nuclear power plants. Especially the KWU-typed PWR Biblis B is taken as the reference plant for the study.

## 2. Particular Objectives

Work on the study is divided into two temporally subsequent phases:

Phase A is closely related to the methodic approach and modelling that has been outlined in WASH 1400. Referring to German system design and siting conditions it is one of the aims of the study to conclude some results for risk assessment comparable with those of WASH 1400.

Phase B will be concentrated on a refined analysis of special problems, mainly on weak points obtained as results of Phase A, or respectively on those paths of accidents having found as main contributors to overall risk.

## 3. Research Program

The study will be performed by the Gesellschaft für Reaktorsicherheit (GRS) as the main contractor in cooperation with other coworking organizations, such as the Kernforschungszentrum Karlsruhe (KFK), the Gesellschaft für Strahlen- und Umweltforschung (GSF) and others. The following table gives a survey on the research program and its main subtasks:

3.1 Coordination

3.2 Accident Initiating Events

3.3 Loss of Coolant Accidents

3.4 Transients

3.5 Component Failure Behavior and Failure Data

3.6 Operating Experience from Nuclear Plants

3.7 Containment and Fission Product Release

3.8 Core Meltdown

3.9 External Events

3.10 Radioactive Sources (outside the Core)

3.11 Dispersion and Consequences

3.12 Risk Assessment

## 4. Experimental Facilities, Computer Codes

There is no need for special experimental facilities within the work, referring however to the various subtasks of the study some computer codes must be available for numerical calculations

- Reliability analysis calculations required for system analysis are performed with the GRS programs CRESSEX and FESIVAR.
- Calculations for description of core meltdown and subsequent fission product release from core and containment are performed with the computer codes BOIL and CORRAL used in WASH 1400, /1/.
- Calculations for long-time containment response following a core meltdown accident, specially with respect to pressure and temperature within the containment, are carried out with the GRS-program CONDRU.
- Because of differences in German siting conditions non-comparable to those ones of WASH 1400 and a higher population density within the Federal Republic of Germany an own consequence model UFO has been developed by KFK and GSF,/3/.

## 5. Progress to Date

Within the project in 1977 work has been performed on all subtasks of the program.

### To 3.2 Accident Initiating Events

Definition of potential accident initiating events mainly with respect to loss of coolant accidents, transient events and events due to external causes.

### To 3.3 Loss of Coolant Accidents

Completion of event tree and adjoint system fault tree analysis for different sizes of LOCAs according to the requirements of licensing procedure.

### To 3.4 Transients

Completion of a first systematic approach (Phase A) for an event tree analysis of transients, further work on the detailed event tree and system fault tree analysis for the accident initiating event "Loss of Offsite Power".

To 3.5 Component Failure Behavior and Failure Data

Review of data sources used in WASH 1400, completion of this list by data from other sources and from own field experience /4/, this work has been done partially in cooperation with the Technische Universität Berlin.

Evaluation of failure data from the VdTÜV-Statistics for non-nuclear pressure vessels and steam drums, performed in cooperation of the Rheinisch-Westfälische TÜV and the TÜV Rheinland.

To 3.6 Operating Experience from Nuclear Plants

In order to check the event trees and fault trees, especially with respect to the input failure data used for system analysis, current operating experience of German nuclear power plants is examined.

To 3.7 Containment and Fission Product Release
3.8 Core Meltdown

Subsequent to core meltdown calculations complementary calculations for containment pressure response have been carried out with the GRS-program system CONDRU. Furtheron calculations for potential radioactive release related to various modes of containment failure (especially containment leakage and overpressurization) have been performed with the WASH 1400 code CORRAL.

To 3.9 External Events

Examination of design adequacy of the reference plant with respect to various external events (e.g. air craft impacts, earthquake, floods etc.), estimation of probabilities associated with various external events and first analysis on potential secondary failures.

To 3.11 Dispersion and Consequences

An accident consequence model has been developed for the purpose of the study. This model essentially includes

- atmospheric diffusion and deposition of radioactive release from the reactor into the atmosphere,
- irradiation exposure and dosimetric modeling of health consequences,
- determination of consequences for various health effects (acute fatalities, late somatic and genetic effects) taking into account countermeasures such as evacuation and relocation,
- determination of risk.

6. Results

Main results obtained during the reporting period have been presented on occasion of the German Reaktortagung 1977, /2/, and the GRS-Fachgespräch 1977, /3/.

## 7. Next steps

Phase A of the study is planned to be finished in the second half year of 1978. For this purpose future work on some of the subtasks will be commented briefly.

### To 3.3 Loss of Coolant Accidents
### 3.4 Transients

Subsequent to accident event tree and system fault tree analysis reliability calculations taking into account the uncertainties of component failure data will be performed on the basis of reviewed data sources.

### To 3.6 Operating Experience from Nuclear Plants

Work continued for further examination of operating experience from German nuclear plants, especially Biblis A and B.

### To 3.7 Containment and Fission Product Release
### 3.8 Core Meltdown

Analysis on potential containment failure response due to an energy release from a steam explosion after core melt (assumed to occur in the lower plenum of the reactor pressure vessel).

Compilation of the release categories depending on the results of core melt and containment response analysis.

### To 3.9 External Events

Analysis on potential impacts due to earthquake events.

### To 3.11 Dispersion and Consequences

Numerical calculations for the whole spectrum of accidental fission product release from the containment will be carried out within the framework of the consequence model developed for Phase A of the study, estimates for population risks are given with respect to acute health, late somatic and genetic effects.

## 8. Relation with other Projects

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## 9. References

/1/ Reactor Safety Study
An Assessment of Accidental Risk in US Commercial Nuclear Plants
US Nuclear Regulatory Commission WASH 1400 (NUREG 75/014), October 1975

/2/ A. Birkhofer, F.W. Heuser, K. Köberlein
Zielsetzung und Stand der deutschen Risikostudie
Atomwirtschaft/Atomtechnik, Jahrgang 22, Nr. 6
Juni 1977, S. 331

/3/ GRS-Fachgespräch "Kernenergie und Risiko",
München, November 1977, GRS-10, März 1978

/4/ P. Hömke, H. Krause
Der Modellfall IRS-RWE zur Ermittlung von Zuverlässigkeits-
kenngrößen im praktischen Betrieb,
IRS-W-16, November 1975

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 1. - 31. 12. 1977 | 14 | RS 228 |

| Vorhaben/Project Title | |
|---|---|
| Unsicherheiten der Ausfallraten von Komponenten und der Eintrittswahrscheinlichkeit störfallauslösender Ereignisse<br><br>Uncertainty of the Failure Rates of Components and the Probability of Occurrences of Failure Causing Events | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Technische Universität Berlin<br>Institut für Kerntechnik |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1. August 1976 | 31. Juli 1978 | Prof. Dr. G. Memmert |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | 31. Dezember 1977 | DM 105.850,-- |

## 1. General Aim

The purpose of this project is the determination of the incertainty of the failure rates of components and the probability of occurences of failure causing events.

## 2. Particular Objectives

The reliability data for individual components necessary for the system analysis, particularly failure rates respectively lifetimes as well as their probability distributions have hitherto not been available with sufficient exactitude and specification. Therefore it ist necessary to compile the general accessible references and to compare them with one another. In doing this, it is important to give special consideration to data from German nuclear power plants. Furthermore it seems reasonable to make final decision on which type of distribution function is to be used on the basis of actual fault-trees.

## 3. Research Program

3.1 Compilation of reliability data for components.

3.2 Investigation of mean values and distributions of the reliability data.

3.3 Comparison of the values with those quoted in WASH-1400 /1/.

3.4 Critical assessment of the data.

3.5 Comparison of different distribution functions for describing the data.

## 4. Experimental Facilities, Computer Codes

ad 3.2 Development of a computer code to calculate the necessary statistic parameters of different distribution functions for the data which are available.

## 5. Progress to date

ad 3.2 Characteristic values of different distribution functions were calculated for the available data of failure rates of technical components.

ad 3.3 Original publications cited in WASH-1400 /1/ were reviewed. Data quoted in these publications were compared with the statements made in WASH-1400 /1/.

ad 3.4 Fault-trees were calculated to investigate the influence of uncertainities of input data on the results.

ad 3.5 Distribution functions were examined in order to find those which fit best the reliability data.

## 6. Results

ad 3.2 Statements in different publications were used to compile failure rates and dispersion for technical components.

ad 3.3 Several publications cited in WASH-1400 /1/ are based on the same data and are therefore connected with each other.

ad 3.4 Fault trees of technical systems, which were calculated using lognormal distributed values of reliability data, corresponded well, in most cases, with analogous calculations based on uniexponential distributed values. This may not be generalized, especially for larger dispersion of reliability data.

ad 3.5 The lognormal distribution as well as the uniexponential distribution seem to be most suitable to describe the available data. They were most seldom refused by the used statistical tests.

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## 7. Next Steps

Final Report

## 8. Relation with other Projects

## 9. References

/1/ Reactor Safety Study,
An Assessment of Accidental Risk in US Commercial Nuclear Power Plants, US Nuclear Regulatory Commission, WASH-1400 (NUREG 75/014), October 1975

## 10. Degree of the Availability of the Reports

/1/ free

1449

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1. - 31.12.1977 | 14 | RS 270 |

| Vorhaben/Project Title | |
|---|---|
| Zuverlässigkeitserhöhende Untersuchungen an Meß- und Regelgeräten der Kernkraftwerkstechnik<br><br>Investigations to improve the reliability of measurement and control devices for nuclear power station technology | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: INTERATOM Bergisch Gladbach 1<br>OE 8220 |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 01.07.1977 | 31.12.1981 | H. Banasch (Koord.) |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | 31.12.77 | 1.430.294,-- DM |

1. General Aim

The aim is to improve the reliability of measuring and control devices by applying aging procedures still to be defined.

The suitability of applying effective aging methods shall be proved by means of a comparative analysis between preaged and nonaged devices.

2. Particular Objectives

Elaborating optima aging methods by testing different aging procedures.

Investigating the failures of the devices and verifying the change of their technical data.

Compiling the reliability data of the electronic construction elements, of the construction groups and of complete measuring and control devices used for the investigations in order to calculate the reliability and comparing them with the data determined experimentally.

3. Research Program

3.1 Elaboration of aging specifications taking into account the parameters: time, temperature, acceleration as well as overvoltage and undervoltage.

3.2 Application of selected aging methods, optimization of these applied procedures and definition of aging specifications. Testing the aging procedures in a life test running for approximately 3 years under normal conditions.

3.3 Calculating the reliability of the devices used for the investigations.
Compiling the reliability data of each individual construction element provided in the devices being investigated, taking into account the manufacturer's data. Calculating the failure probability of the devices included in the test, using the preceding data. Comparison of the failure data determined experimentally with the failure probabilities determined theoretically.

4. Experimental Facilities, Computer Codes

Applying a process-computer-controlled test system and various problem-oriented test programmes as well as temperature chambers and a servo-hydraulic vibration table.

5. Progress to Date

To 3.2 The material procurement for manufacturing the necessary additional specimens has been nearly completed. However, there are delays in the delivery of the construction elements required for manufacturing the burst disks' monitoring electronics, so that the manufacturing of this type of device can only be completed in February 1978.
One type of device (constant-current source of INTERATOM) could already be completed.

The material procurement for the test setup for the aging and for the life test of the devices could be completed. The order on the wiring of the test cabinets has been placed to the subcontractors. Thus, the design of the test setup has been nearly concluded. The necessary adaption to the available automatic test system has been initiated.

6. Results

7. Next Steps

To 3.1 Continuation of the work for selecting suitable aging methods.

To 3.2 Continuation of the work for preparing the devices to be

investigated.
Continuation of the mounting work for the test setup and commissioning.

To 3.3 Elaborating the test programme for the long-time test.

8. Relation with Other Projects

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9. References

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10. Degree of Availability of the Reports

—

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| Berichtszeitraum/Period<br>1.7.77 - 31.12.77 | Klassifikation/Classification<br>14 | Kennzeichen/Project Number<br>RS 264 |
|---|---|---|
| Vorhaben/Project Title<br>Collection and Evaluation of Reliability Data at the Nuclear Power Plant Biblis B<br><br>Zuverlässigkeitskenngrößenermittlung im Kernkraftwerk Biblis B | | Land/Country<br>FRG<br><br>Fördernde Institution/Sponsor<br>BMFT<br><br>Auftragnehmer/Contractor<br>Gesellschaft für Reaktorsicherheit mbH<br><br>Glockengasse 2<br>5000 Köln 1 |
| Arbeitsbeginn/Initiated<br>1.7.77 | Arbeitsende/Completed<br>31.12.80 | Leiter des Vorhabens/Project Leader<br>Hömke |
| Stand der Arbeiten/Status<br>continuing | Berichtsdatum/Last Updating<br>December 1977 | Bewilligte Mittel/Funds<br>DM 2.928.230,71 |

1. General Aim

The failure behavior of components is influenced by various parameters, as construction data, operational data and operational stress. By analysing the data collected in the above mentioned data collection it is intended to estimate reliability data which are specified with respect to the most important determining parameter. Where possible it is intended to calculate distributions of reliability data and confidence intervals.

2. Particular Objectives

2.1 Development of the Basic Structure and Methods for the Data Collection

A pilot data collection was carried out in a liquite power plant with this experience a new data base structure and new data sheets have to be developed.

2.2 Collection of Basic Data

Before starting any evaluation of reliability data it is necessary to collect all basic data of interest.

2.3 Setting up of a Data Bank System

For the handling, the update of data and the evaluation of results it was found necessary to set up a data bank system.

2.4 Evaluation of Reliability Data

It is intended to make three steps in the evaluation of data to get the first results of the data collection as soon as possible.

3. Research Program

3.1 Preparation of the Data Collection

The collection is based on a data collection hand book, which discribes the sheets and the method of collection. All the coding and the structure of the data base has to be developed.

3.2 Collection of Basic and Failure Data

The basic data, as system and component data, will be collected for up to 15000 components. The failure data are based on the maintenance sheets, which are all collected and completed.

3.3 Installation of a Data Bank System

The data base is managed and updated with a data bank system. A data bank has to be structured and set up.

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3.4 Input of Data into the Data Store

The collected data will be punched and formal checks are applied. After the storing in the data bank there is a logical check of the data.

3.5 Evaluation of Reliability Data and Documentation of the Results

After the collection of the basic data when enough failures are available, an evaluation of reliability data is planned.

4. Experimental Facilities Computer Codes

The data bank system "System 2000" from MRI is used for storage of the data.

5. Progress to Date

to 3.1 Preparation of the Data Collection

The part of the hand book dealing with the collection of basic data and the according sheets are nearly completed. The sheets are partly printed.

to 3.2 Collection of Basic and Failure Data

The collection of basic data has been started.

to 3.3 Installation of a Data Bank System

A number of data bank systems was investigated in order to select the most appropriate with respect to the different requirements. "System 2000" was found to fit best.

6. Results

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7. Next steps

to 3.1 Preparation of the Data Collection

It is planned, to complete the hand book and to print the missing sheets.

to 3.2 Collection of Basic and Failure Data

The work of data collection is continued.

to 3.3 Installation of a Data Bank System

In February the training courses for the "System 2000" are carried out. The work to implement the system will then start.

8. Relation with other Projects

The project is a successor of the project "Modellfall IRS - RWE - zur Ermittlung von Zuverlässigkeitskenngrößen".

9. References

P. Hömke, H. Krause

Der Modellfall IRS - RWE - Zur Ermittlung von Zuverlässigkeitskenngrößen

im praktischen Betrieb -
IRS-W-16 (November 1975)

B. Wohak, G. Meinlschmidt

Ein Informationssystem zur Gewinnung von Zuverlässigkeitsdaten für Kraftwerkskomponenten

MRR 159 (Juni 1976)

10. Degree of Availability of the Reports

The reports are distributed by Gesellschaft für Reaktorsicherheit mbH, Glockengasse 2, 5000 Köln 1.

Classification: 14

| Title: Vurdering af systemers pålidelighed | Country: DENMARK |
|---|---|
| | Sponsor: Risø National Laboratory |
| Title: Evaluation of the Reliability of Systems | Organization: Risø National Laboratory |
| Initiated date: 1970 / Completed date: | Scientists: Hans Erik Kongsø |
| Status: In progress / Last updating: Currently | |

1. General aim

To develop methods for evaluation of the reliability of systems as an aid in design and safety analysis.

2. Particular objectives

Development of computer codes, primarily based upon the simulation method.

3. Experimental facilities

Not applicable.

4. Project status

4.1 Progress to date

For the purpose of analysing the reliability of systems with a high degree of complexity in design or operation various Monte Carlo programs were developed (ref. 1). The most recent of these programs is MOCARE. The modelling in this program is particularly flexible since all conditions for the occurrence of failures can be specified by means of subsystems. The program will be documented shortly.

4.2 Essential results

The MOCARE program has been tested extensively for instance on a power supply system of a nuclear power station, with very complex operational requirements.

## 5. Next steps

The MOCARE program will be further developed to include options for application of variance reduction techniques as part of a Ph.D. thesis project (see below).

## 6. Relation with other projects

The project has close relations to the Risø projects concerning the FAUNET program, the Ph.D. thesis on Optimization of Reliability Techniques and on Structural Reliability.

## 7. Reference documents

1. REDIS. A Computer Program for System Reliability Analysis by Direct Simulation. H.E. Kongsø. IAEA-SM-195/17.

## 8. Availability

The project information is freely available apart from cases where commercial interests may be violated.

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| Classification: 14 | |
|---|---|
| Title: | Country: DENMARK |
| | Sponsor: Risø National Laboratory |
| Title: Faunet - a program package for fault tree and network calculations. | Organization: Risø National Laboratory |
| Initiated date: 1976 Completed date: Status: continuing Last updated April 1978 | Scientists: O. Platz J.V. Olsen |

1. General aim

To develop a versatile program package for calculation of reliability and availability for systems represented by fault trees or networks.

2. Particular objective

To develop a modularization technique in order to make it possible to perform fault tree analysis on a minicomputer.

3. Experimental facilities and programmes

4. Project status

1. Progress to date Two versions of the package working, one on a 16 K PDP8 and another on a Burroughs B 6700.

2. Essential Results The package has been extensively tested on fault trees and networks taken from the literature.

5. Next Steps

6. Relation with other projects

## 7. Reference Documents

1. O. Platz and J.V. Olsen, FAUNET: A Program Package for Evaluation of Fault Trees and Networks. Risø Report no. 348, September 1976.

2. O. Platz and J.V. Olsen, FAUNET: A Program Package for Fault Tree and Network Calculations. In Proceedings of ANS Topical Meeting on Probabilistic Analysis of Nuclear Reactor Safety, May 8-10 1978, Los Angeles, California.

## 8. Degree of availability

1461

Classification: 14

| | |
|---|---|
| Title: Optimering af pålidelighedstekniske metoder | Country: DENMARK |
| | Sponsor: Risø National Laboratory |
| Title: Optimization of Reliability Techniques | Organization: Risø National Laboratory |
| Initiated date: 1977 Completed date: <br> Status: In progress | Scientists: Kurt Erling Petersen |

1. General Aim

To optimize Monte Carlo methods as well as numerical methods in analysis of reliability of structures and systems.

2. Particular objectives

Development of computer codes based on numerical integration in several variables and Monte Carlo methods with variance-reduction techniques.

3. Experimental facilities

Not applicable.

4. Project status

The project is a part of a Ph.D. thesis and was started in October 1977.

Until now work has concentrated on generators of random numbers and numbers from specified distribution functions.

5. Next steps

Work concerning Monte Carlo methods will concentrate on different variance-reduction techniques. The aim is to optimize the methods, i.e. to minimize the number of simulation trials when a specified accuracy is given.

Work concerning numerical methods will concentrate on integration of functions in several variables. The aim is to minimize the number of evaluations of the function when a specified accuracy is given.

## 6. Relation with other projects

The project has close relations to the Risø-project concerning evaluation of the reliability of systems (classification 14) and the Risø-project concerning reliability of structures (classification 14).

## 7. Reference documents

## 8. Availability

The project information is freely available.

1463

Classification: 14

| | |
|---|---|
| Title: | Country: DENMARK |
| | Sponsor: Risø National Laboratory |
| Title: RIKKE - a program system for automatic fault tree construction | Organization: Risø National Laboratory |
| Initiated date: 1977 Completed date: <br> Status: continuing Last updated May 1978 | Scientists: J.R. Taylor |

1. General aim

Automatic or semi automatic construction of fault trees, cause consequence diagrams, and process plant simulations.

2. Particular objective

To be able to perform routine failure analyses of process plants, and construct simulation models interactively, directly from a flow sheet of the plant, the transformation to mathematical models being carried out by computer.

3. Experimental facilities and programmes

4. Project status

1. Progress to date First version of program now working on DEC PDP 11 and Burroughs B 6700 computers.

2. Future work Practical use of the program .

5. Next Steps

6. Relation with other projects

## 7. Reference Documents

1. Experience with Algorithms for Automatic Failure Analysis. J.R. Taylor, E. Hollo.

2. Nuclear Systems Reliability Engineering and Risk Assessment. SIAM 1977.

## 8. Degree of availability

1465

Classification 14

| Title 1 | COUNTRY Denmark |
|---|---|
| Strukturel pålidelighed | SPONSOR Risø National Laboratory |
| | ORGANIZATION Risø National Laboratory |
| Title 2<br>Structural Reliability | Project leader:<br>P.E. Becher/I. Misfeldt |
| Initiated: 1970 Completed:<br>Status: In progress Last updating: | Scientists:<br>P.E. Becher<br>H.E. Kongsø<br>I. Misfeldt<br>S. Weber |

1. General aim: To develop methods for evaluation of the reliability of structural components.

2. Particular objectives: Development of computer codes, based on probabilistic methods, for evaluation of the reliability of primary components in light water reactors, specifically the steel pressure vessel and the fuel element cladding.

3. Experimental facilities:

4. Project status

As a supplement to the computer code PEP 706 for calculation (by Monte Carlo with Importance Sampling) of the failure probability of a steel pressure vessel an analytical program ANPEP was developed. ANPEP makes a numerical integration of the failure integral by means of discreetizing of all the parameter in the failure criteria. ANPEP has proved to be much faster and easy to work with than PEP 706 and has even been able to take into account correlated variables without making exessiv demands on computer memory or time.

The computercode PFM 690 for calculation (by Monte Carlo) of the statistical crack growth based on Paris's formula has proved the mathematical unstability of this formula when using the most recent experimental data on crack growth caracteristics.

A computer program, FRP, has been developed for the statistical analysis of the fuel performance. The statistical methods employed are either Monte Carlo simulations or a Taylor approximation. The program utilizes a deterministic fuel performance code and a stress corrosion failure criterion, verified against experimental data. The code has been applied to the analysis of irradiation experiments.

5. Next steps:

Steel pressure vessel:

Development of statistical models for timedependent phenomenacs like

a) Crack growth
b) Inservice Inspection
c) Neutron Embrittlement
d) Updating of distribution functions for appropriate parameters when the components in question has surviveded a major transient

Furthermore other areas/other components (steel) and other failure modes/failure criterias should be included.

Fuel element cladding:

a) Verification of the deterministic and statistical models
b) Collection of statistical data for the important material and design variables.

6. Relation with other projects:

Closely related with system reliability

7. Reference documents:

Risø-M-1650: Application of statistical linear elastic fracture mechanics to pressure vessel reliability analysis, September 1973. P.E. Becher, Arne Pedersen.

Risø-M-1918: ANPEP/V2, A computer program for calculation of the probability of failure of structures, March 1977, H.E. Kongsø, K.E. Petersen.

Risø-M-1927: Probabilistic Approach to Reliability predictions for LWR fuel rods, March 1977, I. Misfeldt.

Risø-M-1928: Performance of the fuel model FFRS, March 1977, I. Misfeldt.

The Reliability of Nuclear Fuel. Proceedings of the ANS Topical Meeting on Probabilistic Analysis of Nuclear Reactor Safety, I. Misfeldt.

8. Availability The project information is freely available apart from cases, where commercial interests may be violated.

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1468

Classification $\boxed{6}$ and 14

| | |
|---|---|
| Title 1<br>Fejlanalyse af tekniske Systemer og vurdering af deres pålidelighed | COUNTRY Denmark<br>SPONSOR Risø National Laboratory<br>ORGANIZATION Risø National Laboratory |
| Title 2<br>Failure analysis of technical systems and evaluation of their reliability | Project leader<br>D. Nielsen |
| Initiated 1970 Completed<br>Status: In progress Last updating currently | Scientists<br>J.R. Taylor<br>O. Platz<br>D. Nielsen |

| | | |
|---|---|---|
| 144-1 -02/4105-20<br>154-1 -02 | | |
| **Titre**<br>Développement du programme PATREC pour le calcul de la fiabilité de systèmes complexes. | | **Pays**<br>FRANCE<br>**Organisme directeur**<br>CEA/DSN |
| **Titre (anglais)**<br>Development of the PATREC code for calculation of complex systems reliability. | | **Organisme exécuteur**<br>CEA/DSN - SETSSR<br>**Responsable**<br>DSN - SETS / Fontenay |
| **Date de démarrage** 01/04/72 | **Etat actuel** terminée | **Scientifiques** |
| **Date prévue d'achèvement** 31/12/77 | **Dernière mise à jour** 01/12/77 | |

1 - Objectif général :

Calcul par arbre de défaillance de la fiabilité ou de la disponibilité d'un système complexe.
Les méthodes de base sont : L'algèbre booléenne et la reconnaissance des formes élémentaires.

2 - Objectif particuliers :

1) Optimisation avec l'aide du SERMA de la version PATREC-MC permettant de calculer la sensibilité du résultat aux variations des paramètres d'entrée.
2) Optimisation de la version PATREC RCM.
3) Introduction de modifications dans le programme en vue de pouvoir traiter directement les modes communs.

4 - Etat de l'étude :

Avancement à ce jour :

Améliorations successives apportées aux versions PATREC-MC et PATREC-RCM.

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5 - Prochaines étapes :

- Améliorations(avec l'aide du SERMA) de la version PATREC-MC, permettant de calculer la sensibilité du résultat aux variations des paramètres d'entrée.
- Optimisation de la version PATREC-RCM.
- Traitement des modes communs.

6 - Relations avec l'autres études :

Voir fiche : " Utilisation des techniques de MONTE-CARLO pour les calculs de fiabilité ".

7 - Documents de référence :

" Méthodes nouvelles pour l'évaluation de la fiabilité : reconnaissance des formes ", B.V.KOEN Rapport CEA R 4368, Février 1972.

" Programme de calcul de la fiabilité de systèmes complexes "PATREC-DE" par la reconnaissance des formes", A.BLIN, A. CARNINO, G. JUBAULT Rapport DSN 46, Septembre 1974.

" Notice de présentation et d'utilisation du programme PATREC-DE", A. BLIN , Janvier 1976.

" Le code PATREC-MC", Rapport SERMA 263, Mars 1976.

" The state of the art of PATREC,a computer code for the evaluation of reliability and availability of complex systems ", A.BLIN, A.CARNINO, etc... National Reliability Conference Nottingham 1977.

8 - Disponibilité des documents

Disponibles

| | |
|---|---|
| **Titre**<br>Problème des évènements rares dans l'analyse de fiabilité des centrales nucléaires de puissance PWR . | **Pays**<br>FRANCE |
| | **Organisme directeur**<br>CEA/DSN<br>EDF, FRAMATOME |
| **Titre (anglais)**<br>Problem of rare events in the reliability analysis of nuclear power plants (PWR) | **Organisme exécuteur**<br>CEA/DSN - SETSSR<br>EDF FRAMATOME |
| | **Responsable**<br>DSN/SETS / Fontenay |
| **Date de démarrage** 1/77 — **Etat actuel** en cours | **Scientifiques** |
| **Date prévue d'achèvement** 11/78 — **Dernière mise à jour** 1/78 | |

## I - Objectif général :

Etude générale OCDE/CSIN sur le problème des évènements rares dans l'analyse de fiabilité des centrales nucléaires de puissance. Elle a pour but de faire progresser la connaissance des problèmes soulevés par les évènements rares dans l'analyse de fiabilité des Centrales Nucléaires.

Dans ce but, six groupes de travail ont été constitués :

- Analyse et quantification des erreurs humaines.
- Analyse des défaillances de mode commun.
- Collecte et analyse des données.
- Théorie de la décision et études statistiques.
- Techniques de communication interdisciplinaire.
- Evaluation de la fiabilité d'un système réel.

La France (CEA, EdF, Université, FRAMATOME) qui participe aux travaux des différents groupes internationaux, a la responsabilité d'animer le dernier de ces groupes "Evaluation de la fiabilité d'un système réel ", groupe constitué uniquement d'experts français.

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2 - Objectifs particuliers :

Le système choisi est le système d'arrêt d'urgence de Fessenheim ; ce système étant sollicité à la suite d'un retrait incontrôlé des grappes de contrôle.

4 - Etat de l'étude :

1) Avancement à ce jour :

Cette étude a mis en évidence les possibilités et les difficultés d'effectuer un travail valable dans ce domaine. Elle a fait apparaître que des incertitudes existaient au niveau des données de fiabilité. Le problème des modes communs et de la fiabilité humaine reste pratiquement entier.

2) Résultats essentiels :

EdF a étudié la partie analogique du système (depuis les capteurs jusqu'aux déclencheurs à seuil). Le DSN a étudié la partie logique (jusqu'aux disjoncteurs de maintien des grappes) du système et s'est chargé des calculs.

Les informations regroupées EdF/DSN, ont permis à ce dernier d'établir l'arbre de défaillance du système. Pour ce faire, on a utilisé le code CHAMBOR.

L'étude quantitative a été réalisée grâce au code PATREC-RCM. Elle a mis en évidence l'importance des disjoncteurs de maintien des grappes dans le résultat final, ainsi, comme on pouvait s'y attendre, que l'importance des défaillances de mode commun.

Les valeurs retenues et les résultats sont consignés dans les deux tableaux suivants:

| Dispositif | Taux de défaillance ($\lambda$) ou refus à la demande (d) |
|---|---|
| Voie $\Delta T$ température | $\lambda = 4.10^{-5}/h$ |
| Voie puissance neutronique | $\lambda = 4,8.10^{-5}/h$ |
| Disjoncteur | $d = 3.10^{-5}/d$ |
| Relais | $\lambda = 1,4.10^{-7}/h$ |

.../

| $\beta$ | MCT+MCP | MCR | MCD | Probabilité résultante |
|---|---|---|---|---|
| 0 | 0 | 0 | 0 | $3,6.10^{-7}$ |
| $10^{-2}$ | $4,4.10^{-7}$ | $1.10^{-6}$ | $3.10^{-7}$ | $2,1.10^{-6}$ |
| $10^{-1}$ | $1,6.10^{-5}$ | $1.10^{-5}$ | $3.10^{-6}$ | $3.10^{-5}$ |
| 1 | $1.10^{-3}$ | $1.10^{-4}$ | $3.10^{-5}$ | $1,1.10^{-3}$ |

$$\beta = \frac{\lambda\text{(ou probabilité) Mode Commun.}}{\lambda\text{(ou probabilité) Mode Propre.}}$$

MCT = Mode commun voie températures.
MCP = Mode commun flux neutronique.
MCR = Mode commun relais.
MCD = Mode commun disjoncteur.

Le tableau donne la probabilité résultante pour 4 valeurs relatives des modes communs par rapport aux modes propres.

Le résultat doit être interprété de la manière suivante :

Au temps $t_o$, on fait un test pour s'assurer du bon fonctionnement du système, au bout d'un temps T égal à 1 mois (720 heures) on fait un deuxième test ; les valeurs de la dernière colonne donnent la probabilité de trouver le système défaillant.

Par ailleurs, le groupe de travail "analyse et quantification des erreurs humaines a déjà fait un certain nombre de remarques fort intéressantes sur la manière dont sont effectués les tests en réel et les possibilités d'erreurs humaines liées à des situations gênantes.

## 5 - Prochaines étapes :

L'étude se poursuivra en 1978 par :

- Une analyse plus fine du système.
- L'étude des mécanismes des grappes.
- Une recherche de meilleures données et la reprise des calculs en vue de l'amélioration de l'estimation, en fonction des renseignements provenant d'autres groupes de travail.

| Titre | Pays |
|---|---|
| Probabilités d'accident : Evénements initiateurs d'accidents sur les PWR et séquences d'accidents. | FRANCE |
| | **Organisme directeur** CEA/DSN |
| **Titre (anglais)** Accident probabilities : Accident initiating events on PWR and accidents sequences. | **Organisme exécuteur** CEA/DSN-SETSSR |
| | **Responsable** DSN/SETS. /Fontenay |

| Date de démarrage | Etat actuel | Scientifiques |
|---|---|---|
| 09/76 | en cours | |
| **Date prévue d'achèvement** 12/81 | **Dernière mise à jour** 12/77 | |

1 - Objectif général :

Détermination des événements initiateurs d'accidents par perte des différentes barrières dues à des agressions. Construction des arbres d'événements à partir de ces initiateurs et détermination de séquences accidentelles.

- Etat de l'étude :

Avancement à ce jour :

Principe de la méthode.

5 - Prochaines étapes :

- Détermination des initiateurs.
- Essai de quantification de certaines séquences.

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7 - Documents de références disponibles :

"Détermination of initiating events and sequences of reactor accidents by a barrier analysis" . A. CARNINO, J. DUBAU .
International conference on Nuclear system, reliability and or assesment
Tenessee Juin 1977 .

| 144-1-07<br>154-1-07<br>164-1-02 | | 14<br>* 11.1 |
|---|---|---|
| Titre<br>Modèlisation du taux de défaillance de vannes en fonction de différents paramètres. | | Pays<br>FRANCE<br>Organisme directeur<br>CEA / DSN |
| Titre (anglais)<br>Modelisation of the failure rate of valves in terms of different parameters. | | Organisme exécuteur<br>CEA/DSN - SETSSR<br>Responsable<br>DSN/SETS- Fontenay |
| Date de démarrage 1/12/77 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 30/6/78 | Dernière mise à jour 1/78 | |

1 - Objectif général :

Il s'agit de trouver un modèle de fiabilité pour les vannes permettant de prendre en compte l'influence de différents paramètres.

2 - Objectifs particuliers :

1) A partir des données tirées du fichier de défaillances d'EDF à St. LAURENT-des-EAUX , on s'efforcera d'établir des corrélations entre les caractéristiques des vannes (type, taille, pression, fluide, etc...) et les taux de défaillance observés . Les méthodes de l'analyse statistique (analyse de la variance, etc...) seront utilisées.

2) Une analyse supplémentaire et approfondie d'une partie du fichier de ST. LAURENT des-EAUX sera nécessaire.

3 - Etat de l'étude :

Vient de démarrer.

| 144-1 -09/4105-20<br>154-1 -09/4105-20 | | 14<br>* 11.1 |
|---|---|---|
| Titre<br><br>Construction automatique d'arbres de défaillances à partir de diagrammes logiques. | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br><br>Automatic construction of fault trees from logic diagrams. | | Organisme exécuteur<br>CISI<br><br>Responsable<br>DSN/SETS Fontenay |
| Date de démarrage 1/6/77 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 31/12/79 | Dernière mise à jour 1/78 | |

1 - Objectif général :

L'objectif global de ce projet est la construction automatique des arbres de défaillances pour les analyses de fiabilité de systèmes à partir :

- d'arbres élémentaires stockés en bibliothèque.
- de diagrammes logiques.
- si possible de diagrammes de fonctionnement.

Trois points justifients ce projet :

1) La construction d'un arbre de fautes est complexe, fastidieuse et sujette à erreurs. La construction automatique libère l'ingénieur de cette tâche ingrate.

2) Le stockage en bibliothèque d'arbres élémentaires diminue considérablement les temps de calculs globaux car les calculs partiels ont été effectués une fois pour toutes.

3) Le problème du passage du diagramme de fonctionnement à l'arbre de fautes admet généralement plusieurs solutions. Le temps de calcul varie énormément avec la solution choisie ; l'étude entreprise permettra d'optimiser le choix de la solution.

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2 - Objectifs particuliers :

1) - Construction d'une bibliothèque.

- Les arbres élémentaires représentés en notation polonaise (endorder traverse).
- Les listes des structures.
- Les lois de probabilités.

Cette bibliothèque devra son efficacité (identification, temps d'accès) à l'utilisation du "traitement de liste". ("list processing").

2) - Développement de petits programmes auxiliaires pour augmenter, diminuer, modifier... etc, le contenu de la bibliothèque.

3) - Mise au point du stockage de la bibliothèque sur supports externes (bande, disque... etc).

4) - Compatibilité du code PATREC avec la partie de la bibliothèque relative aux arbres élémentaires.

5) - Compatibilité du code PATREC avec l'entrée "diagrammes logiques" de la bibliothèque.

6) - Si possible, compatibilité du code PATREC avec l'entrée "diagramme de fonctionnement " de la bibliothèque.

4 - Etat de l'étude :

Avancement et résultats obtenus :

Ce projet se déroule en deux étapes pour chaque objectif partiel : Développement d'une théorie informatique d'une part, écriture du programme d'autre part.

La première de ces deux étapes " développement d'une théorie informatique " est franchie par les objectifs partiels 1 à 5 ; la seconde "écriture d'un programme " est franchie pour 3 et en voie de l'être pour 1 et 2.

5 - Prochaines étapes :

Pour la partie théorique la prochaine étape est l'objectif 6. Pour la partie programmation, on travaillera principalement sur l'objectif 2.

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6- Relation avec d'autres études :

Les résultats de cette étude seront utilisés ultérieurement dans le programme PATREC

7- Docuements de référence : - rapports internes non disponibles

1483

| | | |
|---|---|---|
| 144-2 -01 / 4600-50<br>154-2 -01 | | 14<br>* I1.1 |
| Titre<br><br>Paramètres humains de la sûreté | | Pays<br>FRANCE<br><br>Organisme directeur<br>C.E.A/ DgCS |
| Titre (anglais)<br><br>Human factors in nuclear safety | | Organisme exécuteur<br>CEA/DSN - SETSSR<br><br>Responsable<br>DSN/SETS /Fontenay |
| Date de démarrage 1/01/78 | Etat actuel<br>en cours | Scientifiques |
| Date prévue d'achèvement 31/12/81 | Dernière mise à jour 1/78 | |

1 - Objectif général :

Lorsqu'on étudie la causalité élémentaire des accidents ou incidents nucléaires, on constate que les composantes d'origine humaines sont majoritaires. Il est donc capital de s'efforcer d'étudier scientifiquement l'organisation humaine, 3ème composante d'une installation nucléaire.

- Objectifs particuliers :

1) A partir du fichier d'incidents des réacteurs français (de 1972 à 1976) Recherche approfondie des erreurs humaines, avec évolution des facteurs de risque d'erreurs humaines, en fonction de l'âge de la centrale et leur relation avec le degré d'apprentissage des opérateurs.

2) Analyse fine de quelques erreurs humaines ayant entraîné des incidents et concernant les tests de contrôle.

3) Etude détaillée de la procédure de tests de contrôle du système d'arrêt d'urgence de Fessenheim.

4 - Etat de l'étude :

L'étude démarre .

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| 144-1 -04/4112-20<br>154-1 -04 | 14<br>* 11.1 |

| | |
|---|---|
| Titre<br><br>Utilisation des techniques de MONTE CARLO pour les calculs de fiabilité. | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br><br>The use of MONTE CARLO techniques for reliability calculations. | Organisme exécuteur<br>CEA/DRE - SERMA<br><br>Responsable |

| | | |
|---|---|---|
| Date de démarrage 01/01/77 | Etat actuel en cours | Scientifiques |
| Date prévue d'achèvement 31/12/78 | Dernière mise à jour 12/77 | |

1 - Objectif général :

Amélioration du code PATREC par les techniques de MONTE CARLO.

2 - Objectifs particuliers :

1) Les objectifs prévus sont la détermination des bornes de l'intervalle de confiance d'un résultat de calcul par arbre de défaillance compte tenu des intervalles de confiance des composants élémentaires. Diverses applications seront réalisées.
2) Une étude MONTE CARLO sera étendue à la résolution analytique d'un graphe de MARKOV à transitions non constantes.

4 - Etat de l'étude :

1) Avancement à ce jour

Une version de PATREC a été écrite pour le calcul de l'intervalle de confiance du résultat final d'un arbre à partir des intervalles affectés aux données. Un biaisage a été réalisé pour certains cas.

2) Résultats essentiels

Code de calcul PATREC-MC actuellement opérationnel.

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5 - Prochaines étapes :

Réagencement et reprogrammation de certaines parties de PATREC pour y inclure une vérification et une évaluation de modes communs de défaillances dans un arbre de défaillances. Résolution de graphes de MARKOV à transitions non constantes par MONTE CARLO.

6 - Relation avec d'autres études :

voir fiche " Développement du programme PATREC ".

7 - Documents de références disponibles

"The state of the art of PATREC : a computer code for the evaluation of reliability an availability of complex systems" - A. BLIN, A. CARNINO, etc ... National Reliability Conference . Nottingham 1977 .

1487

| 144-1 -03<br>154-1 -03 | | 14<br>* 11.1 |
|---|---|---|
| **Titre**<br>Etude de processus stochastiques<br>(processus de Markov non homogènes) | | **Pays**<br>FRANCE<br><br>**Organisme directeur**<br>CEA/DSN |
| **Titre (anglais)**<br>Study of stochastic processes<br>(non homogeneous Markov Processes) | | **Organisme exécuteur**<br>CEA/DSN - SETSSR<br><br>**Responsable**<br>DSN/SETS Fontenay |
| **Date de démarrage** 1/1/78 | **Etat actuel**<br>en cours | **Scientifiques** |
| **Date prévue d'achèvement** 31/12/79 | **Dernière mise à jour** 1.1.78 | |

1 - Objectif général :

Quand les taux de défaillance et/ou les taux de réparation ne sont pas constants (lois non exponentielles), le calcul de la probabilité de défaillance de systèmes complexes nécessite une méthodologie applicable aux processus de Markov.
La présente étude vise la mise au point d'une telle méthodologie.

2 - Objectifs particuliers :

1) Recherche d'une formule générale correspondant à un système de Markov non homogène ( taux de transition entre états du graphe non constants)
2) Préparation d'un algorithme en vue de l'écriture ultérieure d'un code de calcul permettant de résoudre certains problèmes rencontrés dans les évaluations de sûreté.

4 - Etat de l'étude :

L'étude démarre au 1/1/78

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5 - Prochaines étapes :

Recherche de la formule générale correspondant à l'objectif particulier 1) .

6 - Relation avec d'autres études :

Les résultats obtenus devront permettre de traiter certains problèmes d'évaluation de sûreté des systèmes .

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| 144-1 -01 /4105-20<br>154-1 -01 | 14<br>* 11.1 |

| | | |
|---|---|---|
| Titre<br>Disponibilité de systèmes en attente périodiquement testés.<br>Approche analytique. | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DSN |
| Titre (anglais)<br>Periodically tested standby systems availability<br>Analytical approach. | | Organisme exécuteur<br>CEA/DSN - SETSSR<br><br>Responsable<br>DSN-SETS - Fontenay |
| Date de démarrage 01/12/75 | Etat actuel<br>en suspens | Scientifiques |
| Date prévue d'achèvement 1978 | Dernière mise à jour 12/77 | |

1 - Objectif général :

- Calcul de la disponibilité instantanée et moyenne des systèmes en attente testés à intervalles réguliers.
- Optimisation de la disponibilité moyenne.

2 - Objectifs particuliers :

1) Elaboration d'une formule d'indisponibilité utilisable par le code PATREC.
2) Calcul de la disponibilité d'un système à 1 composant.
3) Calcul de la disponibilité d'un système complexe.
4) Prise en compte de paramètres divers.

.../

3 - Etat de l'étude :

1) Avancement à ce jour :

Les objectifs N° 1 et 2 sont réalisés, une méthode a été proposée pour aborder l'objectif N° 3, quand à l'objectif N° 4 il vise à l'amélioration du modèle mis au point dans les 3 premiers objectifs. C'est à dire qu'il n'a pas de limite définie a priori mais des résultats partiels ont été obtenus.

2) Résultats essentiels :

- Mise au point d'un modèle conduisant par un formalisme rigoureux à l'élaboration de formules analytiques pour :
  - le calcul de la disponibilité instantanée prévisionnelle d'un système considéré comme faisant un seul bloc.
  - le calcul de la disponibilité moyenne d'un tel système.
  - l'optimisation de l'intervalle entre test d'un tel système.
- Mise au point d'une méthodologie permettant grâce à un raisonnement "physique" basé sur des hypothèses d'approximation généralement vérifiées en pratique de faire les mêmes calculs que ci-dessus.
- Regroupement des 2 méthodes : les formules approchées trouvées directement par le raisonnement "physique" se retrouvent bien comme développements limités des formules rigoureuses.
- Mise au point de la théorie permettant de calculer par PATREC la disponibilité de système complexe en utilisant les formules ci-dessus comme des lois à entrer dans les feuilles des arbres de défaillance.
- Résolution analytique rigoureuse d'un système à 2 composants identiques en redondance parallèle et d'un système dont la durée du test ne peut plus être considérée comme négligeable.
- Résolution analytique rigoureuse d'un système dont l'efficacité du test n'est pas de 100 %.
- Mise au point de version provisoire des codes INDI-1 et INDIGO basés sur le modèle ci-dessus. Ces 2 codes ne sont que des études de faisabilité destinés à la vérification des formules mais non destinés à un usage extérieur dans leur éta actuel.

4 - Prochaines étapes :

Cette étude est à l'heure actuelle en suspens. Néanmoins les prochaines étapes devraient être :

- Mise au point de versions définitives des codes INDI-1 et INDIGO.
- Affinement du modèle pour la prise en compte du maximum de paramètres permettant de caractériser plus finement le système en attente de fonctionnement périodiquement testés.

Il faut considérer qu'une étude comme celle-ci est ouverte c'est à dire n'a pas de point d'achèvement bien déterminé.

.../

5 – Relation avec d'autres études :

- Relation avec l'étude du code PATREC

- Relation avec l'étude menée en collaboration avec l'IRIA .

7 – Documents de référence :

- Rapport DSN N°113 : Disponibilité de systèmes en attente périodiquement testés .
  JP SIGNORET – Septembre 1976

- Rapport DSN N°129 : Suite du rapport DSN N°113
  Calculs approchés – Optimisation
  JP SIGNORET – Décembre 1976

8 – Degré de diponibilité :

Libres .

| PROJECT TITLE : Fault analysis of the conventional island in a LWR nuclear power plant. | CLASSIFICATION 14 |
|---|---|
| SPONSORING COUNTRY : | ORGANISATION : FRANCO TOSI S.p.A. |
| DATE INITIATED : April 15, 1975<br>DATE COMPLETED : April 15, 1976 | PROJECT LEADER : Vittorio BEDOGNI |

Description :

1) Research program

- Fault tree definition of the conventional island of a LWR nuclear power plant
- Fault data collection of the system's components. Analysis and treatment of the fault data.
- Reliability and availability evaluation of the systems.
- Parametric analysis of the systems reliability varying the failure rate of the critical elements and of the components whose fault data are not available or not sufficiently reliable.

2) Facilities

- Computer and computer codes.

3) Reference documents

- R.E. Barlow, F. Proschan "Mathematical theory of reliability" John Wiley & Son.,Inc., New York
- A.G. Colombo "CADI, a computer code for system availability and reliability evaluation" - Report EUR 4940 e (1973)
- J.B. Fussel "A formal methodology for fault-tree construction" Nuclear Science and Engineering, 52 (1973), pp. 421-432

4) Related projects

None (F.Tosi)

5) The works is done in relation to the design of the ENEL V and VI nuclear power stations.

| PROJECT TITLE : Reactor Safety Studies via Noise Analysis | CLASSIFICATION 14 - 8 |
|---|---|
| SPONSORING COUNTRY : Italy | ORGANISATION : C.N.E.N. |
| DATE INITIATED : 1.1.1975<br>DATE COMPLETED : ........ (in progress) | PROJECT LEADER . N. Pacilio, LFCR |

Description :

The research program deals with applications of the theory of stochastic processes to reactor safety studies. They are directed to assessing the correct performance of in-core and ex-core instrumentation, to safety monitoring and early detection of abnormal operating conditions and/or malfunctions. (Preliminary research for early failure detection).

Efforts up to now are :

- set-up of a general theory for multi-detector reactor-noise analysis in ergodic conditions, non-equilibrium conditions and during pulsed experiments ;
- experimental determination of nuclear reactor kinetic parameters and control of multiplying assemblies that must be kept subcritical ;

The general theory will soon be expanded to include problems related to heat transfer ( fluctuations in temperature, pressure and void volumes ) and to a preliminary analysis of significance of acoustic noise analysis.

Facilities

- four light-water reactors ( ROSPO, RANA, RITMO, TRIGA ) and a copper-reflected highly-enriched fast reactor ( TAPIRO )

Reference list

- N.PACILIO, V.M. JORIO - Review of International Noise Conference held in Rome ( October 21-25, 1974) CNEN Report, RT/FI(74)47

| Title 1 (Original language) | Classification<br><br>14 |
|---|---|
| Title 2 (English)<br><br>Interactive Synthesis of Fault Trees for Safety Assessment of Nuclear Power Plants | Country ITALY<br>Sponsor CNR-EURATOM<br>Organisation<br>CESNEF-Politecnico Milano |
| Date initiated 1976<br>Date completed 1978<br>Last updating April 1977 | Project Leader<br><br>S. GARRIBBA |

DESCRIPTION

Each component from the system is the domain of local functions which express, for all normal and abnormal operating conditions, the flow variables (i.e. the variables characterizing transfers of fluid, load, heat, electrical current, etc.) or which express the constitutive variables (i.e. the variables describing rupture, damage, blockage, etc.). The consideration of a number of additional virtual domains also allows to make explicit external influences (e.g. human operators, maintenance, fires, missiles, etc.) in terms of variables. The local functions are interrelated through transfer matrices. Localized transfer matrices are pieced together to form the Approximate Descriptive Model (ADM) of the system. Then, steps towards the generation of an isomorphic fault tree are : i) Definition of the top event. ii) Numbering process. Numbers are assigned to certain points of flow sheets in a certain canonical manner. iii) Translation process. iv) Free reduction process. These steps will be implemented by computer program I-SYNFT (Interactive Synthesis of Fault Tree).

| Title 1 (Original language) | Classification |
|---|---|
| Sviluppo di una procedura avanzata per l'analisi di sicurezza di un PWR seguendo l'approccio probabilistico. | 14 |
| **Title 2 (English)** Development of an advanced procedure for the PWR safety analysis following the probabilist approach | **Country** ITALY<br>**Sponsor** FIAT-T.T.G.<br>**Organisation** Nuclear Energy Division |
| **Date initiated** June 1976<br>**Date completed** 1980<br>**Last updating** April 1977 | **Project Leader**<br>G.P. Pozzi |

1. General aim

Development of a procedure for the design of the engineered safety features of a PWR plant using the reliability techniques.
Preparation and linking of digital programs to perfom the safety analysis of a PWR plant through a synthesis of the accident analysis methods with the reliability probabilistic methods.

2. Particular objectives

Application of the above procedure and digital programs to the design of the experimental loop CLEOPATRA to be inserted in the ESSOR reactor.
Application of the same techniques to the analysis and "interiorization" of the engineered safety features of a PWR Westinghouse nuclear plant.

3. Experimental facilities

None

4. Project status

For each accident to be analyzed, a detailed "event tree" has been developed in order to establish: a) the different path the accident can follow; b) the features of the digital programs necessary to analyze the consequences of the accidents.
An analysis of the various reliability programs available has been performed.
With the available programs a probabilistic analysis of the LOCA in the CLEOPATRA loop was made.

5. Next steps

a) The digital programs necessary for the analysis of each accident path will be set up (deterministic programs)

| Title 1 (Original language) | Classification |
|---|---|
| Sviluppo di una procedura avanzata per l'analisi di sicurezza di un PWR seguendo l'approccio probabilistico. | 14 |

b) The results of the deterministic programs will be coupled with the results of the probabilistic programs.

c) The complete procedure will be used for the design of the PWR engineered safety features.

6. Relation to other projects

The digital programs for the LOCA analysis described in the other project ("Development of a chain of digital programs for the LOCA analysis of a PWR) will be utilized.

7. Reference documents

Not yet prepared

8. Degree of availability

To a limited extent

9. Budget, personnel involved

8 engineers for 5 years.

| Title 1 (Original language) | Classification<br>14 |
|---|---|
| Title 2 (English)<br>Fault-Tree Sensitivity Analysis for Reliability Calculations in Nuclear Power Plants | Country ITALY<br>Sponsor IAEA<br>Organisation<br>CESNEF-Politecnico Milano |
| Date initiated 1975<br>Date completed 1977<br>Last updating Sept. 1976 | Project Leader<br>S. Garribba |

## DESCRIPTION

The research project has the main scope of developing a method and a computer code for the sensitivity analysis of a system with respect to variations of input parameters for primary events. Topics to be covered by the research are as follows (i) problem definition with a particular concern for nuclear safety systems; (ii) sensitivity analysis with respect to primary events. The most suitable approximation for system availability computation and the most significant parameters to be perturbed will be investigated.
The study will imply distributions of failure and restoration of the types: exponential, normal, lognormal, gamma, Weibull.
(iii) sensitivity analysis with respect to the minimal cut sets. The main purpose of this part of the work is to weight each minimal cut set with respect to system availability.
(iv) overall sensitivity analysis. The behavior of the system will be investigated as depending upon perturbations in the whole set of primary events. The most significant cut sets will be thus extracted referring to the joint perturbative contribution of all primary events. This type of an analysis results particularly useful in the design and operation of nuclear power plants and safeguards.
(v) development of a computing code as in (ii), (iii), and (iv).

| TITLE 1 (original language)<br>Studi di affidabilità | Classification<br>14 |
|---|---|
| TITLE 2 (english)<br>Reliability studies | Country: ITALY<br>Sponsor: CNEN<br>Organisation: CNEN |
| Date initiated October 1974<br>Date completed in progress<br>Last updating June 1976 | Project Leader<br>G. Tomassetti |

General aim

Reliability studies of reactor components.

Particular objectives

Failure mode effect analysis of mechanical components of experimental loops and data collection techniques.

Experimental facilities

Reactors and out of pile loops of Casaccia Nuclear Research Center.

Project status

The formats for recording component events and related codifying techniques have been prepared. Work is under progress for 3 plants (2 experimental reactors, 1 thermohydraulic loop).

Next steps

Expand the work to other plants.

1505

| Title 1 (Original language) | | Classification |
|---|---|---|
| Raccolta dati di affidabilità | | 14 |
| **Title 2 (English)** | | Country ITALY |
| Reliability data collection | | Sponsor CNEN |
| | | Organisation CNEN |
| Date initiated | I974 | Project Leader |
| Date completed | In progress | |
| Last updating | April I977 | G. Tomassetti |

General aim

Studies on reliability data banks through experimental and metodological work on data collection field.

Particular objectives

Analysis of reliability data transferability from one plant to other ones in different general contexts.

Experimental facilities

CNEN experimental and conventional plants

Project status; next steps

Data from a sodium loop were elaborated; data collection from Diesel groups and nuclear facilities is under progress.

Reference documents

"Problems of experience data collection and use"
II Reliability Data Bank Seminar, Stockholm, March I977.

Degree of availability

Open

1507

| TITLE 1 (original language)<br>Fault tree analysis for nuclear power plants | Classification<br>$\boxed{6}$-14 |
|---|---|
| TITLE 2 (english) | Country: ITALY<br>Sponsor: CESNEF, Politecn. di Milano<br>Organisation: " " |
| Date initiated 1973<br>Date completed 1975<br>Last updating | Project Leader<br>S. GARRIBBA |

1509

| TITLE 1 (original language) | Classification |
| --- | --- |
| Fault analysis of the conventional island in a LWR nuclear power plant. | $\boxed{6}$ - 14 |
| TITLE 2 (english) | Country: ITALY<br>Sponsor: Franco Tosi S.p.A.<br>Organisation: " " " |
| Date initiated April 15, 1975<br>Date completed April 15, 1976<br>Last updating June 1976 | Project Leader<br>V. Bedogni |

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| | |
|---|---|
| TITLE 1 (original language)<br>Diagnostica e Sicurezza tramite Analisi di Rumore | Classification<br>$\boxed{8}$ - 14 |
| TITLE 2 (english)<br>Reactor Safety Studies Via Noise Analysis | Country: ITALY<br>Sponsor: CNEN<br>Organisation: CNEN |
| Date initiated 1/1/1975<br>Date completed in progress<br>Last updating June 1976 | Project Leader<br>F. Norelli |

1513

| | |
|---|---|
| PROJECT TITLE : Statistical analysis of random signals for safety problems | CLASSIFICATION $\boxed{10}$ - 14 - 8 |
| SPONSORING COUNTRY : Italy | ORGANISATION : CNEN |
| DATE INITIATED : 1966<br>DATE COMPLETED : in progress | PROJECT LEADER : A. Federico |

1515

| Title 1 (Original language) | Classification |
|---|---|
| Statistical analysis of randome signals | $\boxed{I}$ - 3 - 4 - 8<br>IO - I4 |
| Title 2 (English) | Country ITALY<br>Sponsor<br>Organisation } CNEN |
| Date initiated 1966<br>Date completed in progress<br>Last updating April 1977 | Project Leader<br><br>A. Federico |

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| Title 1 (Original language)<br>Diagnostica e Sicurezza tramite Analisi di Rumore (Parte A) | Classification<br>$\boxed{8}$ - 14 |
| --- | --- |
| Title 2 (English)<br>Safety and Diagnostics Via Noise Analysis (Part A) | Country ITALY<br>Sponsor CNEN<br>Organisation CNEN |
| Date initiated 1/1/1975<br>Date completed in progress<br>Last updating June 1977 | Project Leader<br>N. Pacilio |

| Title 1 (Original language) Diagnostica e Sicurezza tramite Analisi di Rumore (Parte B) | Classification 8 - 14 |
|---|---|
| Title 2 (English) Safety and Diagnostics Via Noise Analysis (Part B) | Country ITALY<br>Sponsor CNEN<br>Organisation CNEN |
| Date initiated 1/1/1976<br>Date completed in progress<br>Last updating June 1977 | Project Leader<br>A. Serra |

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| PROJECT TITLE : Evaluation of the risk of the rupture of a PWR reactor pressure vessel | LWR 14 |
|---|---|
| SPONSORING COUNTRY : Commission of the European Communities | ORGANISATION : J.R.C. Ispra |
| DATE INITIATED : 1976<br>DATE COMPLETED : 1977 | PROJECT LEADER : G. Volta/A.Lucia |

Description :

1. General aim

Rupture risk of a PWR pressure vessel.

2. Particular objectives

Development of a general computation method for evaluating the probability of failure of a nuclear vessel, on the basis of probabilistic fracture mechanics and of input data given as histograms.

3. Project status

The Coval 2 code allowing to handle dependent variables has been completed and it has been applied successfully to first cases proposed by CEA.

4. Next steps

The methodology that has been set-up will be used for investigating the general problem of pressure vessel reliability : variation in load condition, in initial fault dimension and location, in material toughness, etc. will be taken into account.

5. Reference documents

JRC Ispra Safety Programme Semiannual Progress Report 1977.

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| PROJECT TITLE : Status Report on Data Systems | LWR 14 |
|---|---|
| SPONSORING COUNTRY : Commission of the European Communities | ORGANISATION : JRC - Ispra |
| DATE INITIATED : 1976<br>DATE COMPLETED : 1977 | PROJECT LEADER : G. Volta |

Description :

1. General aim

Investigation of possibilities of pooling data on L.W. nuclear power plants in some form at a European scale in the frame work of the Reactor Safety Programme.

2. Particular objectives

In the first instance it is necessary for Ispra to have a review of all the existing systems and the existing source of data and how they are being used.

It is pertinent that this review should in fact begin to show the problems that may exist in communication of data-information between the European organizations.

There is a need for uniformity in the various concepts, such as

- ways and means in which data are collected
- how they are analysed
- how the final results of any data scheme are presented.

We intend to put some limitations on the total area of the activity, to avoid that the problem becomes too great.
It has been agreed to look at systems and components that are pertinent to LWR systems, but to take also account of systems or components from other commercial plants which may have a relevance to the LWR field.

The suggestion has been made to look at a selection of the higher system or sub-system levels, at the same time as the component level. It has also been recommended to consider areas where sufficient data are available to make useful comparisons, but to look also to some areas where until now data are missing, especially for mechanical components and/or subsystems.

31.719

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3. Project status

The following systems so far have been analyzed as for input, processing, output : IRS (Germany), ENEL (Italy), EDF (France), SRS (England), NPRDS (USA).

4. Next steps

Investigation on abnormal occurrencies as a supplement to the Reliability Data Status Report. Compatibility study on components/systems and failure characteristics.

5. Reference documents

JRC Ispra Safety Programme Annual Progress Reports 1976.

| | |
|---|---|
| PROJECT TITLE : Validation Experiments of a Reliability Analysis | LWR 14 |
| SPONSORING COUNTRY : Commission of the European Communities | ORGANISATION : JRC Ispra |
| DATE INITIATED : 1976<br>DATE COMPLETED : 1977 | PROJECT LEADER : J. Amesz/G. Mancini |

1. General aim

Analysis of abnormal accurrences leading to pressure variations.

2. Particular objectives

Validation of a probabilistic approach to abnormal occurence analysis by comparison of fault-tree analysis "a priori" results with "historical" data on a PWR.

3. Outline of the analysis

The "hystorical" data are selected from records of several years of operation of the reactor K.W.O. at Obrigheim, Germany. The "a priori" data are computed using the method described in 1974 Safety Annual Report (Probabilistic Accidental Transients Analysis).
The analysis will be performed according to the following steps:

- Listing of the power excursionsof whatever origin.
  Time series of these power excursions can be given in a diagram - $\Delta W$ (power variation) against t (time). Input data can be taken from the annual and monthly reports.

- For each power excursion the relevant initiating event (cause) should be given.
  Statistics of the initiating events in order to find the most frequent one.
  Input data as for point 1.

- For this initiating event the following analysis are foreseen:
  - Transforming the power decrease ($\Delta$ W) in primary pressure variation ($\Delta$p) taking into account also the pressure gradient $\Delta p$.
  - Determining for each pressure variation the state (failed or not failed) of the relevant safety and control systems.
  - Constructing the experimental truth table
  - From the foregoing table the experimental probability distribution is found.
    Input data can be obtained from the failure and switch-reporting system.

- Theoretical investigation through modelling the initiating event on the basis of elementary data and fault-tree analysis.

1526

For this investigation a technical knowledge about the concerned initiating event and the interventing systems is needed. This information could be obtained from functional blockdiagrams and flow-sheets.

- Confrontation of the final results of point 3 (experimental) and point 4 (theoretical).

## 4. Project status

### 4.1 Progress to date

#### 4.1.1 Pressure transients records analysis

Previously the analysis of the pressure transients caused by scrams and turbine trips has been performed [1]. During this period a systematic analysis of all pressure transients ($\pm$ 1 atm) has been started (Annex 1).

#### 4.1.2 Theoretical analysis

Considering the relevant importance of the shut-down system for the frequency of the transients and for their consequence a fault-tree analysis of this system has been started (Annex 1).
This analysis will be supported by a set of computerized tools for which a list is given in Annex 2.

## 5. Next steps

### 5.1 Pressure transients records analysis

Completion of the collection of pressure variations and their initiating events for the period 1970-1977.

Classification of initiating events and interventing systems

### 5.2 Theoretical analysis

Failure mode and effect analysis or fault-tree analysis of certain initiating events chosen from the data collection (It is expected to analyze 6 initiating events).

Fault-tree analysis of the most important interventing safety features and regulation systems, shut-down system, turbine trip, pressure regulation system.

1527

| PROJECT TITLE : Probabilistic Accidental Transients Analysis | LWR 14 |
|---|---|
| SPONSORING COUNTRY : Commission of the European Communities | ORGANISATION : JRC Ispra |
| DATE INITIATED : 1974<br>DATE COMPLETED : 1977 | PROJECT LEADER : G. Volta/J. Amesz |

Description :

1. General aim

To assess the probability distribution of loads acting on the reactor primary circuit.

2. Methodology and application

The presented method consists in a combined probabilistic-deterministic approach. This approach is probabilistic in the sense that for each combination of system stated a probability figure is determined by means of a fault-tree analysis. It is deterministic in the sense that the corresponding consequence (e.g. magnitude of overpressure or overtemperature) for each combination of states is computed by deterministic codes.

The approach has already been applied to PWR overpressures in the frame of the collaboration contract no. III-73-PIPGF CCR Euratom-Framatome.
The same approach will be applied to the temperature transients at a defined point of the primary circuit of the fast reactor PEC in the frame of the collaboration contract no. 144-75 PIPG/RN 163 CCR Euratom- CNEN.

3. Project status

3.1 Progress to date

A first conclusion of the work performed in the frame of a collaboration contract with CNEN (n° 164-75 PIPG/RN 163) has been presented at the International Meeting on Fast Reactor Safety at Chicago [1].
The reliability analysis of the electrical supply system has been completed assuming the various accidents as mutually exclusive events [2].
For this the computer code ELISA has been developed. The code accepts a fault-tree with NOT gates [3].

51.719

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3.2 Essential results

See reference documents.

4. Reference documents


/1/ J. Amesz, G. Volta, F. Cesari, R. Righini, "Probabilistic Analysis of Accidental Transients in a LMFBR" presented at FRS Meeting at Chicago (USA) 5-8 Oct. 1976.

/2/ J. Amesz, F. Lomazzi, "Analisi di affidabilità del sistema di alimentazione elettrica sul reattore PEC" Nota tecnica P.E.R. 126/77.

/3/ M. Astolfi, "ELISA, Codice per il calcolo dell'affidabilità di sistemi. Descrizione e modo d'uso". Nota tecnica P.E.R. 098/76.

1529

| Netherlands Energy Research Foundation (ECN) | | CLASSIFICATION: 14 |
|---|---|---|
| **TITLE:** Faalweg en faalkansvoorspellingen van reaktorsystemen. | | COUNTRY: THE NETHERLANDS |
| | | SPONSOR: Ministry of Social Affairs<br>ORGANIZATION: ECN |
| **TITLE (ENGLISH LANGUAGE):** Failure mode and failure rate prediction of reactor systems | | PROJECTLEADER: K. Terpstra |
| INITIATED: June 1974 | LAST UPDATING: May 1978 | SCIENTISTS: - |
| STATUS: final stage | COMPLETED: Summer 1978 | |

General aim

To predict the probability of failure of systems on basis of possible failure modes and component reliability, taking into account accidental as well as systematic failures due to external causes or inherent faults of components.

Particular objectives

The first stage of the program consists of:
- making an inventory of the general available methods of reliability analysis and "data banks" on failure rates
- establishing a standard procedure to collect data on failures in nuclear power stations in The Netherlands
- developing procedures to evaluate the reliability of reactor systems for specific cases applicable to Dutch power reactors.

Experimental facilities: None

Project status

a. An assembly of computercodes called "RECAL" has been made operational. Main calculations (including repair) are:
   * Determination of MCS
   * System, MCS and component characteristics
   * Ranking of MCS and components according to their importance (several criteria available)
b. A draft report, titled "System Reliability Analysis", has been written, containing:
   * The theoretical background of "RECAL"
   * How to use "RECAL"
   * An introduction into the state of the art of Data Bank Systems for LWR

Next steps

The commission fo the Ministry of Social Affairs will be terminated. A new commission is foreseen which will deal with the addition in the RECAL codes of:
* Confidence intervals
* Phased missions
* Dependency between basic events

Relations with other projects: -

Reference documents

Reliability analysis of systems;
Part I : Theory
Part II : Annexes
Terpstra K : ECN-78-draft; April 1978.

Degree of availability

The reports are written in the Dutch language. They are submitted to the Ministry of Social Affairs, Postbus 69, Voorburg.

Budget: -

Personnel: 0.2 manyear

1531

| N.V. KEMA | CLASSIFICATION : 14 |
|---|---|
| TITLE : Faalkans-analyse met behulp van gebeurtenissen- en foutenbomen | COUNTRY: THE NETHERLANDS |
| | SPONSOR : KEMA<br>ORGANIZATION : KEMA |
| TITLE (ENGLISH LANGUAGE): Failure analysis by application of event- and fault trees | PROJECTLEADER : R.W. van Otterloo |
| INITIATED : - LAST UPDATING : 1978 | SCIENTISTS : R.W. van Otterloo |
| STATUS : - COMPLETED : 1977 | |

General aim

To analyse new systems or changes in existing systems. And by application of this technique to make decisions concerning these new systems or changes in existing systems in the right way.

Particular objectives

- Analysis of the unavailability of reactor safety systems and of the unreliability of reactor systems.
- Analysis of the probability of different groups of radioactive releases of a nuclear power reactor.

Experimental facilities

Not applicable.

Project status

Methods and computer codes have been compared.

Next steps

Not applicable

Relation to other projects

This project was started to do the "Risk analysis of the fuel cycle in the Netherlands" (RASIN-study) which was finished in June 1975.

Reference documents

See 6.
Several applications of this method are written in the Dutch and English language.

Degree of availability

Through the organization KEMA

AS THIS PROJECT IS TERMINATED, NO FORMAT WILL BE ISSUED IN THE NEXT NUCLEAR SAFETY RESEARCH INDEX

# 15. INTERRELATION BETWEEN REACTOR PLANT AND OPERATING PERSONNEL

1533

Classification 15

| Title 1 | COUNTRY Denmark |
| --- | --- |
| Operatør studier | SPONSOR Risø National Laboratory<br>ORGANIZATION Risø National Laboratory |
| Title 2<br>Study of the Process Operator | Project leader:<br>L.P. Goodstein |
| Initiated: Approx 1966 - in its present form 1973<br>Completed:<br>Status: progressing<br>Last updating: 1976 | Scientists:<br>L.P. Goodstein<br>J. Rasmussen |

1. General aim

2. Particular objectives

To study the process operator - his work situation and procedures - together with methods for supporting him, especially in abnormal situations.

3. Experimental facilities and programme

- Hybrid computer
- Interactive CRT graphics terminal

4. Description and status

Work to date has been centered on the study of human behaviour in process plants as well as on various trouble-shooting tasks - in order to establish ways and means of providing support - especially in critical situations - but also in order to ultimately be in a position to include the effects of human actions/errors in systematic reliability and safety analyses.

In this work, we utilize tape recordings from power stations and other work situations as an aid in formulating ideas regarding the data, procedures and models utilized by the human in the various tasks. These can then be used in the design of the man-machine interface to provide improved aid to the operator. In addition, we expect to continue our studies of human error - either from published case stories or from simulator experiments.

## 5. Next steps

- A program to continue with the testing and validation of these models of the operator is being planned. Improved methods for the evaluation of risks which include human error will be evaluated.

## 6. Relation with other projects

A cooperative Scandinavian program is planned to the 1977-1980 period.

## 7. Reference Documents


J. Rasmussen 1969
Man-Machine Communication in the Light of Accident Records
July, 1969, S-1-69
Reprinted from IEEE-GMS, ERS International Symposium on Man-Machine Systems, Cambridge, 1969.

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# 16. ENVIRONMENTAL PROTECTION

# 17. NUCLEAR ACCIDENT RECOVERY AND DECOMMISSIONING

1537

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.1977 - 31.12.1977 | 17.1 | PNS 4411 |
| **Vorhaben/Project Title** Entwicklung von Dekontaminationsverfahren<br><br>Development of Decontamination Methods | | **Land/Country** FRG<br>**Fördernde Institution/Sponsor** BMFT<br>**Auftragnehmer/Contractor** KfK Projekt Nukleare Sicherheit, ABRA |
| **Arbeitsbeginn/Initiated** Jan 1973 | **Arbeitsende/Completed** Dec 1977 | **Leiter des Vorhabens/Project Leader** T. Dippel, D. Hentschel, S. Kunze |
| **Stand der Arbeiten/Status** Completed | **Berichtsdatum/Last Updating** Dec 1977 | **Bewilligte Mittel/Funds** |

1. General Aim

Decontamination processes for the decontamination of metal surfaces are optimized and new processes aimed at high decontamination efficiency and a minimum production of decontamination wastes. The processes are intended to be applied in a facility for conditioning nuclear components after incidents and for final disposal. The chemical agents used are compatible with the treatment and solidification processes for the radioactive liquids. In case selected chemicals can undergo special treatment to avoid disturbance in the before mentioned processes.

2. Particular objectives

Particular objectives are the reduction of waste volumes containing big loads of chemicals, compatibility of the decontamination agents with waste treatment and solidification processes, and the application of the decontamination processes, by remote working techniques.

3. Research Program

The essential experiments are the preparation of contaminated standard samples, variation of decontamination paste composition, study of the paste effectiveness in relation to composition and of the behavior of their components on waste treatment and solidification and estimation of waste production rates in comparison to liquid decontamination chemicals followed by the development of a application technique in thin layers for the paste.

Basically the same experiments are carried out with molten salts. Optimization experiments are to be started for better cleansing effectiveness with a liquid cleanser and decontamination chemicals.

4. Experimental Facilities

No special experimental facilities are necessary. Standard laboratory equipment is used for the experiments.

5. Progress to Date

5.1 Optimization of a liquid cleanser

Base material for a liquid cleanser have been selected by DT analysis with respect to their thermal stability. With the most stable types cleansers have been mixed with foam regulating agents. To this basic mixtures complexing agents have been added and the decontamination effectiveness was tested.

All available substances to avoid turbidity in a cleanser concentrate have been tested for thermal stability, foam generation and their influence of the effectiveness of the cleanser with respect to decontamination. Additional tests with stainless steel samples, glass samples and coating samples have been carried out to prove the effectiveness with respect to the decontamination capability.

5.2 Optimization of the Decontamination of Components

A selection of chemicals currently used for decontamination and applicated in dipping techniques have been tested for their effectiveness with respect to working temperature and concentration Furhter experiments are on the way. Out of others $HNO_3$, $HNO_3/HF$, alcaline $KMO_4$ solution, citric acid and oxalic acid are tested. Experiments showed, that the decontamination of nuclear components wtih currently used cleansers is not effective as long as these are used alone. A pretreatment with alkaline potassium permangante solution (AP process) followed by another cleansing step with the other decontamination liquid gives best results.

For the AP-solution generates the bulk of the decontamination waste, tests to reduce the concentration of the AP were carried out with samples contaminated in a reactor. A further reduction of waste was tested by using paste type cleansers.

## 6. Results

### 6.1 Optimization of a Liquid Cleanser

The above mentioned mixtures show a definitely better thermal stability in comparison to currently used liquid contamination cleansers. Up to now the decontamination results compared with those of the Papan-Dekopan cleanser are better with respect to Ru-106, practically the same with respect to Cs-137, and poorer with respect to Co-60.

The turbidity in the optimized, alkaline clanser could be eliminated by additives based on neutral phosphoric acid esters. There is no influence of this substance on foam generation, thermal stability and decontamination effectiveness. Comparitive tests with 25 mixtures showed, that only three of them are effective and cause no disturbance on waste treatment and solidification for disposal.

### 6.2 Optimization of the Decontamination of Components

Experimental results show that decontamination results are better at a working temperature of 80°C. With respect to effectiveness and amount of waste generated 4 m nitric acid is the best choice. $HNO_3$/HF 4 concentration can be reduced by a factor of 2, leaving the effectiveness unchanged. The same result is found for the concentration of alkaline $KMnO_4$ solution; the removal of Ru-106 with it is excellent. Regarding organic acids, the 1% oxalic acid and the 0.2 citric acid/0.3 m oxalic acid mixture are the most effective.

Decontamination experiments with an AP-process as pretreatment showed the following results:

- The optimum combination in AP-solution ( 3g $KMnO_4$ + 9 g NaOH per liter) and oxalic acid (1%)
- The reduction on waste mounts up to about 50% with respect to other decontamination processes.
- To use this decontaminat in form of pastes gives worse results with samples contaminated in a reactor
- Decontamination results are poor using AP-solution together with various concentrations of citric acid.
- The same results are gained by addition of EDTA (Na-salt)
- In general, with respect to the APOx, Citrox and the N Citrox processes, there is a reduction possible of the concentration of the chemicals applied.

7. Next Steps

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7.1 Optimization of a Liquid Cleanser

The R&D-work is completed

7.2 Optimization of the Decontamination of Components

The R&D-work is completed

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<br>1.10.1977-31.12.1977 | Klassifikation/Classification<br>17.2 | Kennzeichen/Project Number<br>RS 300 |
| Vorhaben/Project Title<br>Specifications of PWR system condition after a LOCA required to study the resulting decontamination and shipping problems<br>Spezifizierung des Anlagezustandes eines DWR nach einem Kühlmittelverluststörfall für die Untersuchung der daraus folgenden Dekontaminations- und Transportprobleme | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>NIS Frankfurt<br>Abt. Reaktoranlagen |
| Arbeitsbeginn/Initiated<br>1. 10. 1977 | Arbeitsende/Completed<br>31. 3. 1978 | Leiter des Vorhabens/Project Leader<br>A. Gasch |
| Stand der Arbeiten/Status<br>continuing | Berichtsdatum/Last Updating<br>31.12. 1977 | Bewilligte Mittel/Funds<br>187.970 |

1. General Aim

Basing on the NIS-study RS 155 "Result analysis of LOCA's for the dismantling of nuclear power plants" the post LOCA system condition of a 1300 MW PWR should be researched at which new boundary conditions are defined. The research results should deliver input data for the RS 156 project.

2 Particluar Objectives

In this study the conditions of nuclear components and equipments in some neuralgic regions of the reactor building are specified, which are important for the post LOCA decontamination and shipping problems.

3 Research Program

3.1 Description of the course of LOCA

3.2 Investigation of the radioactivity

3.3 Locale dispersion of the radioactive nuclides

3.4 Locale dose rates on component surfaces

3.5 Chemical conditions in the plant

3.6 Mechanical conditions of components and equipments

4 Experimental Facilities, Computer Codes

The core radioactivity will be calculated with the computer code ORIGEN. The calculation of radioactivity produced by activation in the core components will happen with the NIS code AKAT 2.

Further the calculations of locale dispersion of radioactive nuclides will be done with the well known computer code CORRAL

5 Progress to Date

Most of the necessary input date could be provided. The computer codes ORIGEN and AKAT 2 were actived for the new problems. The code CORRAL is in the status of testing.

6 Results

Definite results are not available.

7 Next steps

In the following periode the discribed calculations and investigations will be started.

8 Relations for other Projects

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9 References

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10 Availability

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 17.3 | PNS 4421 |

| Vorhaben/Project Title | |
|---|---|
| Development of Methods and Techniques for the Decommissioning and Ultimate Disposal of Nuclear Facilities<br>Entwicklung von Methoden und Verfahren zur Stillegung und Endbeseitigung nuklearer Anlagen | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: KfK Projekt Nukleare Sicherheit RBT/PB |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| January 1974 | 31.12.1977 | G.W. Köhler |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| completed | December 1977 | |

1. General Aims

Development of methods and techniques for the decommissioning and ultimate disposal of nuclear facilities.

2. Individual Tasks

2.1 Establishment of a general analysis for the safe decommissioning and ultimate disposal of nuclear power stations by evaluation of results obtained at home and abroad.

3. Program of Investigations

3.1 Dismounting of components

3.2 Means of crushing large components

3.3 Means of crushing concrete

3.4 Dismounting and conditioning of large components

3.5 Ancillary and auxiliary systems

3.6 Means of transportation for reactor components

3.7 Waste conditioning

3.8 Time schedule and preparations for decommissioning

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3.9 Spatial arrangement of reactor components

3.10 Reactor design suitable for later decommissioning

3.11 Typical working program.

## 4. Test Facilities, Computer Programs

## 5. Work Performed

Investigations relating to 3.3, 3.5, 3.7, 3.8, 3.9, 3.10 and 3.11 were completed during the period of reporting.

## 6. Results

Ad 3.3 The techniques used in dismantling the concrete structures mainly include:

a) Mechanical techniques (tools, pneumatic and hydraulic hammers, tubular drills).

b) Thermal techniques (equipments: oxygen lance, powder cutter).

c) Use of explosives (detonators, flat-type explosives, cutting charges).

Ad 3.5 Quite a number of ancillary and auxiliary systems are required during decommissioning work in order to protect the staff and the environment.

They mainly include:

devices and systems to be employed at the working place (e.g. tent, sucking off, filters, mist spray systems);

personnel protctive equipment (e.g. protective clothing, oxygen mask);

equipments and systems providing information about boundary conditions prevailing at the working place (e.g. audio and video service systems, radiation, temperature, and other measuring instruments).

Ad 3.7 The following possibilities are eligible in principle for waste conditioning:

- Conditioning in the nuclear power station using exclusively the means available in the nuclear power station (feasible only for small sized low-level components).
- Conditioning in the nuclear power station with assistance from outside (preferably for decommissioning due to age).
- Conditioning in the nuclear power station of dismountable facilities (both in case of decommissioning due to age and failure and in case of component replacement).
- Conditioning in a central facility (both in case of decommissioning due to age and failure and in case of component replacement; major transport problems might be encountered, depending on the size of the component).

Ad 3.8 Work scheduling comprises all the planning activities required to decommission the nuclear power plant. The preparations for actual decommissioning have to be performed parallel with planning and, partly, take a much longer time than planning.

Work scheduling is reasonably divided into a <u>preliminary planning phase</u> (definition of the target of the concept, estimate of work required, time schedule, necessary funds, etc.) and a phase of detailed planning (evaluation and specification of single working steps, safety and health physics program, organizational chart and time schedule). Prior to dismounting and demolishing work quite a number of preparatory steps are required (evaluation of boundary conditions by measuring technology, provision of tools etc.).

Ad 3.9 The accessibility of components and the space conditions in the reactor contribute decisively to the amount of radiation doses taken up by the personnel during decommissioning work.

The space conditions in light water demonstration reactors have to be generally classified as insufficient.

In recent light water reactors a tendency can be clearly recognized towards a more favorable arrangement of components and fittings.

Ad 3.10 It is recommended to take into account more than previously future decommissioning (complete removal) at the planning stage for a nuclear power plant.

These are the problems of major importance:

- Ratio of dimension to space available.
- Ratio of component weight to carrying capacity of lifting devices.
- Increased dose rate.

Ad 3.11 Decommissioning of a nuclear power station involves a number of typical sequences of operations occurring every time when large components are disposed of. They include:

evaluation of boundary conditions by measuring technology, decontamination (only if components have undergone noticeable contamination),

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disconnection of loop lines,
partial dismounting,
partial disassembly,
removal of bearings,
transport to fuel element storage basin,
cutting into parts suitable for ultimate storage,
waste fixation,
provision of transport shielding,
bag-out of containers,
transport to the final storage facility.

7. Plans for Future Work

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8. Relations to Other Tasks

–

9. Literature

–

10. Availability of Reports

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.10.77 | 17.3 | RS 236 |

| Vorhaben/Project Title | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Messerschmitt-Bölkow-Blohm, Schrobenhausen<br>Abt.: AF 24 |
|---|---|
| Controlled Blasting Demolition of Radioactive Heavy Components after Nuclear Power Plant Shutdown<br><br>Sprengtechnische Zerstörung von radioaktiven Großkomponenten bei der Stillegung von Kernkraftwerken | |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.10.77 | 31.5.78 | Dipl.Ing. Größler |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | Dec. 1977 | DM 363 222,00 |

## 1. Scope

Disposal of nuclear power plants after their lifetime of approx. 30 years.

## 2. Task

Nuclear power plants can be disposed of in various ways. Dismantling and removing heavy radioactive components by conventional means of engineering involves considerable difficulties. Therefore, the present research program aims at examining, both theoretically and experimentally, the blasting techniques of controlled demolition.

Hence, it is the task of this preliminary study to examine possible applications of blasting means and techniques in removing nuclear power plants after the end of their service time (normal decommissioning), and in removing specific individual, heavy radioactive components during the service period. Other disposal alternatives are to be compared to the blasting techniques.

## 3. Program

3.1 Analysis of the different radioactive heavy components

3.2 Examination of demolition means

3.3 Considerations as to the application of blasting means

3.4 Analysis of necessary auxiliary devices and aids

3.5 Component decontamination

3.6 Comparison of the different demolition techniques.

4. Test Facilities

ac 3.1 In order to determine the radiation intensity of the activated radioactive material, it is intended, together with the local technical staff to carry out radiation measurements at the nuclear power plant at Obrigheim, since this is the plant having the longest operation period in the FRG (Federal Republic of Germany). This should be done in order to obtain an up-to-date basis for extrapolating to the radiation intensity after 30 years from now.

ad 3.2 Utilization of the E-52 military test range at Oberjettenberg
Owing to safety considerations, the blasting facilities of the E-Stelle Oberjettenberg (German Army test range) shall be used for model-scale tests of the selected demolition techniques. The tests will be recorded partially with MBB-owned high-speed film cameras (Hycam and Locam types), so that secondary effects that go along with blasting due to fragments or target spalling can be determined.

5. Work carried out

ad 3.1 Analysis of the different radioactive heavy components
After two talks with the KWU and Hochtief corporations and with staff members of the GRS (Mr. Lerach), KWU has furnished to us the details of the following radioactive heavy components for analysis:
- reactor core tank with flanges and piping
- steam generator
- Main coolant pump
- pressure equalizer tank
- biological shield

A nuclear power plant of the Unterweser type was chosen for the present study, since the required details of this type can be obtained promptly from KWU. The talks were held on October 7 and November 17, 1977.

As to questions concerning radiation safety regulations, possible risks occurring during the demolition of radioactively contaminated components at decommissioned nuclear power plants have been discussed with Mr. Bendler from the Physics Departm. of Munich Technical University at Garching. This talk was held on November 11, 1977. Mr. Bendler is in charge of radiation safety at the Garching reactor. The permitted amounts of radiation to which individuals may be exposed within various durations of time are one criterion in the possible application of blasting techniques.

The permissible radiation is 5 rem per year, or 2,5 rem per quarter, a maximum of 1 rem per month or of about 2,5 mrem per hour. This last figure may be critical to the application of blasting techniques since these will probably take only very short time.

A doctor's thesis, by W.Buschmann ("Disposal of decommissioned nuclear power plants ..."), and the Nukem report RS-156 have been procured in order to obtain data on the radioactive contaminaiton in nuclear power plants after 30 or 40 years of operation. Moreover, a copy of the AIF report on "An engineering evaluation of nuclear power reactor decommissioning alternatives" has been ordered. The KWU and TU Munich specialists have the opinion that long-term predictions are always rather unreliable. KWU has indicated that the radiation dose values given by Buschmann in his thesis are probably too high. It may therefore be necessary to carry out measurements at the Obrigheim nuclear power plant so as to obtain better predictions on the induced radioactive radiation and, if possible, also on the contamination.

ad 3.2 Examination of demolition means

Steel pipes and linear shaped charge cords have been ordered for the intended model-scale tests.

In a letter to the BMFT, Ref. 313 (dated November 16,1977), we have asked for official support to get permission for making the model-scale tests at the E-52 Oberjettenberg military test range. In Schrobenhausen, such tests (with model-scale pipes and concrete structures) can be made only with an extreme amount of safety provisions. The use of the E-52, on the other hand, depends greatly on the weather (snow) which will possibly delay the schedule of this study, which was to be terminated by May 31, 1978, by approx. 2 months. However, this will not involve any additional financial requirements.

A liquid explosive (Teledet) has been ordered for the tests concerning the reactor biological shield.

ad 3.3 Considerations as to the application of blasting means
We have visited the nuclear power plant at Unterweser on November 29, 1977 in order to learn about the installations and to get a general idea of how to apply the blasting means. On this occasion, photographs were taken of various critical structural situations.

ad 3.4 Analysis of necessary auxiliary devices and aids
cf. 3.3 (under work)

ad 3.5 Component decontaminaiton
(under work)

ad 3.6 Comparison of the different demolition techniques
- various data on linear shaped charges have been furnished to Dr.Freund of Battelle-Institut (Frankfurt) for an oral presentation, in which various ways of size reduction of metal components were discussed (paper given at the 4th conference on the "Application of Explosives in Manufacture" at Essen, September 12-16, 1977).
- Evaluation of the report for the comparison of the different size reduction ("demolition") techniques which is to conclude the study.

## 6. Results

ad 3.1 Analysis of the different radioactive heavy components
- Determination of the radioactive heavy components and concrete structures (reactor care tank, steam generator,

biological shield, pipes, main coolant pump).
- Evaluation of the Nukem report and Buschmann thesis to predict radioactive contamination after 30 years of operation.
- The use of blasting means must be based on permissible radiation doses of 2,5mrem/h or up to 1 rem in case of very short exposure. Together with the estimate of radioactivity, which has yet to be accomplished, these figures are a basis of judging the applicability of blasting techniques (time of exposure).

ad 3.2 Examination of demolition means

The following items have been ordered and will be used in the tests and model-scale tests:
- various grade linear shaped charge cords
- liquid explosive
- steel pipes of different size
- concrete structure elements
- linear shaped charges of 1, 3 and 5 cm base width have been ordered to make.

ad 3.3 Considerations as to the application of blasting means

(under work)

ad 3.4 Analysis of necessary auxiliary devices and aids

(under work)

ad 3.5 Component decontamination

Steel pipes and concrete blocks have been ordered; they are to be used in studying spalling effects caused by detonative shock waves.

ad 3.6 Comparison of the different demolition techniques
- Determination of the performance (cutting depth) of 10 - 15 cm wide linear shaped charges as a basis to the paper of Dr. Freund.
- Evaluation of the Nukem report RS 156.

7. Planned Activities

ad 3.1 - Talk to the managing director of the Obrigheim nuclear power plant with the aim to obtain radiation measurement data for a better prediction of radiation intensity after 30 years of operation. It is planned also to have Dr. Buschmann attend this discussion.
- Examination of the drawings of heavy components, furnished by KWU, and inspection of various heavy components and, if possible, procurement of parts thereof (reject).

ad 3.2 - Manufacture of different blasting means such as linear shaped charges, conical shaped charges etc.
- test firings of the smaller size linear shaped charges in Schrobenhausen.

ad 3.3 Considerations as to the application of blasting means
On the basis of the different demolition techniques, designs of the combined application of linear and conical shaped charges etc. on the reactor components will be worked out.

ad 3.4 Analysis of necessary auxiliary devices and aids
The installations found available at the Untwerweser nuclear power plant will be listed; the additional installations required when blasting techniques are used will be checked.

ad 3.5 Component decontamination
- Blasting test will be made to determine spalling from steel pipes and concrete structures produced by directly attached plastic high explosive sheets.
- During the visit at the Obrigheim nuclear power plant we shall inquire about recent results concerning the primary circuit contamination and to adapt the spalling-firings to "weak spots".

ad 3.6 Comparison of the different demolition techniques
Evaluation of published material, such as Buschmann (thesis), Nukem report (RS-156), Freund (conference paper), U.S. report (AIF/NESP-009 S.R.).

8. Connections with other projects

none

9. Literature

- Nukem report 312 (RS 156), Hanau (1977)
- W.Buschmann, "Die Beseitigung von stillgelegten Kernkraftwerken mit Druckwasserreaktoren am Beispiel des Kernkraftwerkes Obrigheim", Doctor's thesis, Ruhr-Universität, Bochum (1976)

10. Availability of reports

MBB - Schrobenhausen.

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 01.09.1977-31.12.1977 | 17.3 | RS 274 |

| Vorhaben/Project Title | |
|---|---|
| Investigations on the Applicability and Efficiency of Powder Cutting and Plasma Cutting for Dismantling Components of Nuclear Power Stations<br>Untersuchungen über die Anwendbarkeit und Leistungsfähigkeit des Pulverbrenn- und Plasmaschneidens für das Zerteilen von Kernkraftwerkskomponenten | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Salzgitter AG 3320 Salzgitter 41 |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 01.09.1977 | 31.12.1979 | Dipl.-Ing. H. Deipenau |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | December 1977 | DM 691.966,-- |

## 1. General Aim

Under the Federal Government's research programme for reactor safety, it is necessary to investigate dismantling methods for activated and contaminated components in connection with the shutting down and dismounting of neclear power stations.

## 2. Particular Objectives

This project aims to check the applicability of flame cutting by powder and plasma cutting for taking to pieces of components of a nuclear reactor so that if a specific dismantling case arises, a decision can be taken as to which thermic cutting method may be used for taking a component to pieces and what conditions must exist for this.

The great number of components existing in a nuclear power station makes it necessary to collect in a catalogue the available plant information for a referece nuclear power station. The catalogue is to give information on dimensions and weights, material composition and surface dose. This collection is to serve as a basis for a typification of the plant components in order to reduce the number of the different dismantling tasks. Out of the typified components a selection will be made on which one is to be taken to pieces by means of flame-cutting by powder and plasma cutting. For each process, an application can then be found related to the individual components. The dust values obtained give basic reference values for the design of an efficient exhaust unit in order to prevent the activated dust form spreading and to keep the exposure level of personnel to radioactivity as low as possible.

## 3. Research Program

3.1 Drawing up a catalogue of the activated and contaminated components of a referece nuclear power station (DWR)

3.2 Experiments for demonstating the efficiency of the powder cutting and plasma cutting process

3.2.1 Powder cutting of reinforced concrete

3.2.2 Powder cutting of steel

3.2.3 Plasma cutting in the atmosphere

3.2.4 Plasma cutting under overpressure in the air

3.2.5 Plasma cutting under overpressure in water

3.3 Judgement on the powder cutting and plasma cutting process in relation to the dismantling of reactor components

## 4. Experimental Facilities

re 3.2.1 and 3.2.2 The experimental station for powder cutting is equipped with 2 gas cutting outfits of different capacity 650 mm and 1,200 mm wall thickness).
Propane is used as fuel gas. The cutting process is observed from a control room by television cameras and from there it is also operated by means of a simple mechanical control. An existing filter plant is used for suction. It can suck out up to 60,000 $Nm^3/h$ of spent air form the cutting cubicle. The dust particles for the plant are removed by means of a bypan.

re 3.2.3 The experimental station for plasma cutting in the atmosphere consists of one chamber with far simpler mechanical manipulation of the television camera. By means of an induced draught fan, the welding waste gases (about 2,000 $m^3/h$) can be extracted and analysed.

re 3.2.5 The cutting chamber is designed in such a way that by flooding even the first preliminary investigations for plasma cutting and the television observation can be tested in water.

re 3.2.4 and 3.2.5 For plasma cutting in overpressure conditions in air and water, a 2 bar pressure chamber has been constructed, in which cutting experiments can be tested in water 10 m deep by remote control. The final design can, however, not be presented until the component test runs have been completed. With the available sources of electricity the burner can reach loads of up to 500 A. The peak load can be raised to about 650 A by additionally connecting further sources of electricity.

## 5. Progress to Date

re 3.1 Work was started on drawing up the catalogue on the activated and contaminated components of a reference nuclear power station.

re 3.2.1 and 3.2.2 The planning of the experimental test floor is complete, so that the contracts can be awarded for the manufacture of the experimental station.

re 3.2.3 The planning of the experimental test floor is complete. Work is still in progress for the equipment of the experimental floor.

re 3.2.4 The planning of the experimental test floor and the pressure chamber is largely complete. Work has not yet started on of construction of the presure chamber.

re 3.2.5 Work has begun on the preliminary experiments for using a television camera.

## 6. Results

re 3.1 and 3.2 — Noteworthy intermediate results are not yet available. KFA-Karlsruhe will publish a report on the final removal of the HDR core shell in Großwelzheim.

## 7. Next Steps

re 3.1 — The work will be continued and it will probably not be possible for it to be completed in the coming report period, as power station documents on contaminated plant components are still missing. The work under 3.2 and 3.3 will not be affected by this postponement.

re 3.2.1, 3.2.2 and 3.2.3 — Equipping of the experimental station as well as optimisation of exhaustion and sampling will be completed, so that initial cutting experiments can be started.

3.2.4 — Construction of the pressure chamber has been started.

3.2.5 — The preliminary experiments will be continued for the use of underwater television cameras.

## 8. Relation with other Projects

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## 9. References

Reports on the following BMFT research projects:

RS 220

RS 155

RS 156

## 10. Degree of Availability of the reports

–

# 18. FUEL CYCLE

1561

| 140-3-02<br>251-2-01 | | 18 |
|---|---|---|
| Titre<br><br>COMPORTEMENT DES COMBUSTIBLES PWR EN COURS DE TRANSPORT | | Pays<br>FRANCE<br><br>Organisme directeur<br>CEA/DSN/BEST |
| Titre (anglais)<br><br>PWR FUEL BEHAVIOUR DURING TRANSPORTATION | | Organisme exécuteur<br>DSN/BEST/Fontenay<br><br>Responsable<br>DSN/BEST/FAR |
| Date de démarrage Janvier 1978 | Etat actuel : en cours | Scientifiques |
| Date prévue d'achèvement 1982 | Dernière mise à jour 15/01/78 | |

## 1/ OBJECTIF GENERAL

Des combustibles irradiés qui étaient stockés dans l'eau sont exposés à l'air. Leur température augmente et ils sont soumis à diverses agressions mécaniques lors du transport.

Le but de l'étude est de mettre en évidence les conséquences de cette situation sur les éléments apparemment sains, comme sur ceux dont l'état est douteux.

## 2/ OBJECTIFS PARTICULIERS

L'assemblage combustible a été modélisé par diverses équipes du point de vue mécanique, (FRAMATOME, CEA-Saclay)

Dans le cadre de cette sous tâche on explorera, dans un premier temps, la possibilité d'utiliser ces modèles pour définir la réponse des combustibles à des sollicitations susceptibles de survenir en cours de transport ou en cours d'accident de transport (chocs et vibrations).

Par ailleurs, on étudiera le problème de l'oxydation de l'UO2 en U3 O8 au contact de l'eau en fonction de la température.

Cette réaction qui s'amplifie à haute température et qui est exothermique peut devenir dangereusedans certaines conditions.

## 3/ PROGRAMME

Au cours de cet exercice on prendra contact avec FRAMATOME afin d'évaluer les possibilités d'utilisation de leurs modèles mécaniques dans le cadre des transpor normaux ou accidentels.

## 4/ ETAT D'AVANCEMENT

Lancement.

1563

| | |
|---|---|
| TITLE 1 (original language)<br>Valutazione di proprietà chimiche e fisiche di rifiuti radioattivi solidificati | Classification<br>18. |
| TITLE 2 (english)<br>Evaluation of chemical and physical properties of active solidified wastes. | Country: ITALY<br>Sponsor: CNEN<br>Organisation: CNEN |
| Date initiated 1975<br>Date completed 1977<br>Last updating June 1976 | Project Leader<br>F. GERA |

Description : The purpose of the project is to control the main chemica and physical properties of active solidified wastes to reduce possibl long-term environmental impacts.

Radioactive solid wastes of nuclear reactors are fixed in beton or plastic compounds. On the resulting solid wastes many tests will be carried out to evaluate the following characteristics: leaching, thermal stability, radiation damage, compression, etc.

| TITLE 1 (original language) Ricerca e sviluppo su contenitori di trasporto per fissili | Classification 18 |
|---|---|
| TITLE 2 (english) Research and development on fuel casks | Country: ITALY<br>Sponsor: CNEN<br>Organisation: (*) |
| Date initiated 1974 (present phase)<br>Date completed In progress<br>Last updating June 1976 | Project Leader<br>CNEN - Divisione Ricerche Sicurezza |

(*) CNEN-ENEL-AGIP Nucleare-FIAT Nucleare -Nuovo Pignone-Università di Pisa.

General aim

Researches on spent fuel shipping casks.

Particular objectives

Preliminary design, experimental researches, calculations (shielding, mechanics, heat trasnfer), model elaboration, tests, optimization studies on type of casks and means of transport related to power plants and reprocessing facilities.

Project status

A first set of research contracts (CNEN-AGIP Nucleare, CNEN-FIAT Nucleare, CNEN-Pisa University) and additional studies is completed.

Next steps

Researches related to some aspects of casks development and licensing.

Relation to other projects

18 (Pisa University)

| TITLE 1 (original language)<br>Ricerche e sviluppo su un contenitore di trasporto. | Classification :<br>18 |
|---|---|
| TITLE 2 (english)<br>Research and development of a fuel cask. | Country: ITALY<br>Sponsor: C.A.M.E.N.<br>Organisation: C.A.M.E.N. - UNIVERSITY OF PISA |
| Date initiated : 1975<br>Date completed ; In progress<br>Last updating June 1976 | Project Leader :<br>G. SARNO (CAMEN)<br>G. FORASASSI (UNIVERSITY) |

Description :

Preliminary design, experimental researches, calculations (mechanics, heat trasfer, etc. ....), model elaboration, tests and optimization studies on the necessary modification of an obsolete cask for spent fuel.

1569

| Title 1 (Original language)<br>Sicurezza del trasporto del materiale radioattivo. | Classification<br><br>18 |
|---|---|
| Title 2 (English)<br>Safety Problems in Design of Packaging and Transportation of Radioactive Materials. | Country ITALY<br>Sponsor CNEN-CNR<br>Organisation University of Pisa |
| Date initiated 1968<br>Date completed 1980<br>Last updating May 1977 | Project Leader<br><br>FORASASSI Giuseppe |

1) General aim
Study of safety problems in design, construction and use of transport casks for radioactive materials;

2) Particular objectives
Present research activity is referred to determination of energy absorption characteristics of steel structures and to test models and components to support the design of spent fuel shipping casks.

3) Experimental facilities and program
Facilities in Scalbatraio Center of University of Pisa can be utilized for experimental studies as well as to carry out IAEA standard test series on packagings for radioactive materials.
The following equipments are operable:
a) Drop test tower for casks and models up to 2000 Kg of weight;
b) Guided impact hammer (weighting up to 2300 Kg) for dynamic tests;
c) Thermal test station (open fire);
d) Vessel for hydraulic and tightness tests;
e) Instrumentation for measures of acceleration, displacements, forces and temperatures during tests.

4) Project status
- Research and testing series, completed two years ago, have brought to finalize the design of a low cost type B, fissile class II container;
- Studies and test program have been carried to determine mechanical energy absorption properties of several types of metallic structures related to the design of casks, as:
  -Steel honeycomb
  - Straight and circular fins.

5) Next steps
Up to date study and test programs include:
- Several tests on finned surfaces
- study of containment problems as well as performances of gaskets
- Studies and dynamic tests on bolts and tie-downs
- Study of problems related to the use of models in standard tests.

1570

| Title 1 (Original language) | Classification |
|---|---|
| Sicurezza del trasporto del materiale radioattivo. | 18 |

6) Relation to other projects
18 (Other programs)

7) Reference documents:


1. G. ELETTI, R. ZAMBRINI, G. FORASASSI, P. SACCOMANNO
Sicurezza del trasporto del materiale radioattivo.
Atti del convegno sulla sicurezza negli Impianti Nucleari.
Pisa 21-26 Settembre 1970.

2. P. CITTI, G. FORASASSI, B. GUERRINI
Rilievi su prototipi di un contenitore per il trasporto di material
radioattivo.
Atti del 1° Convegno AIAS - Palermo 1972.

3. G. ELETTI, G. FORASASSI
Italian testing station following IAEA Regulation; CF6 a fissile Class II Type B packaging.
4° International Symposium on packaging and transportation of Radioactive materials.
22-27 Sept. 1974.

4. P. CITTI, G. FORASASSI, S. REALE
Problemi di similitudine connessi con due prove standard per contenitori per trasporto di materiale radioattivo .
Ist. Imp. Nucl. Università di Pisa. RL 171(74).

5. G. FORASASSI, B. GUERRINI
Comportamento di una struttura metallica a celle esagonali in prove di simulazione di "fuel cask drop".
Ist. Imp. Nucl. Università di Pisa. RL 204(75).

6. G. FORASASSI
Assorbimento di energia meccanica mediante superfici alettate.
Ist. Imp. Nucl. Università di Pisa. RL 209(75).

| Title 1 (Original language) | Classification |
|---|---|
| Ricerche e sviluppo su un contenitore di trasporto. | 18 |
| **Title 2 (English)** | Country ITALY |
| Research and development of a fuel cask. | Sponsor: C.A.M.E.N.<br>Organisation C.A.M.E.N. - UNIVERSITY OF PISA |
| Date initiated : 1975<br>Date completed: In progress<br>Last updating : April 1977 | Project Leader<br>G. SARNO (CAMEN)<br>G. FORASASSI (UNIVERSITY) |

Description :

Preliminary design, experimental researches, calculations (mechanics, heat trasfer, etc. ....), model elaboration, tests and optimization studies on the necessary modification of an obsolete cask for spent fuel.

| Title 1 (Original language) | Classification |
| --- | --- |
| Valutazione delle proprietà chimiche e fisiche di rifiuti radioattivi solidificati. | 18 |
| **Title 2 (English)** Evaluation of chemical and physical properties of radioactive solidified wastes | **Country** ITALY<br>**Sponsor** CNEN<br>**Organisation** CNEN |
| **Date initiated** 1975<br>**Date completed** 1978<br>**Last updating** 1977 | **Project Leader**<br>A. DONATO |

General aim

The technical and experimental control of the main chemical and physical properties of solidified radioactive wastes, for obtaining basic technical data to evaluate possible long-term environmental impacts. Liquid radioactive wastes (evaporation concentrates, exhausted ions exchangers etc.) and solids are incorporated in cement, concrete, bitumen or plastic. The following characteristics are taken into account: leachability, thermal stability, radiation damage, compressive strength and weathering resistance.

Particular objectives

Set-up of methods for testing

Project status

Urea-Formaldheyde and Polymer Impregnate Cement have been evaluated

Degree of availability

Cooperation may be envisaged for standardization of testing methods. Exchange of information can be arranged.

| Title 1 (Original language) | Classification |
|---|---|
| Ricerche su contenitori di trasporto per materiale fissile | I8 |
| **Title 2 (English)** | **Country** ITALY |
| Researches on shipping casks for fissile material | **Sponsor** CNEN<br>**Organisation** (*) |
| **Date initiated** I974 (present phase)<br>**Date completed** In progress<br>**Last updating** April I977 | **Project Leader**<br><br>CNEN - Divisione Ricerche Sicurezza |

(*) CNEN-ENEL-AGIP Nucleare-FIAT Nucleare - Nuovo Pignone-Università di Pisa.

| | |
|---|---|
| General aim | Research and development on shipping casks for fissile material. |
| Particular objectives | Development of spent fuel shipping casks. Design, experimental researches, calculations (shielding, mechanics, heat transfer) model elaboration, tests, optimization studies on type of casks and means of transport related to power plants and reprocessing facilities. |
| Project status | A first set of research contracts (CNEN-AGIP Nucleare, CNEN-FIAT Nucleare, CNEN-Pisa University) is completed; other additional studies have been carried out. |
| Next steps | Researches related to some aspects of casks development and licensing. |
| Relation to other projects | I8 (other programs) |

1577

| N.V. KEMA | CLASSIFICATION : 18 |
|---|---|
| TITLE :<br>Ontwikkeling van een computercode voor het schrijven van een éénduidige en optimale herladingsprocedure | COUNTRY: THE NETHERLANDS<br>SPONSOR : KEMA<br>ORGANIZATION : KEMA |
| TITLE (ENGLISH LANGUAGE):<br>Development of a computercode which writes an unambiguous and optimum reload procedure | PROJECTLEADER : K.P. Termaat |
| INITIATED : Jan. 1976 — LAST UPDATING : June 1978 | SCIENTISTS : K.P. Termaat |
| STATUS : see below — COMPLETED : 1978 | |

General aim

The programme "RELOAD" will provide the operator of a nuclear power plant with a stepwise written optimum reload procedure. The programme is based on octant symmetric reload patterns, however non symmetric fuel assembly movements can be included.

Particular objectives

The programme is developed to be applied in the Dodewaard nuclear power plant. The objective is to minimize the number of refuelling steps and the quantity of time to reload the core. The programme will prevent errors which can possibly be made by handwriting the elaborous procedure, especially when a large number of reload elements, shuffle elements, dummy elements and inspection elements are involved in one reload scheme.

Experimental facilities

Not applicable.

Project status

The programme "RELOAD" has succesfully been applied in the Dodewaard nuclear power plant during the 8th refuelling outage. Minor modifications will be introduced. "RELOAD" will also be used for the next refuelling outage in 1979.

Next steps

Complete development and test with previous handwritten reload procedures.

Relation to other projects

Physics reload scheme, fuel inspection programme, fuel test programme.

Reference documents

Three reports (in Dutch) are available through the organization N.V. KEMA.

Degree of availability

Through the organization KEMA.

# 19. ECONOMICS OF SAFETY

# 20. OTHER TOPICS

1579

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.1977 - 31.12.1977 | 20 | RS o123 A und B |
| Vorhaben/Project Title<br>HDR-Sicherheitsprogramm<br><br>HDR Safety Program | | Land/Country<br>FRG<br><br>Fördernde Institution/Sponsor<br>BMFT<br><br>Auftragnehmer/Contractor<br>Kernforschungszentrum Karlsruhe |
| Arbeitsbeginn/Initiated<br>April 1, 1974 | Arbeitsende/Completed<br>1981 | Leiter des Vorhabens/Project Leader<br>Müller-Dietsche |
| Stand der Arbeiten/Status<br>continuing | Berichtsdatum/Last Updating<br>December 1975 | Bewilligte Mittel/Funds<br>14.986.ooo,-- DM (A)<br>34.379.5oo,-- DM (B) |

## 1. General Aim

The safety program which will be performed on the decommissioned HDR-plant near Frankfurt/Germany, serves for clarification of important problems within the framework of the light water reactor safety research program of the Federal Minister of Research and Technology (BMFT).

The unrestricted availability of this reactor for investigations concerning safety technology offers the rare possibility to check on a complete plant, the validity of engineering theories, applied design methods and the results achieved in current research projects up to the actual limits of stress bearing capacity of specific reactor components.

## 2. Particular Objectives

The HDR safety program is divided into five Specific Projects (SP):

HDR Safety Program

| Spec.Proj. 1000 | Spec.Proj. 2000 | Spec.Proj. 3000 | Spec.Proj. 4000 | Spec.Proj. 5000 |
|---|---|---|---|---|
| Non-destructive Tests | Pressure vessel and Piping Studies | Blowdown Studies | Earthquake Studies | Leak rate Studies |

These five Specific Projects were determined due to the main objectives presently pursued under the Reactor Safety Research Program of the Ministry after harmonization with the involved authorities.

## 3. Experimental Facilities and Program

### 3.1 Experimental Facilities

The HDR with loo MWth power was built to demonstrate the possibility of nuclear superheating. Typewise, it largely corresponds to the German water boiling reactor designed around 1967/1968. The main data can be seen from Table 1. Particular importance is attached to a full pressure containment and the reactor pressure vessel. No nuclear operation is envisaged for the experiments. The experimental conditions will be obtained with an electrically heated boiler.

### 3.2 Program

#### 3.2.1 Specific Project 1000 : Non-destructive Testing

Non-destructive testing serve for controlling the reactor pressure vessel and the piping under pressure tests and blowdown-tests and for testing in-service inspection systems under development. The following objectives will be investigated:

- Assessment of record of the initial conditions
- Detection of failure initiation and failure propagation
- Testing of inspection methods; proof of the suitability of new test methods
- Comparative evaluation of non-destructive testing systems, such as ultrasonic impulse testing, magnetic particle testing, penetration testing, eddy current method, ultrasonic scattering, acoustic emission, acoustic holography, radiography.

#### 3.2 Specific Project 2000 : Pressure Vessel and Piping Studies

Studies of pressure vessel and piping should be a further step of intensifying and securing the basic knowledge of their safe design. The most important aspect is a study of the effective safety margin above the design ratings of components. The following objectives are pursued:

- Material properties at the beginning of the experiments.
- Component behavior under special loading conditions,

e.g., operating conditions, blowdown, thermalshock, earthquake.

-- Component behavior under specific weakening - fabricational and operational defects and crack formations - to determine boundary strengths or critical defects.
-- Reliability of the employed methods of calculation.

3.2.3 Specific Project 3000 : Blowdown Studies

In the blowdown studies it is possible to measure radial stresses of reactor pressure vessel internals, containment structures and full scale valves by means of reactor typical impulse and mass-flow excitations in their realistic dimensions. The studies comprise the following objectives:

- Stress behavior of the reactor pressure vessel internals and containment structures
- Behavior of full scale valves under mass-flow excitation.
- Testing and further development of the various fluid and structural dynamic computer codes.

3.2.4 Specific Project 4000 : Earthquake Studies

Vibration-studies serve to improve and correct the basic knowledge about the safe design of building structures and pipe-systems with respect to the effects of earthquake. This implies the following objectives:

- Verification and optimization of available analytic methods
- Influence of material and soil characteristics and design conditions
- Testing of experimental techniques
- Advancement of analytical procedures for the non-linear range.

3.2.5 Specific Project 5000 : Leak Rate Studies

These studies serve to determine the previously unsettled parameters with the aim to standardize the leak rate procedure and the in-service studies of nuclear power stations.

This involves the following objectives:

- Leak behavior of a cold plant.
- Leak behavior of a plant at operating temperature.
- Verification and optimization of existing analytical procedures.

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# 4. Project Status

## 4.1 Progress to Date

### Central Preparation of the HDR-Experiments

The experimental test loop incorporating electric heating which is used to simulate approximate operating conditions prior to loss of coolant experiments was both installed and checked out. In conjunction with this the central data acquisition system for the approximately 450 quick and 150 slow response time measuring stations which are anticipated, was installed in the control room of the plant. The leads to all the measurement transducers were enclosed in protective cables to guard against damage during blowdown experiments. For off-line data processing and storage, a computer supported documentation system was developed, and brought into operation prior to the production of the first results. Technical safety analyses were made which confirmed that neither the building nor the structure were endangered by the first blowdown experiments (s. SP 3ooo).

### Specific Project 1000 : Non-destructive Tests

The evaluation of both the first automatic ultrasonic, and complementary visual inspections of the pressure vessels was completed and its results compiled in a report. Acoustic emmision and stationary ultrasonic probes were investigated for use under simulated operating conditions ( 110 bar, 310°C). A program was specified for theoretical and experimental analyses of crack growth in conjuction with the planned thermal shock experiments (SP 2000).

### Specific Project 2000 : RPV and Piping Studies

The loading of the material in the vessel was experimentally analysed under warm static conditions (110 bar, 310°C) as well as during the first blowdown experiments under boiling water conditions. For these, detailed strain measurements were made on both the inside and outside of the RPV walls. The reaction of the exit pipe system for the blowdown experiments involving a test valve was determined in the same manner. Laboratory experiments on the HDR Piping system sections were specified and prepared as was the thermal shock test program.

Specific Project 3000 : Blowdown Studies

After completion of the preparation of both plant equipment and the measurement system, the first series of blowdown experiments with a steam isolation valve was carried out. Four experiments were performed under different conditions:

| | |
|---|---|
| Nominal data: | 7o bar, 280°C, water level 4 m in a 12 m high RPV, rupture nozzle NW 450 |
| Variable data: | |
| Experiment 1 : | through put 1oo %, pure steam flow, valve closure without intentional delay |
| Experiment 2 : | through put 2oo % otherwise as in 1 |
| Experiment 3 : | 2oo %, intentional valve closure delay of 4 seconds, 2-phase flow, water - steam |
| Experiment 4 : | through put 4oo %, otherwise as in 1 and 2 |

Accompanying the experiments were both fluid and valve dynamics calculations. The final evaluation and analysis of the results is not yet completed. The installation of equipment for the next set of blowdown experiments with a feed water check valve (NW 2oo) has been initiated.

Specific Project 4000 : Earthquake Studies

The calculations decribing the behavior of the building, piping systems and pressure vessel in their natural vibrational modes were completed and compared with the experimental results from low level excitation tests. Moreover, various procedures were utilized to calculate their behavior under forced vibration. The comparison of these results with experimental measurements is still in progress. Specification and analytical preparation was initiated for experiments with higher excitation levels.

Specific Project 5000 : Leak Rate Studies

Additional measurements were performed on the plant in a cold state with the purpose of gaining an exact knowledge of the containment volumes and the level of the critical swelling pressure. This evaluation is still in progress and experiments on the plant in a warm state are not foreseen until it is completed.

## 4.2 Essential Results

The experimental analysis of strains and temperatures (SP 2000) measured during cold and warm pressure tests on the Reactor Pressure Vessel (RPV) show quite plainly that, even in steady state conditions, a relatively high temperature difference can occur - particularly in flange and bolting regions - which can lead to considerable thermal strains. In none of the blowdown experiments conducted to date, has the loading in the RPV exceeded that caused by intial conditions.

The blowdown experiments (SP 3000) with the steam isolation valve NW 500 demonstrated that the valve can withstand the loading which occurs in a loss of coolant accident several times with neither damage nor impairment of function. There was a relatively good agreement between calculated and measured closure behavior. The mass flow corresponded to what was expected. No substantial damage occured in the containment.

The comparison of the analytical results (SP 4000) describing behavior of the building, piping systems, and pressure vessel in their natural vibrational modes with measurements taken under low excitation gave a good agreement in part, but in several of the parallel calculations a definate departure was observed. The cause for the departure lies in the modelling representation. The volumes of the containment relevant to the leak rate investigations (SP 5ooo) can now be exactly determined. The critical swelling pressure does not appear to be as well defined a transition point as was shown by the first measurements.

## 5. Next Steps

Several points which will be included in further investigations are:

- measurement of the enthalpy contours with the core barrel in place, and checking out of the test loop for experiments on the RPV internals
- a series of blowdown experiments utilizing a feed water check valve of NW 2oo (SRV 200)

- additional experiments to accompany the previously performed blowdown experiments with the steam isolation valve (DIV Part II)
- blowdown experiments with a feed water check value of NW 35o (SRV 35o)

In addition thermal shock experiments on the HDR pressure vessel as well as containment blowdowns are in preparation. Whether or not these experiments can be performed in 1978 is still not decided. For all experiments both calculations and comparisons of these calculations with experimental measurements will be performed.

## 6. Relation with Other Projects

1. Investigation of the Phenomena Involved in the Depressurization of Water Cooled Reactors — RS 16/2
2. Investigation of the Phenomena Occurring within a Multicompartment Containment after Rupture of the Primary Cooling Circuit in Water Cooled Reactors. — RS 5o
3. Reaction and Impingement Forces on Components and Structures Caused by a Two Phase Jet Discharged at Primary Pipe Breaks. — RS 93
4. Joint Reactor Safety Experiments in the Power Station of Marviken, Sweden. — RS 33
5. Ultrasonic Pulse-Echo Spectroscopy in Ultrasonic NDT. — RS 54
6. Investigations on a Continuously Operating System for Crack Growth Surveillance in Pressure Vessels, Part IV: Further Development of Acoustic Emission with Regard to the Application at the Reactor. — RS 31/3

   Nondestructive Inservice Inspection for Reactor Pressure Vessel with Eddy Current Methods. — RS 89
7. Development of Non-Destructive Testing Methods for In-Service Inspections of Reactor Pressure Vessels. — RS 27/2

## 7. Reference Documents

- Paper for IEA - SLT - ERD - Expert Group on Reactor Safety "Safety Investigations in the Decommissioned HDR-Plant," April 1975 (in English)
  Paper: HDR Safety Program - General Program - Status as of December 1975 Gesellschaft für Kernforschung (in English)
- Quarterly reports: in the IRS-Research Report series (in German) Reports IRS - F - 21, 23, 25 - 27
- Annual report : IRS - F - 24 (in English).
- Annual report : IRS - F - 25

## 8. Degree of Availability

Unrestricted distribution.

1587

| Berichtszeitraum/Period<br>1.1.77 - 31.12.77 | Klassifikation/Classification<br>20 | Kennzeichen/Project Number<br>RS 139 |
|---|---|---|
| Vorhaben/Project Title<br>Programmentwicklung zur mehrdimensionalen Kontinuumsmechanik<br><br>Code Development in multidimensional continuummechanics | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>INTERATOM<br>Bergisch Gladbach 1<br><br>713o |
| Arbeitsbeginn/Initiated<br>1.9.74 | Arbeitsende/Completed<br>31.12.79 | Leiter des Vorhabens/Project Leader<br>H. Banasch (Koord.) |
| Stand der Arbeiten/Status<br>continuing | Berichtsdatum/Last Updating<br>31.12.77 | Bewilligte Mittel/Funds<br>2.251.900,-- DM |

1. General Aim

Developments of methods and computer codes for calculations in multi-dimensional hydrodynamics coupled with multi-dimensional structural dynamics.

2. Particular objective

Development of a computer code treating "KOntinuumsmechanik mit Euler-Lagrange-Koordinaten und Schalentheorie (KOELSCH)". Validation of developed methods and codes via available experiments. Participation in the COde VAlidation-Programme at Euratom, Ispra (COVA).

3. Research Programme

3.1 ELK

Computational methods for multi-dimensional hydrodynamics with "Euler-Lagrange-Koordinaten" (for the three-dimensional case only Eulerian coordinates).

3.2 BLASE

Methods for the description of expansion of bubbles (vapour bubbles or bubbles of explosive charges) in two dimensions.

3.3 SCHAL/PLAT

Methods for the two-dimensional structural dynamics of shells and plates.

3.4 EOS (Equation of State)
Treatment of material properties.

3.5 KOELSCH
Main programme for coupling the various routines.

3.6 INKOEL/OUTKOEL
Routines for input, output and plotting.

3.7 Validation, COVA-Programme at Euratom/Ispra
Calculation of appropriate experiments and participation in the COVA-Programme at Euratom/Ispra

4. Experimental Facilities, Computer Codes
In section 3 there are given the names of the various routines which will be developed and used.
For participation in the COVA-Programme at Euratom/Ispra the code ARES is used.

5. Progress to DATE
ad 3.1 3D-versions in cylindrical and toroidal coordinates have been written and tested for the hydrodynamics part of the problem.
The 2D-modules for hydrodynamics and structural dynamics have been coupled and tested.

ad 3.3 The treatment of structural dynamics has been improved by including shear stresses and bending moments (SCHAL)

ad 3.4 The treatment of elastic-plastic behaviour of shells has been tested by calculating the pressure wave propagation in a fluid filled thinwalled pipe.

ad 3.5 Concepts for an optimum coupling of all available physical moduls have been reviewed,

ad 3.7 The Belgonucleaire/Ispra experiment E114 has been calculated with the 2D hydrodynamics module.
An INTERATOM-test of pressure wave propagation through a fluid filled elbow has been calculated with the 3D hydrodynamics module.
Within the frame of the participation in the COVA-programme at Ispra the ARES code has been continuously updated according to the improvements achieved at INTERATOM. An IBM-version of ARES in single precision has been established and transfered also to KFK. ARES-calculations have been performed for the COVA-tests IT5 and IT7.

6. Results

ad 3.1 The first part of an interim report (see item 9) has been completed which describes the theory of the 1D-, 2D- and 3D-hydrodynamics code versions.

ad 3.2 The treatment of the expansion of a high pressure bubble surrounded by compressible fluid is described in the interim report (item 9).

ad 3.3, 3.4 The end constraints of a fluid filled pipe through which a pressure wave is propagating, induces a multi-axial stress situation and thus very much reduces the strains achieved.

ad 3.7 The calculation for the Belgonucleaire/Ispra experiment E114 with the 2D hydrodynamics module shows good agreement with the experimental results and with the results obtained by the Lagrangian code ARES.
The calculation for the INTERATOM-test of pressure wave propagation through a fluid filled elbow with the 3D-hydrodynamics module shows good agreement with the experimental results.

Within the frame of the participation in the COVA programme at Ispra it turned out that special attention has to be paid to the single precision IBM-version of the REZONE routine in the ARES-code. The ARES-calculations for the COVA-tests IT5 and IT7 were taken to rather long problemtimes (12 msec for IT5 and 3o msec for IT7). The results compare well with the results obtained by other codes (ASTARTE, REXCO-H).

7. Next Steps

The treatment of the structural dynamics, the coupling of hydrodynamics and structural dynamics, as well as several test cases will be documented in part 2 and part 3 of an interim report. The coupling of the hydrodynamics and structural dynamics routines will be further tested.
A concept for optimum coupling of all physical routines (KOELSCH) will be selected and tested.
Within the frame of the participation in the COVA-programme at Ispra plot routines will be added to the IBM-version of ARES. Further tests of the COVA series will be calculated with this code. Special attention will be paid to further improvement of the treatment of perforated structures within containment codes.

8. Relation with other Projects

9. References

L. Lange, E. Blokker, S. Hentschel
Zwischenbericht des Forschungs- und Entwicklungsauftrages RS 139 - Programmentwicklung zur mehrdimensionalen Kontinuumsmechanik. KOELSCH (KOntinuumsmechanik in Euler-Lagrange-Darstellung und SCHalentheorie). Teil 1: Fluiddynamik.INTAT 77.67, Mai 1977

1o. Degree of Availability of the Reports

On request by INTERATOM GmbH, 5o6o Bergisch Gladbach 1

1591

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.7.77.- 31.12.77 | 20 | RS 272 |

| Vorhaben/Project Title | |
|---|---|
| APRICOT, Phase B<br><br>APRICOT, phase B | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>INTERATOM<br>Bergisch Gladbach 1<br><br>OE 7130 |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.7.77 | 30.9.78 | H. Banasch (Koord.) |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | 31.12.1977 | 116.983,-- DM |

## 1. General Aim

Within the APRICOT-Program (Analysis of PRImary Containment Transients) a series of benchmark problems are to be calculated under international participation with continuum mechanics codes designed for the description of the dynamic behaviour of the primary containment of sodium cooled fast reactors. Their results will be compared with each other and with experiments specially chosen for this purpose.
The knowledge gained here can be used to solve engineering problems more economically and to increase confidence in usage of internationally validated codes for safety cases.

The APRICOT-Program was initiated and organized by USERDA.

## 2. Particular Objectives

The participation in the APRICOT-Program takes place by treating the proposed benchmark problems with the INTERATOM code ARES. Results and comparisons with experiment will be evaluated by a group of independent experts from which an increased confidence in the codes will result.
Moreover the participation in the APRICOT-program will give access to foreign experimental results and will permit to become acquainted with the capabilities of the other codes.

3. Research Program

3.1 Ten organisations from five countries participate in the APRICOT-program with eleven computer programs. For comparison with experiments results from explosion tests of the Code-Validation-Program in England and Ispra (EURATOM) and from explosion tests of the Stanford Research Institute/USA (SRI) are available.

3.2 The APRICOT-program consists of two phases. Phase A contained three tasks. This phase, where INTERATOM participated in cooperation with GRS (Köln) is terminated. For phase B (considered here) tasks four through seven were defined. With ARES the tasks six and seven will be treated and the results will be compared with the corresponding SRI-experiments.

Task six consists of the calculation for an explosion test in a water filled thin walled cylinder. It contained a thick walled inner tank made of steel. Pressure and strain gauges which were fixed to the cylinder walls produced experimental curves which can be compared to the calculations.

Task seven treats a similar SRI-experiment which differs only in the inner container, which consists of lead with a thin surrounding shell of aluminium.

3.3 The geometrie layout of these experiments is modelled with ARES which calculates the loading and deformation history of the tanks.

3.4 The calculated results are compared with the experimental ones and with those of the other codes.

4. Experimental Facilities, Computer Codes

The Interatom-code ARES is a two-dimensional (cylinder symmetrical) compressible Lagrange program which works by the finite difference method. If pertinent initial and boundary conditions are given, the code calculates pressure wave propagation, hydrodynamic flow and the structural deformation by means of the programmed differential equations, equations-of-state and stress-strain behaviour of the materials. As results coordinates, velocities, pressures, stresses, strains, energies and other variables can be plotted as functions of space and time.

5. Progress to Date

Input data for tasks six and seven were set up and first calculations were done to obtain an optimal mapping of the experimental structure within the code.

6. Results

Tasks six and seven were carried to about a quarter of the desired problem time. The corresponding local pressure and strain histories are at hand.

7. Next Steps

The APRICOT problems six and seven will be calculated up to the desired problem time an a suitable modelling. The calculated results will be compared with the results of the SRI-experiments.

8. Relation with other Projects

The work described here is considered as an important support and extension of the international code validation work, where Interatom participates in the code validation program in England (RS 272) and in Euratom /Ispra (RS 139, RS 162).

## 9. References

a) W. Kellner
APRICOT-Problems 1 and 2
Interatom-Notiz 7o.158.5 (1976)

b) Meier (GRS/Köln), Kellner, Doerbecker (IA)
ARES-computations for APRICOT 3
Interatom-Notiz 7o.245.3 (1976)

c) The APRICOT Program: Comparison and bench marking of computational methods for analysis of LMFBR structural response to HCDA pressure loads. Phase 1 report. SAN-1112-1, Oct. 77
1977

## 1o. Degree of Availability of the Reports

On request

1595

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 20 | RS 235 |
| Vorhaben/Project Title<br>Theoretische und experimentelle Untersuchungen zur Bestimmung der Strömungs- und Temperaturfelder im oberen Reaktorplenum bei transienten Vorgängen<br><br>Theoretical and Experimental Investigations on Transient Velocity- and Temperature Fields within the Reactor Outlet Plenum | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>INTERATOM<br>Berg.-Gladbach 1<br><br>OE 7010, 8230 |
| Arbeitsbeginn/Initiated<br>1.1.77 | Arbeitsende/Completed<br>28.2.79 | Leiter des Vorhabens/Project Leader<br>H.Ing. grad. Banasch |
| Stand der Arbeiten/Status<br>Continuing | Berichtsdatum/Last Updating<br>31.12.77 | Bewilligte Mittel/Funds<br>1.158.168,-- DM |

1. General Aim

Safety Design of Sodium Cooled Breeder Reactors against Transient Thermal Stresses.

2. Particular Objectives

Development of an Experimentally Proved Computer Program Concerning Time Dependent Temperature Fields within a Reactor Outlet Plenum.

3. Research Program

3.1 Theoretical Investigations

3.1.1 Computation of a Representative Transient Behavior of Fast Breeder Reactors by the Use of already Existing Computer Codes

3.1.2 Discussion of Fundamental Flow Model Design by the Use of an Elementary Code

3.1.3 Literature

3.1.4 Development of a New Computer Program

3.1.5 Coordination of Model Parameters and Experimental Results

3.2 Experimental Investigations

3.2.1 Discussion of the Experimental Program and the Specified Parameters

3.2.2 Reconstruction of the Existing Outlet Region Feature Model

3.2.3 Determination and Providing Measuring and Control Technique

3.2.4 Instrumentation

3.2.5 Time of Beginning of Operation

3.2.6 Test Procedure

3.2.7 Data Processing

## 4. Experimental Facilities, Computer Codes

Test will be run using the 1:4 scale model. A specific reconstruction of this test facility is carried out. To start with theoretical investigations input data from projected experiments are figured out for the computer code TIRE. To make this program compatible with the geometry of the special test facility some parts of the TIRE code are going to be changed.

## 5. Progress to Date

ref. 3.1.1 First informations about the charakteristic transient SNR 2 outlet plenum behavior could be evaluated from preliminary TIRE calculations.

ref. 3.1.2 A quantitative description of the different parameters needed to properly design the test facility is completed. Computer runs concerning different tests are going to be specified.

ref. 3.1.3 Two digital programs (MIX, VARR II-A) are discussed and analysed where by the program MIX was made compatible with the CDC-Cyber 172 computer.

ref. 3.2.1 The list of experimental investigations is completed. All parameters necessary to design the test facility as well as the loop system are marked out. The facility is shown in Figure 1.

ref. 3.2.2 Tests areto be performed using water with a 1:4 scale model. The capacity of the inlet flow volume has been reduced so that inlet temperature transients could be developed. Plenum liner and suppressor plate of the upper Plenum were supplied with boreholes similar to SNR 300 design. Furthermore a cylindrical liner was installed on the top of the suppressor plate to stop any fluid flow in this region.

ref. 3.2.3 The measuring and control technique is largely at hand, and electrically cabeledand wired to the data aquisiton system.

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ref. 3.2.4 Work is done concerning the installation of the measuring technique of the flow model. Thermocouples with a response time of 8 msec are installed at the plenum liner, the vesselwall and the outlet nozzel. Further thermocouple instrumentation will be done in a 1/3 sector of the inner part of the upper plenum. A outlet plenum thermocouple map is going to be sketched.

6. Results

ref. 3.1.1 The characteristical transients in the upper plenum of a commercial breeder reactor have been calculated using the existing program TIRE for the following three basic cases: scram from 100% power with a maximum decay heat, scram from 100% power with a minimum decay heat, scram from 30% power with a minimum decay heat.

ref. 3.2.1 Different reactor transient conditions to be simulated in the experimental program can be achieved with the model test conditions. The experimental program includes variations in test conditions from the basic reactor trip simulation. Flow rate and density changes can be varied to study the effect of Archimedes and Froude number.

ref. 3.2.3 The operational test concerning the measuring and control technique of the test facility is done.

7. Next Steps

ref. 3.1.2 TIRE code computations for test design
ref. 3.1.3 Discussion of computor code MIX.
ref. 3.1.4 Discussion of an new computer program
ref. 3.2.4 Flow model instrumentation

8. Relations with Other Projects

Till this time there are no relations with other projects.

9. References

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10. Degree of Availibility of the Reports

–

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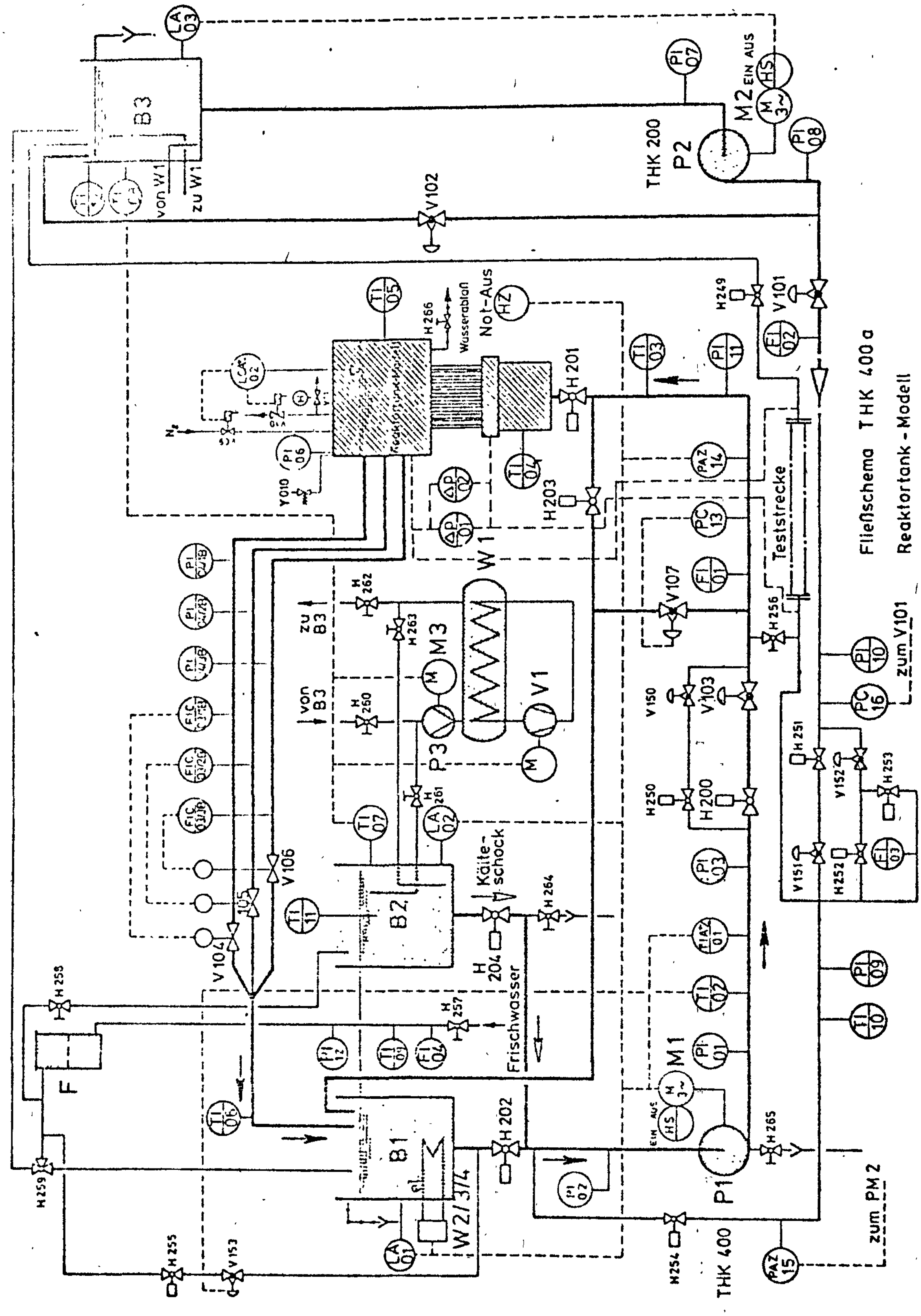
Fließschema THK 400 a
Reaktortank - Modell
Teststrecke
Not-Aus
Wasserablaß
Kälte-schock
Frischwasser
zum V101
zum PM 2
THK 200
THK 400
B1
B2
B3
W1
W2/3/4
P1
P2
P3
M1
M2
M3
V1
F

Fig. 1

| Berichtszeitraum/Period<br>1.6.77 - 31.12.1977 | Klassifikation/Classification<br>20 | Kennzeichen/Project Number<br>RS 269 |
|---|---|---|
| Vorhaben/Project Title<br>Untersuchung zur tankinternen Notkühlung unter Berücksichtigung von Sieden<br><br>Examination of the in-tank emergency cooling with regard to boiling | | Land/Country<br>FRG<br><br>Fördernde Institution/Sponsor<br>BMFT<br><br>Auftragnehmer/Contractor<br>Fa. Interatom<br>Bergisch Gladbach 1<br><br>OE 8230, 7120 |
| Arbeitsbeginn/Initiated<br>1.6.77 | Arbeitsende/Completed<br>30.4.79 | Leiter des Vorhabens/Project Leader<br>H. Banasch (Koord.) |
| Stand der Arbeiten/Status<br>continuing | Berichtsdatum/Last Updating<br>31.12.1977 | Bewilligte Mittel/Funds<br>406.265,-- DM |

. General Aim

Higher power densities are aimed at in larger breeder plants. This can lead to sodium boiling in the case of emergency cooling. Short-term boiling may be permitted in the fuel elements, if it can be proved that the capability of the boiling elements is guaranteed for the boundary conditions of the in-tank natural circulation.

The operational capability of the emergency cooling has till now been proved by the INTERATOM-computer code NOTUNG / 1 /. As this program is restricted to the simulation of single phase flow processes an extension of the code is necessary to take account of boiling phenomena in single fuel elements under natural circulation conditions.

The aim of this project is to develop a simplified boiling module which together with NOTUNG will enable predictions regarding flow and temperature fields that may arise in emergency cooling phases if the local boiling temperature of the coolant is exceeded in the bundle area.

A feasibility study has to be carried out in parallel with the above to produce an experimental model which can be used in boiling experiments at a later date.

2. Particular Objectives

- Evaluating of literature concerning previous boiling experiments
- Development of a theoretical model to calculate natural circulation boiling phenomena
- Verifying of the theoretical model by means of present boi-

ling experiments

- Insertion of the boiling model into the code NOTUNG
- Production of a feasibility study for boiling experiments under natural circulation conditions

3. Research Program

3.1 Literature study
Evaluating of previous boiling experiments. Critical examination of the applicability of these experiments.

3.2 Theoretical model
Laying down of the program's structure

3.3 Determination of the testparameters
Similarity considerations. Laying down of the test parameters

3.4 Report
Documentation of interim results

3.5 Development of concepts for the construction and measuring techniques. Constructional design selection of measuring techniques

3.6 Program development
Production of the computer code

3.7 Calculations
Checking of existing boiling experiments

3.8 Re-working of the program
Program modification resulting from the information gained from the checking

3.9 Module insertion
Insertion of the boiling module into NOTUNG

3.1o Final report
Description of the program. Evaluation of the results obtained and analysis to see if more tests are necessary with the aim of further modifying the program.

4. to 3.9 Experimental Facilities, Computer codes
The flow and temperature fields in the case of emergency cooling are calculated by the computer code NOTUNG. This program is restricted to the simulation of single phase flow processes.

5. to 3.1 Progress to Date
The results of the boiling experiments have been examined for their suitability in terms of code verification.

to 3.2 The equations, which are necessary for the calculation of a natural convection circuit (with and without boiling) have been drawn up. These equations have been inserted into the program flow chart.

to 3.3 The physical variables, which have the most influence in determing the natural circulation of a two-phase flow in the reactor, have been examined. From these the main characteristics of a test section were deduced.

to 3.4 An interim report / 2 / has been produced which closes the first phase of the working programme.

6. to 3.1 Results
Existing boiling experiments have been carried out almost solely for high power densities and forced convection. They, on the whole, aim at the simulation of transient boiling phenomena which are not comparable to the prevelant natural circulation conditions. With the help of experimental results several authors draw up equations which can determine the two-phase pressure drop in sodium flow. These equations can be used for the determination of the theoretical model. One author / 6 / reports on boiling experiments with natural circulation. The power densities used in this connection correspond to those

values which are typical during the emergency cooling.

to 3.2 The calculation scheme is almost the same as in NOTUNG / 1 /. For each time step $\Delta t$ a heat and power balance is carried out using explicit difference method. The current state is based on the temperature and the flow rate distributions at the previous time step. The boiling model calculates the vapour quality in the heat balance and the pressure-gradient of the two-phase flow in a power balance. For the calculation of the two-phase pressure drop the correlation of Lockhart-Martinelli / 3 / is used. The relationship for the pressure drop multiplier, used for the present by us, is an experimental correlation developed by A. Kaiser and W. Peppler / 4 /. The vapour fraction which determines the higher buoyancy due to boiling is described by a correlation from Levy / 5 /.

to 3.3 In order to have the same main limiting qualities in the reactor and in the test section, the following demands must be fulfilled:

- the same heated lengths and geodetical heights
- the same time scale
- the same axial power density distribution
- the same system pressure
- the same share of the pressure loss in the pin bundle, the core bypass and in the throttles
- the minimum number of pins in the pin bundle is 37
- a larger wrapper tube wall thickness, that the ratio of power input into the pin bundle to the heat absorption of the core bypass is the same

The investigation is shown in / 2 /.

7. <u>Next steps</u>

to 3.5 Preparation of the constructional and measuring draft

to 3.6 Program development

8. Relation with other Projects

Under the title "Lokales und integrales Sieden in SNR-Bündeln", the IRE (Institut für Reaktorentwicklung, Kernforschungszentrum Karlsruhe, 75oo Karlsruhe, Postfach 364o) conducts investigations into decay-heat-removal under natural convection conditions for a 37-pin bundle.

9. References

/ 1 / F. Rösgen, F. Timmermann, H. Vossebrecker
Wärmetechnische Berechnungen zur Notkühlung des SNR-3oo
KTG-Fachtagung der Fachgruppen Reaktorsicherheit und Thermo-Fluiddynamik in Stuttgart, Januar 1975

/ 2 / K. Brockmann, U. Willrodt
Zwischenbericht des Förderungsvorhabens RS 269: Untersuchung zur tankinternen Notkühlung unter Berücksichtigung von Sieden,INTAT-Nr. 78.6

/ 3 / R. W. Lockhart, R. C. Martinelli
Proposed Correlation of Date for Isothermal Two-Phase Two-Component Flow in Pipes
Chem. Ing. Progress 45, 1949, S. 39 - 48

/ 4 / A. Kaiser, W. Peppler
Sodium Boiling Experiments in a Seven-Pin Bundle: Flow Patterns and Two-Phase Pressure Drop
Nuclear Engineering and Design 43, 1977, S. 285 - 293

/ 5 / S. Levy
Steam Slip, Theoretical Prediction from Momentum Model
J. Heat Transfer 82, 196o, S. 113 - 124

/ 6 / A. Kaiser, W. Peppler, M. Straka
Decay Heat Removal from a Pin Bundle
International Meeting on Fast Reactor Safety and Related Physics
October 6 - 8, 1976, Chicago

10. Degree of Availability of the Reports

/ 1, 2 / Fa. Interatom
5060 Bergisch Gladbach 1
Friedrich-Ebert-Str.

/ 6 / Institut für Reaktorentwicklung
Kernforschungszentrum Karlsruhe
7500 Karlsruhe
Postfach 3640

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| Jan.1,77 - Dec.31,77 | 20 | RS 242 |

| Vorhaben/Project Title | Land/Country |
|---|---|
| Verification on Computer Codes for Tank and Tankcomponents with the UK Containment Code Validation Tests<br><br>Verifizierun von Rechenprogrammen für Tank und Tankeinbauten in Verbindung mit dem englischen COVA-Programm (Explosion Containment Code Validation Tests) | FRG |
| | Fördernde Institution/Sponsor: BMFT |
| | Auftragnehmer/Contractor: Kernforschungszentrum Karlsruhe |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| Dec. 1,1976 | Nov. 3o,78 | H. Knuth |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | Dec. 31, 77 | 912.010,-- DM |

1. General Aim

Within the frame of safety considerations for sodium cooled fast reactors hypothetical power excursions have to be treated. To study the mechanical impacts of these excursions on the primary containment computer codes are used. A verification of the computer codes used is necessary for the safety evaluation of the reactor vessel. There has not been a fully validated computer code until now in the Federal Republic of Germany. An agreement has been concluded between KFK in Karlsruhe and the United Kingdom Atomic Energy Authority (UKAEA) providing for the exchange of R&D results. The agreement stipulates inter alia that all the experimental results from the English code validation program are transferred to Karlsruhe. The experiments are performed with water filled vessels with and without internals. The power excursion is simulated by an explosive charge. A selection is made among the roughly 4o main experiments. A comparison is performed between the experimental and calculated results. So statements can be made on the reliability and accuracy of the theoretical computer models used. Necessary modifications have to be indicated. Results of these activities are also integrated in the licensing procedure for the SNR 3oo fast breeder reactor.

2. Particular objectives

The British COVA program includes experiments with short tanks (pool type design) and with long tanks (loop type design). Since the SNR line is based on the loop type, the advance calculations and recalculations will concentracte on experiments with long tanks (see Annex.)

3. Research Program

In 1977 computations and evaluations were made for the following experiments from test programs under way in Great Britain:

WT 5: long rigid tank,
WT 7: long thin tank
WT 8: long thin tank with different charge weight,
WT 9: short rigid tank, rigid inner,
WT 12: short rigid tank, tank inner, rigid diagrid,
FT 4: short rigid tank
FT 6: short thin tank
FT 7: short thin hemispherical-bottomed tank

4. Test Facilities, Computer Codes

A german working group was established which is made up of representatives of KFK, GRS and Interatom. Advance calculations and recalculations were performed on selected experiments on a work sharing basis, using above all the ARÉS code developed by Interatom (2-dim.Lagrange-code).

Work Performed, Results Obtained

KFK

5. Work Performed

Setting up programs for
- reorganizing ARES 3B data by the program;
- plotting major variables (pressure, pulse, strains, etc.) versus time and representation of fields (mesh grid, pressure, velocity)
- reading and plotting experimental data stored on magnetic tape.

This work became necessary since the standard plotting routines for ARES cannot be used on the KFK computer system. Advantage was taken of this situation in order to adapt the plotting programs to the particular requirements of ARES verification.

Recalculation of the COVA experiment FT 4 (short rigid tank without internals) up to roof impact. Eight parameter calculations in total were performed in which the mesh size, the predeformation of meshes around the charge, the transferable tensile strain in the fluid, and modelling of the cover gas were subjected to variations.

Extension of the ARES 3B program by implementing the shell theory from ARES 4A.

First computations of the COVA experiment FT 6 (short thin tank without internals).
The work was supplemented by
estimates based on calculations in order to determine an effective wall thickness of the thin shells with circumferential variations of the wall thickness;
development of a computer program allowing interpolation of dynamic stress-strain curves;
estimates of the influence of uncertainties regarding load, geometry and material data, on permanent deformation, using a simple example.

3. Results

The completion of different plot programs allows to adopt in the graphical representation of results both the conventions of the UKAEA report and to represent in an overall diagram the calculated and experimental results.

Computations relating to the experiment WT5 and based on the IBM version ARES 3B (KFK) and the CDC version ARES 4A (IA) essentially yield the same results; however, it has to be considered that these computations are of a pure hydrodynamic nature. Provisional comparisons with the nominally identical COVA experiment IT 5/3 show that, on an average, the peak pressure and pulse (at 2 ms) is underestimated by 22% and 8%, respectively, at the botoom and by 42% and 19%, respectively, at the tank wall.

A first comparison with the experiment FT 4 (1) also yields a considerable underestimation of peak pressures. Until about 1.5 ms the pulse curves show a relatively good agreement; however, beyond about 2 ms, the pulse is clearly overestimated.

The test computations referring to the experiment FT 6, using the ARES 3B version extended with respect to the shell theory, have not yet provided satisfactory results; the results suggest that the extended ARES 3B version has still some shortcomings. In these calculations interpolated material data were used which had been determined for EN 321 with the help of the interpolation program already mentioned. It is based on the visco plastic material law obtained by a fitting process and characterzized by a logarithmic dependence of the strain rate.

Recalculations of dynamic tensile tests showed a good agreement of the stress-strain curves, in particular at the lower ($3.7x10^{-3}s^{-1}$) and the upper ($190s^{-1}$) limits of the strain rate range.
The parameter studies on the influence on permanent deformation exerted by little uncertainties in material data, etc., yielded a considerable uncertainty in plastic strain on the assumption of a pessimistic combination. However, the transferability of these results to COVA experiments with thin-walled tanks has to be regarded with reservations since in that case the load and deformation conditions are much more complex and above all the load and structural deformation are coupled to each other, which is not included in the simple model used in the parameter study.

GRS

5. Work Performed

Since the working group is not in a position to evaluate all the British COVA experiments, it was agreed that only those experiments will be evaluated which are of significance for the German reactor line. These include the tests with a long tank, with a hemispherical bottom or with internals such as shield tank, dip plate or grid plate. GRS started evaluating the test WT 7 (long flexible tank, 2 oz LDE charge). The calculations are performed on a CDC computer system Cyber 175 at Interatom in Bensberg, using the ARES computer code. Since ARES offers to the user different numerical methods to treat a problem and since some parameters have to be defined arbitrarily in the calculation, a parameter study was performed at first. Starting from a reference calculation of the test WT 7, the following parameters were varied systematically (see COVA note 4/76):

the Lagrange grid,
the rezoning frequency,
the acceleration method for fluid elements,
the time integration method,
the artificial viscosities

After standard values for the respective parameters had been determined in this way, the test WT 7 was calculated. A particular difficulty was encountered in modelling the upper shell region sliding in a lip (see COVA note 4/77). Subsequently, calculation started with the test WT o8 (long flexible tank with 4 oz LDE charge).

A special feature of this test is the phase in which the rising water column gets reflected by the roof ("roof impact"). At this point the distortions of the Lagrange grid are particularly large, resulting in numerical inaccuracies (see COVA-note Nr.1o/77). Since, in the initial phase, a satisfactory agreement was not found for test WT o8 between the experimentally measured and calculated deformations occurring in the outer shell, the studies related to the extent to which the shell theory implemented in ARES is suited to describe the experiment (see COVA note No. 9/77).

It was also examined, whether the nominal data applicable to the stress-strain diagram for EN 321 are valid for the COVA tanks used (see COVA Note No.18/77). In the calculation relating to the test WT o8 the steel data were subjected to a parameter variation (see COVA Note No.2o/77). However, the question whether the steel data are applicable, cannot be answered until the results of the respective material investigations performed in the United Kingdom will be known. At the end of the period of reporting the test WT 12 was calculated. A documentation has not yet been prepared.

6. Results

The parameter variation showed that the grid distortion exerts by far the highest influence on the accuracy of computation. For this reason, all the calculations have to be planned such that grid distortions remains as low as possible. After the calculation relating to the test WT 7 had been terminated, the British experimentors told us that the charge had burnt incompletely. Consequently, the strain curves measured in the test cannot be used to verify the computer program. The same effect was found in the experiments WT 8 in the United Kingdom and IT 7 in Ispra. The tests were subsequently repeated. Since the British test results were not communicated until November, a final comparison has not yet been possible. Although with respect to the hydrodynamic variables such as pressure and pulse, adequate agreement existed between the calculation and the experiment, considerable deviations occured between the calculation and the experiment, considerable deviations occured between the calculation and the measured strains. An examination of the shell theory used in ARES showed that the model equations should sufficiently describe these phenomena. However, some further theoretical investigations showed that it is doubtful whether the stress-strain diagrams for steel EN 321 so far used will satisfactorily describe the dynamic behavior of the COVA-tanks. Additional material investigations were agreed, the results of which will be available during the first quarter of 1978. Until that date it will not be possible to make a final judgement concerning the quality of the computer code.

IA

5. Work performed

Calculations were performed for the tests WT 5 and IT 8 using the ARES program. The results of calculations were documented in the relevant reports. Supplementary studies were performed on the following items:

a) Energy balance in the ARES calculation of the experiment WT 5.
b) Calculation of the elastic-plastic stresses in the ARES computer program.
c) Estimate of the stress-strain values for the IT 9 tank.
d) Treatment of shells by ARES.

6. Results

In the comparison calculation performed between KFK and IA and relating to the test WT 5 the older KFK program version rewritten to fit IBM machines essentially provided the same results as the newer IA version developed for CDC machines. It should be noted that in WT 5 only the hydrodynamic part of the computer program is tested. A preliminary comparison with the nominally identical Ispra test IT 5 (3) shows that the ARES computations underestimate by 8% on an average the pulses occurring at the bottom and by 2o% on an average those occurring at the tank wall.

In the test IT 8 considerable differences appear between the calculation and the experiment regarding the ultimate strains. The maximum circumferential strain in the experiment is about 8%, in the calculation about 5%. The measured meridional strains show pronounced maxima in the range of the hemispherical bottom and in the upper tank region, which are not reproduced by the calculation. The deviations are probably due to the fact that the material properties of the tanks, especially the hemispherical bottom, are not known sufficiently well.

If one compares the calculations performed with the ARES and ASTARTE programs on the same test (IT 8 and FT 7, respectively), it can be found that despite the differences in input data and model assumptions the calculations agree better with each other than the calculations do with the relevant experiment. The uncertainty in material data is also a feature of the other COVA tests with flexible tanks. To elucidate this point, the material data were subject to examinations in the United Kingdom, using tensile test specimens.

7. Planned Work

Summarizing, the following can be said about the status of work so far performed: Calculations have been performed with respect to seven experiments using the ARES program. These experiments were selected by the existing working group after discussions had been held. Difficulties with the resulting delays of the planned working schedule have been due to two main reasons. The first reason lies in the fact that in quite a number of experiments the explosive charge did not completely burn. For this reason, these experiments were repeated. The impacts on current work of the group have been that a number of calculations performed on the basis of non-relevant experimental results had to be calculated again using the results of the repeated experiments.
The second reason lies in the insufficient validation of material data for the experiments. This required additional activities on this subject and, at the same time, is the reason why until now a final report has not been written. In spring 1978 the material data will be available on the tank material used.

Then additional calculations will be performed using these data. The final reports on the processed tests WT 5, WT 9, FT 4, and FT 6 will be available by mid 1978. Only then a preliminary statement can be made on the ARES computer program. By the end of 1978 the tests WT 8, WT 12 and FT 7 will be terminated by preparation of a final report. Until that date some additional experiments will have to be recalculated. To obtain a validated statement on the computer programs for the tank and tank internals to be used for the LMFBR, another 8 to 1o experiments are considered necessary. Of the roughly 4o UK experiments 15 to 17 experiments would then have been handled. This means an extension of one year, appreciated by the working group, of the date of termination of work presently fixed at November 3o, 1978.

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9. Literature

COVA-Notiz 4/76, W. Salz
Eine Parameterstudie zu ARES, Int.Bericht GRS I-1, January 77
COVA-Notiz 1/77, H.Schäfer
ARES-calculation of the COVA-experiment WT 5, IA Notiz Nr.7o.4oo:4, Jan. 3o,1977
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COVA-Notiz Nr. 4/77, W. Salz
Theoretical calculation report on COVA-experiment 1.3o7o621 (WT 7) fixed March 22,76, Int.Bericht GRS I-4, Feb.77
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COVA-Notiz Nr. 9/77, W. Salz
Die Schalentheorie in ARES, April 77
COVA-Notiz Nr. 1o/77, W. Salz
Theoretical calculation report on COVA-experiment 1.3o8o621 WT 8, ARES-Calculation
Int.Bericht GRS I-5, May 77
COVA-Notiz Nr. 13/77, H. Schäfer
ARES-calculation for experiment IT 8, June 77
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COVA-Notiz Nr. 15/77, Sept. 2o,77, Y.S. Hoang
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COVA-Notiz Nr. 16/77, W. Salz
Protokoll der COVA-Besprechung vom 22.9.77 bei IA Bensberg
COVA-Notiz Nr. 18/77, W. Salz
Die Materialdaten für die COVA-Experimente WT 8/IT 7, Oct. 77
COVA-Notiz Nr. 19/77, T. Malmberg
Effects of uncertainties in load, geometry and material data on the permanent plastic deformation: A simple example, Oct.18,77
COVA-Notiz Nr. 2o/77, W. Salz
Nachrechng. des COVA-Experiments WT 8 mit ARES, Nov. 77
COVA-Notiz Nr.21/77, H.Doerbecker, the treatment of shells in ARES, Nov.5,77

## Test Program

| Test | Description | Test | Description |
|---|---|---|---|
| WT 1 | short rigid tank, HE (same as FT 1) | FT 1 | short rigid tank, HE (same as WT 1) |
| WT 2 | repeat of WT 1 | | |
| WT 3 | repeat of WT 1 | FT 2 | repeat of FT 1 |
| WT 4 | long rigid tank, HE | FT 3 | repeat of FT 1 |
| WT 5 | long rigid tank, LDE | FT 4 | short rigid tank, LDE |
| WT 6 | short thin tank (same as FT6) | FT 5 | short rigid hemispherical bottomed tank |
| WT 7 | long thin tank | | |
| WT 8 | repeat of WT 7 with different charge weight | FT 6 | short thin tank (same as WT 6) |
| WT 9 | short rigid tank, rigid inner | FT 7 | short thin hemispherical-bottomed tank |
| WT 1o | short rigid tank, thin inner | FT 9 | short rigid tank, thin inner |
| WT 11 | long rigid tank, thin inner, rigid diagrid | FT 1o | short rigid tank, rigid inner, perforated diagrid |
| WT 12 | short rigid tank, thin inner, rigid diagrid | FT 11 | short rigid tank, thin inner, perforated diagrid |
| WT 13 | long rigid tank, thin inner, perforated diagrid | FT 12 | short thin tank, thin inner, perforated diagrid (same as WT 15) |
| WT 14 | long thin tank, thin inner, perforated diagrid | FT 13 | side supported rigid diagrid |
| WT 15 | short thin tank, thin inner, perforated diagrid (same as FT 12) | FT 14 | side supported perforated diagrid, thin inner |
| WT 16 | long thin tank, thin perfor. inner, perforated diagrid | FT 15 | side supported rigid diagrid, lengthened thin inner to allow split level coolant |
| WT 17 | short rigid tank,touching NS, inner + outer constraining tanks | FT 16 | short rigid tank, spaced NS, outer constraining tank |
| WT 18 | short rigid tank, spaced NS, outer constraining tank (same as FT 16) | FT 17 | short thin tank, spaced NS, outer constr. tank |
| WT 19 | long rigid tank, thin inner, perfor.diagrid, perfor.dip plate | FT 18 | side supported perf. diagrid, spaced NS, outer constraining tank |
| WT 2o | long ridig tank, thin perfor. inner, perf.diagrid, perf. dip plate | | |
| | | FT 19 | As FT 18 but with deflector having rigid conical face but collapsile skirt |
| | | FT 2o | As FT 19 but add grid faced slotted cylinder |

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 30.12.77 | 20 | RS 276 |
| Vorhaben/Project Title<br>Untersuchung reaktivitätswirksamer axialer Brennstoffverlagerung in der flüssigen und festen Phase<br><br>Investigations on slumping of the fuel in the liquid and solid phase | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Fa. INTERATOM<br>Bergisch Gladbach 1<br><br>OE 9220 |
| Arbeitsbeginn/Initiated<br>1.8.1977 | Arbeitsende/Completed<br>31.3.1979 | Leiter des Vorhabens/Project Leader<br>H. Banasch (Koord.) |
| Stand der Arbeiten/Status<br>continuing | Berichtsdatum/Last Updating<br>31.12.1977 | Bewilligte Mittel/Funds<br>225.545,-- DM |

1. General Aim

The important phenomenon of local radial fuel relocation in normal operation resp. the behavior of fuel bundles is well understood and there exist a lot of always improved theoretical models. The understanding of all kinds of axial behavior of fuel pin, however, is almost in the beginning, especially for all effects on reactivity with regard to reactor safety.
The aim of this project is to clearify the problems of reactivity changes in the fuel pin of a LMFBR caused by axial fuel shifting (slumping).

2. Particular objectives

- Preparations for a problem orientated analysis with the code IAMBUS
- Interpretation of irradiation experiments and data collection for IAMBUS calculations
- Revision of IAMBUS with the aim to get a recording of the problem of all slumping phenomena within the fuel pin with regard to all effects of reactivity
- Final evaluation and documentation
- Final decision on a special test program.

3. Research Program

3.1 Special analysis of own irradiation experiments
3.2 Special analysis of foreign irradiation experiments
3.3 Literature review with respect to these special problems
3.4 Preparation of data and data collection for the IAMBUS-Code

3.5 IAMBUS-Calculations, i.e. Analysis of experimental data

3.6 Final report.

4. Experimental Facilities, Computer Codes

IAMBUS (Interatom Model for Burn-Up studies on fuel rods) is a computer code to describe the thermal and mechanical behavior of cylindrical fuel rods for power reactors. The general aim of this computer code is the analysis of fuel rod irradiation experiments as well as the transfer of these results on the design and the prediction of the in-pile behavior of fuel rods of LMFBR's.

5. Progress to Date

to 3.1 Examination of available reports on own irradiation experiments and special analysis of the data.

to 3.2 Summary of experiment description including data collection and single evaluation of foreign irradiation experiments.

This summary is finished now. A corresponding report is to be published.

to 3.3 A systematically review of the open literature has been started and will be continued further on.

to 3.4 First test calculations have been carried out with the computer code IAMBUS.

6. Results

to 3.4 In the frame of this special project it is not possible to use all experimental results of own irradiation tests, because some of these tests were run under different aims.

to 3.2 Mainly TREAT-experiments seem to be suitable for the examinations required within this project. First experiences exist with regard to here mentioned questions. Some of the TREAT experiments refer directly to slumping phenomena. There exist sufficient experimental data. A summarizing survey of all test results is under preparation. Using this survey it will be easier to use all these data directly for all IAMBUS calculations.

to 3.4 During the preparatory work for treating these special

problems with the computer code IAMBUS some test calculations have been carried out referring to the results of own irradiation experiments.
These calculations have to be regarded as precalculations.

7. Next steps

to 3.1 The data of selected irradiation experiments will be prepared for the use in IAMBUS.

to 3.2 After selection of experiments, checking of completeness of data sets the survey of data has to be summarized in a report.

to 3.3 The review of the open literature will be continued.

to 3.4 Based on the present experimental results and the test calculations it is possible to start with a rivision of the IAMBUS-code.

8. Relation with other Projects

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9. References

TANSAO 26,1-610 (1977); ANL-RDP-23, Dec. 1973;
TANSAO 22-1, 1-836 (1975); TANSAO 23, 1-637 (1976);
HEDL-TME 76-47; HEDL-TME 74.15

10. Degree of Availability of the Reports

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.1977-31.12.1977 | 20 | RS 254 |
| Vorhaben/Project Title<br>Entwicklung fernbedienter Ultraschall-Prüftechnik für Schnellbrutreaktoren<br><br>Development of Remote-Controlled Ultrasonic Inspection Technology for Fast-Breeder Reactors | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Maschinenfabrik Augsburg-Nürnberg<br>Aktiengesellschaft |
| Arbeitsbeginn/Initiated<br>1.1.1977 | Arbeitsende/Completed<br>31.12.1979 | Leiter des Vorhabens/Project Leader<br>Dipl.-Ing. H.-J. Otte |
| Stand der Arbeiten/Status<br>continuing | Berichtsdatum/Last Updating<br>30.12.1977 | Bewilligte Mittel/Funds<br>1.282.636,-- DM |

## 1. General Aim

The objective of the work carried out under this project is to further develop equipment for remote-controlled ultrasonic inspection of the reactor tank of liquid-metal-cooled fast breeder reactors so as to permit remote-controlled volumetric inspection of the austenitic reactor inner tank according to the present status of the art on completion of this work.

## 2. Particular Objectives

2.1 Development and preparation of complete engineering for the following equipment systems:

- Inspection equipment for the cylindrical tank region up to the level of the s-shaped bend of the guide tracks
- Probe module carrier for the cylindrical tank region and the spherical bottom head, including supply and disposal of couplant and extraction device
- Additional equipment for the inspection of the cylindrical tank region above the bend of the guide track
- Equipment for inspecting the nozzle-to-pipe transition welds
- Contoured probe module carrier with extraction device for the nozzle-to-pipe transition welds
- Additional equipment for testing the spherical bottom
- Cooling system for TV camera

2.2 Development of a special control system for adapting a multipurpose control system to match the specific requirements of the SNR inspection facility.

2.3 Testing of equipment system

## 3. Research Program

3.1 Development of the equipment systems referred to under 2.1 with the exception of the cooling system for the TV camera

3.2 Provision of a two-dimensional model to study the space conditions when traversing the s-shaped bend of the guide track

3.3 Development of an alternative to the currently envisaged drive concept

## 4. Experimental Facilities, Computer Codes

Not applicable

## 5. Progress to Date

Ad 3.1 The inspection equipment for the cylindrical tank region was reinforced in respect of all essential components to withstand higher forces on the probe module carrier. Subsequently, the detailing was started.

As a basis for discussion, an inspection concept has been developed for the cylindrical part and bottom of the inner tank. The extraction system for the couplant has been redesigned.

The development of the equipment for inspecting the nozzle-to-pipe transition welds as well as the bottom and the cylindrical tank region above the bend of the guide tracks has been started. In the light of initial studies, it is necessary to allow for angle variations of $5^{\circ}$ with respect to the weld direction during the inspection of the transition welds.

Ad 3.2 On the basis of the model that has meanwhile been completed, the critical positions as regards space conditions have been subjected to a detailed review.

Ad 3.3 Since doubts have arisen regarding the performance of the proposed drive concept, alternative solutions have been sought.
Discussions with U.S. manufacturers have not yet been completed.

6. Results

It has not yet been possible to complete the work in hand. Considerable difficulties are being encountered in respect of the existing space conditions.

7. Next Steps

The work under 3.1 and 3.3 will be continued. The development of a specific control system for matching a multipurpose control system to the specific requirements of the SNR test facility will be started.

8. Relation with other Projects

The problem definition has resulted in interfaces with the work done by Interatom, Bergisch-Gladbach, - RS 25o and the Bundesanstalt für Materialprüfung (BAM) - RS 244.

9. References

No technical reports have been completed during the period under review.

1o. Degree of Availability of the Reports

Not applicable.

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| | | |
|---|---|---|
| Berichtszeitraum/Period<br>1.1.1977 - 31.12.1977 | Klassifikation/Classification<br>20 | Kennzeichen/Project Number<br>RS 244 |
| Vorhaben/Project Title<br>Ultrasonic Testing Techniques for Pre- and Inservice Inspections of Fast Breeder Reactors<br><br>Ultraschallprüftechnik zur Null- und Wiederholungsprüfung von Schnellbrutreaktoren | | Land/Country<br>Bundesrepublik Deutschland<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>Bundesanstalt für Materialprüfung<br><br>Fachgruppe 6.2: Zerstörungsfreie Prüfung |
| Arbeitsbeginn/Initiated<br>1.1.1977 | Arbeitsende/Completed<br>31.12.1979 | Leiter des Vorhabens/Project Leader<br>Dr.-Ing. E. Neumann |
| Stand der Arbeiten/Status<br>Continuing | Berichtsdatum/Last Updating<br>December 1977 | Bewilligte Mittel/Funds<br>1.323.203,-- DM |

1. General Aim

The objective of the project is to develop an ultrasonic testing technique for austenitic welds because the licensing procedure for the sodium cooled fast breeder reactors requires ultrasonic testing during production and furthermore the acceptance testing and the in-service inspections by ultrasound during shut down periods. The difficulties in testing austenitic welds with ultrasound are arising from the special grain structure in austenitic weld metal. The limited reliability of the ultrasonic inspection of austenitic welds is caused by the high attenuation of ultrasound due to grain scattering. Whereas the US attenuation of fine grain austenitic base material is found to be as low as in common ferritic steels, the US attenuation in austenitic weld material is by a magnitude higher.

For the inspection of these austenitic steel welds special techniques had to be developed which are effective in materials with a rather high attenuation coefficient of the order $\alpha \sim$ 0,3 dB/mm which results in a low level of flaw echo signals and a high noise level due to grain scattering.

## 2. Particular Objectives

- Development of testing techniques for scattering materials
- Adaptation of testing techniques to curved specimen surfaces (e.g. tubes, nozzles etc.)
- Adaptation of testing techniques to elevated temperatures ($\sim 200^\circ$ C)
- Methods of signal processing to increase the signal to noise ratio

## 3. Research Program

3.1 Ultrasonic properties of the specimen's and ultrasonic probe materials

3.2 Development of ultrasonic testing techniques for
- plane coupling surfaces of the specimen
- complex shaped coupling surfaces of the specimen

3.3 Adaptation of testing techniques to elevated temperatures ($\sim 200^\circ$ C)

3.4 Application of developped ultrasonic testing techniques in the workshop.

## 4. Experimental Facilities, Computer Codes

- Laboratory equipment for ultrasonic testing at elevated temperatures
- Computerized equipment for signal processing
- Computer code for calculation of data for construction of ultrasonic probes.

## 5. Progress to Date

To 3.1 Measurement of velocity of sound and ultrasonic attenuation both as a function of temperature in different austenitic steels and ultrasonic probe components (wedge materials: ceramics, plastics).

To 3.2 Calculation: The main part of the investigations was concerned with further development of the transmitter receiver technique. The problem to calculate the transducer dimensions and the wedges geometry is solved by application of the Kirchhoff diffraction theory. This calculation is complicated and several severe approximations are necessary. Meanwhile a computer program which calculates these data for a given sensitivity distribution of the probe and a given coupling surface geometry of the specimen has been developped.

Properties of the transmitter receiver probes: Because of the limitation of the sensitivity range of the probes for the testing of thick walled parts several angular transmitter receiver probes have to be employed whose sensitivity ranges cover the whole weld under inspection. Size and position of the sensitivity range in the specimen and distance amplitude correction curves have been measured. As for specimens with curved surfaces special probes with adapted coupling surfaces are necessary. The test results are gradually deteriorating the more specimen radius R and probe radius r are deviating. According to now existing provisional experiences the gap may not be much greater than $\Delta s$ = 0,2 mm. The surface finish of the specimen has considerable influence on the testing results. It is desirable that the surface should be smooth-cut and that there should be no long-wave ripples with a depth of more than 0,2 mm. Further improvement in suppresion of reflections from coarse grain combined with a desirable high resolution is obtained by employing short, broad banded ultrasonic pulses. For increase of the effective bandwith of the ultrasonic testing system methods of signal processing too e.g. inverse filtering (or deconvolution) of the ultrasonic signals have been used. Meanwhile transmitter receiver probes for testing of longitudinal and transversal flaws for plane testing surfaces as well as for curved testing surfaces are available.

To 3.4 Applications of the transmitter receiver probes in the workshop: In all cases test specimens with equal geometry and from same material served to demonstrate the testability by ultrasound with especially tailored transmitter receiver probes. Kind, size and position of the reference reflector for sensitivity calibration were given by testing instructions.

- Inlet nozzle of an austenitic (X 6 Cr Ni 1811) pump barrel.
- Flange of a steam generator tube plate with varying radius from 99 to 150 mm.
- Circumferential weld joint between ferritic and austenitic material the ferritic part of the compound tube beeing cladded.
- Weld joints in a heat exchanger (X 8 Ni 6).
- Weld joints in a valve housing (X 20 Cro Mo V 121 and GS-12 Cr Mo 910 or 10 Cr Mo 910).
- Weld joints in a preheater for chemical industry (X 10 Ni Cr Al Ti 3220).

6. Results

The results already achieved have been described above.

7. Next Steps

- Development of ultrasonic testing techniques for complex shaped welded specimens especially thinwalled tubes (6 - 15 m ).
- Adaptation of the testing techniques to elevated temperatures (~ 200° C).
- Application of signal processing methods to increase the signal to noise ratio at ultrasonic testing of coarse grain materials.

8. Relation to other Projects

This project is linked with the projects RS 250 and RS 254 by a cooperation treaty.

9. References

Joint final report of BAM, M.A.N. and INTERATOM on the project RS 143, Ident-No. of the INTERATOM - INTAT 77.19-23.770 19.3-March 1977

E. Neumann, M. Römer, T. Just, K. Matthies, E. Nabel, E. Mundry: Development and Improvement of Ultrasonic Testing Techniques for Austenitic Nuclear Components;
ASM/ASTM/ASNT/ANS Conference: Nondestructive Evaluation in the Nuclear Industry, 13-15 Febr. 1978 - Salt Lake City - Utah - USA

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 01.01.77-31.12.77 | 20 | RS 250 |

| Vorhaben/Project Title | |
|---|---|
| Basis inspection and in-service inspection of austenitic reactor vessels and components with ultrasonic<br><br>Null- und Wiederholungsprüfung von austenitischen Reaktorbehältern und -komponenten mit Ultraschallverfahren | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: INTERATOM Bergisch Gladbach 1<br>OE 0310, 6330, 8000 |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 01.01.77 | 31.12.79 | H. Banasch (Koord.) |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | December 1977 | 2.011.570,-- DM |

1. General Aim

The objective of the project is to develop and to test a fully automatic test system for the non-destructive basic and in-service inspection of the austenitic walls of reactor vessels and components for sodium-cooled breeder reactors. By applying ultrasonic test methods to be further developed, it shall be possible to detect defects possibly prevailing in the austenitic material in order to judge or to prove the integrity of the high-temperature-loaded sodium systems. Besides the development and the qualification of the ultrasonic test method to be applied at approx. 200°C, the main items of the development program are the development of a remotely operable, movable test device and finally the testing of the overall system at a scale of 1 : 1 under simulated reactor conditions in a test stand.

2. Particular Objectives

- Conceptual design, layout and construction of a test stand to test the overall test system

- Conceptual design, layout, procuring or making available of the electronics for the ultrasonic test device and of the ultrasonic signal pick-up and processing system as well as ensuring the compatibility of these units for a system being under development to control the movable test device.

- Conceptual design, constructive design and manufacturing of the box (the drive unit for the vertical movement of the movable unit is arranged within the box. In addition, the coupling or the change of the test carriages to the link-carriage/chain system of the movable test device and the equipment of the test system carrier with the ultrasonic transducers required for the respective inspection are carried out within the box.)

- Supplementary work to support the development objectives of the involved partners
  - . Cooperation at the development of the ultrasonic transducers regarding their applicability at temperatures of approx. 200°C under nitrogen atmosphere
  - . Search for a suitable conpling medium
  - . Selection and testing of ultrasonic signal conductors
  - . Experimental measurement of the thermal losses of a $LN_2$-supply line for cooling a television camera
  - . Experimental investigations in order to limit the leak rates of the conpling medium at the dynamic conpling of the ultrasonic transducers to the subject to be tested

3. Research Program

3.1 Conceptual design of the test stand

3.2 Investigations to ensure the accessibility to the contaminated movable device

3.3 Continued investigations about preselected conpling media

3.4 Conceptual design of the ultrasonic data logger and processing unit

3.5 Investigations to limit the leakage of the coupling medium

3.6 Development of production methods for connecting the ultrasonic transducer components

4. Experimental Facilities, Computer Codes

The hither to underlying conceptual design for the test stand, which provided for the use of in part already available components for the test setup, was rejected in the 4th quarter because some requirements had not been met. Now such a test stand will be constructed, which will permit the testing of the overall test system at a SNR-300 scale of 1 : 1, setting up a reactor vessel section - partially filled with sodium - as object to be tested. Furthermore, it is planned that at the testing of the test system a universal control system shall be used, which is presently being developed by another organization.

5. Progress to Date

To 3.1 Due to the change of the conceptual design agreed upon in the 4th quarter for the test stand design, the detailed planning for this item shall be carried out in 1978. In this context the hitherto valid job descriptions for this program must be revised.

To 3.2 The study - provided for in accordance with the work program-on the procedure or concept regarding the change of the different test carriages at the box will be completed in the IInd quarter of 1978 after having clarified various residual problems. Contrary to the hitherto underlying program, the conceptual design and then the constructive design of the box will be initiated at the beginning of 1978 because this component has turned out to be on the critical path of the time schedule as a result of the modified testing concept.

To 3.3 By the end of the report period further qualification tests with the reference coupling medium have been initiated. Thereby for one part the irradiation behaviour of the medium shall be investigated and for the other part it shall be determined whether and how the ultrasonic detection will be affected by a solidified residual coupling medium layer on the austenitic structures.

To 3.4 Specifications have been elaborated for the different units of the ultrasonic data logger and processing system. Accordingly, a central master computer shall control the ultrasonic transducers which are mainly due to the largely different thermal expansion coefficients of the used materials. In the meantime, however, first progresses have been achieved with one of the three provided production methods.

6. Results

In the main, the results already achieved have been described above. At present final results are not yet available.

7. Next Steps

The work to be performed in 1978 shall mainly refer to the continuation of the investigations described above.

8. Relation to other Projects

This project is linked with the projects RS 244 and RS 254 by a cooperation agreement. There is another close relation with the RS project "Development of an universal control system".

9. References

Joint final report of BAM, M.A.N. and INTERATOM on the project RS 143, Ident-No. of the INTERATOM - INTAT 77.19-23.770 19.3 - March 1977

10. Degree of Availability of the Reports

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 01.09. - 31.12.1977 | 20 | RS 280 |

| Vorhaben/Project Title | |
|---|---|
| Development and testing of a gas bubble detection system controlling the reactor entrance pipes with ultrasonic sensors in sodium cooled fast breeder reactors.<br><br>Entwicklung und Erprobung eines Gasblasendetektionssystems zur Überwachung der Reaktoreintrittsleitungen mit Ultraschallsensoren bei natriumgekühlten Brutreaktoren | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: INTERATOM Bergisch Gladbach 1<br>OE 8220 |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 01.09.1977 | 31.12.1980 | H. Banasch (Koord.) |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | 31.12.1977 | 2.883.329,-- DM |

1. General Aim

Development of a bubble detection system in order to monitor critical gas contents in the SNR coolant by detecting bubbles in the reactor inlet lines.

2. Particular Objectives

Performance of a study on the conceptual design of a bubble detection system taking into account: safety criteria, plant layout criteria, bubble theory, selection of the transducer types, elaboration of proposals for the transducer design. Investigation of alternative measuring methods such as inductive or ultrasonic methods. Estimate on the detection limits, on the measuring range, on the reparability, on the development state, on the development keypoints and on the test expenditure. Elaborating constructive designs for the water tests.

3. Research Program

3.1 Performance of a study on the conceptual design of bubble detection systems.

3.2 Selection of the measuring methods

The measuring methods to be investigated are based on the inductive or ultrasonic method. Thereby it will be distinguished whether the measurement can be made inside or outside the pipe.

3.3 Elaboration of conceptual designs for the relevant measuring methods

After having elaborated the conceptual designs, it shall be possible to decide which of these methods have chances for a promising development and which of them will require a basic research.

3.4 Constructive design of the test equipment for tests under water

In this respect constructive designs of the measuring section and of the gas supply shall be elaborated.

## 4. Experimental Facilities, Computer Codes

As for the water tests, the available water tank (SNR-300 tank section) shall be used. The test setup including the gas supply will be installed within the water tank.

## 5. Progress to Date

To 3.1:
In the study the relevant measuring methods have been described. Taking into account the safety criteria and the currently known plant layout criteria, in the main the inductive and the ultrasonic methods have been investigated.

To 3.2:
From the investigated measuring principles, five methods based on inductive measuring principle and four methods based on the ultrasonic measuring principle have been selected. Furthermore, the elaboration of the marginal conditions to be observed for the necessary detection limits has been initiated.

To 3.3:
Data sheets have been elaborated, which contain the following data:

- Check of the development state
- Determination of the penetration depth and of the measuring range
- Interactions between measuring transducers and effects on the redundant system layout
- Demand and need of exciter capacity
- Temperature resistance and influence on the measuring transducers
- Resolving property and detection limits of the measuring transducers
- Response property
- Susceptibility to disturbances from outside such as speed, temperature fluctuations, wetting, electric disturbances, magnetic stray effects due to the cabling, vibrations in the plant etc.
- Necessity of intermediate amplifiers and adaption of cabling
- Linearity in the measuring range
- Demountibility during operation and maintenance.

To 3.4:
As for the water tests, the following constructive designs are being elaborated:

- Gas feed
  The gas feed occurs by means of 4 swivel arms each provided with approx. 20 bores having diameters between 50 µm and 0.5 mm. It has been provided for that bubbles can be fed with different sizes and at a random position.

- Test rig
  In the main, the test rig for the water tests has been provided for the investigation of the ultrasonic methods; it is designed with ring-shaped adjustable supports.

6. Results

7. Next Steps

To 3.2:
The elaboration of the marginal conditions to be observed for the detection limits is continued.

To 3.3:
The conceptual designs included in the data sheets are investigated in more detail by means of theoretical investigations.

To 3.4:
The constructive designs are continued.

Furthermore, the following work shall be carried out in 1978.

- Completion of the test rig for the water tests
- Conceptual design of the evaluation electronics
- Constructive design of the ultrasonic sensors for application in sodium
- Constructive design of sodium test sections in small loops
- Evaluation of the water tests
- Testing the inductive measuring methods in small loops.

8. Relation with other Projects

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9. References

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10. Degree of Availibility of the Reports

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 01.01.77 - 31.12.77 | 20 | RS 256 |

| Vorhaben/Project Title | |
|---|---|
| Versuche zur Prüfung von Reaktorkomponenten mit Wirbelstromprüfverfahren<br><br>Investigations for inspection of reactor components by eddy-current methods | Land/Country<br>FRG<br><br>Fördernde Institution/Sponsor<br>BMFT<br><br>Auftragnehmer/Contractor<br>INTERATOM<br>Bergisch Gladbach 1<br><br>OE 8120 |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1. 3. 1977 | 30. 9. 1978 | H. Banasch (Koord.) |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | 31. 12. 1977 | 219.575,-- DM |

1. General Aim

In cooperation with KWU, IFR, IzfP and Interatom investigations are conducted for automatic examinations of reactor components by multi-frequency eddy current testing for production control and inservice inspection. Tests carried out by Interatom have to determine the suitability of eddy current methods in components of LMFBR.

2. Particular Objectives

The first part of the investigations has the subject to

- require the access for inservice inspections,
- give the proof that eddy current testing is a successful method for the examination of sodium contaminated ferritic and austenitic heat exchanger internals.

3. Research Program

3.1 Investigation of accessibility

In this part boundary conditions are made up for inservice inspections by eddy current testing. Problems of accessibility are regarded from the aspect of construction, geometrical character of the heat exchangers and their internals, radiological and temperature conditions as well as the influence of sodium deposits.

3.2 Theoretical and experimental tests of eddy current application

By eddy current tests of SNR-300 heat exchanger tubes statements are made about the suitability of this method. The investigations are particularly concentrated on the influence of austenitic and ferritic materials and of sodium deposits on the tube surface.

4. Experimental Facilities, Computer Codes

To 3.2 In basic tests non destructive examinations are conducted in steam generator- and IHX-tubes, which are provided with well-defined artificial discontinuities and wetted with sodium in a test container.

5. Progress to Date

To 3.1 The investigation of accessibility, which refers on SNR-300-components was nearly concluded until the end of fiscal year 1977.

Completion caused by actual construction variations of the components will be taken into consideration in the summary report.

To 3.2 Construction and fabrication of the test setup were started. The experimental part of nondestructive tube testing is made in cooperation with IzfP - Saarbrücken.

6. Results

By compiling the boundary conditions for inservice inspections of SNR-300 steam generators, - IHX and -large primary pipes conditions for the application of eddy current systems in LMFBR's are defined.

In detail the following SNR-components are checked up:

- straight tube and helical coil tube evaporators and superheaters
- intermediate heat exchangers
- large primary pipes

In consideration of the special factors, listed in 3.1, some results are summarized:

SNR-300 steam generators are generally accessible for inservice inspections. The single heat exchanger tube inlets are reached through the steam outlet at the upper side of

the components. Radiation, temperature and sodium coolant don't restrict inspection.
The accessibility of the SNR-300 IHX tubes is more complicated because of the sodium contamination of the inner and outer tube surfaces. The only way of testing the heat exchanger tubes without demounting the tube bundle leads through the secondary sodium outlet nozzle but for this way of inservice inspection some restrictions are given concerning temperature, radiation and sodium influence. Nondestructive testing of large primary pipes is performed with remotely controlled manipulators. Concepts for the accessibility with removable pipe insulations for SNR-300 are being tested.

to 3.2
In the construction of the test set up the important SNR-conditions were realized such as dimensions and materials of the heat exchanger tubes. On the other hand it was necessary to protect the sodium on the tube surface against contact with oxygen. Therefore two straight tubes of each SNR exchanger type are welded into a portable closed metal container. So the outer tube surfaces can be wetted with sodium to verify test conditions for inservice inspection.

The type and dimensions of the artificial flows in the tube walls were chosen in coordination with the IzfP. In combination with sodium on the outer tube surface discontinuities only on the same side were expected to be important. Therefore in this step of the investigation program longitudinal and circumferential notches and drilled holes with defined dimensions were provided.

Fabrication of the sodium container and preparations for the connection with the Interatom test facility were started.

7. Next Steps

After fabrication of suitable eddy current probes by the IzfP the tubes are tested without sodium influence to have a standard of reference. Then the tubes will be set into the container and wetted with sodium. After draining the sodium next eddy current tests will follow as provided.

8. <u>Relation to other Projects</u>
RS 89, RS 255, RS 257

9. <u>References</u>
–

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

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.1977 | 20 | RS 248 |

| Vorhaben/Project Title | |
|---|---|
| Elementverbiegung im Kernverband<br><br>Element Bowing in Reactor Cores | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: INTERATOM Bergisch Gladbach<br>OE 911o |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.1.1977 | 31.12.1979 | Ing. grad. Banasch |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | 31.12.1977 | 870.740,-- DM |

## 1. General Aim

In order to fulfill the safety and economics requirements for the core design of large, sodium cooled fast breeder reactors, a program is needed to calculate the three dimensional subassembly bowing in a reactor core.

## 2. Particular Objectives

This project has the aim to develop a method step by step. So it will be possible to calculate the three dimensional geometrical, mechanical and reactivity behaviour of subassemblies in a reactor core with the consideration of thermal gradients, differential swelling, irradiation enhanced creep and friction between the subassemblies.

## 3. Research Program

3.1 Three dimensional force equilibrium in a reactor core
3.2 Consideration of swelling and irradiation enhanced creep
3.3 Testing and description
3.4 Consideration of friction
3.5 Calculation of torsion of subassemblies
3.6 Consideration of friction and torsion
3.7 Other mechanical models
3.8 Verification and qualification of the program

## 4. Experimental Facilities, Computer Codes

TO 3.1 The computer Cyber 172 is used.
A computer program to solve a set of linear equations with the following subroutines will be developed:

- Data preparation
- Bowing due to temperature gradients
- Calculation of gap widths between subassemblies and between subassemblies and the restraint rings
- Solving a set of linear equations
- Testing the solution and controlling the iteration
- Plotting and printing the results

## 5. Progress to Date

TO 3.1 A test program was written to find a possible and economical iteration technique. After successful achievement of this goal the writing of the program DDT was started. This program will manage to calculate any configuration of elements using available symmetry properties. Most subroutines have been written and are being tested.
The modul TEMTRA was written to establish the axial temperature distribution necessary for DDT and to collect the temperatures of all assemblies belonging to the calculated part of a reactor core and to store them together.

TO 3.2 Work was performed to transfer the neutronfluence and energy from the program KASY to DDT.

TO 3.3 Three comparisons with NUBOW-3D were completed.

TO 3.4 Assuming the linearisation of the friction forces, a possibility was found to incorporate friction forces between subassemblies and the test program was supplemented.
The schedule is fulfilled by the test program, except for point 3.2 which will be done after program DDT is completed.

## 6. Results

TO 3.1 At first, a point-relaxation process was developed. By this method, the largest force discrepancy is corrected, but the computer running time was extremely large. Therefore, other techniques were looked for and the gap iteration technique was found. Solving the linear equation several times, using the solution of the last step one obtains a converging gap distribution which has to be checked.
The thermohydraulic code IACOB was supplemented to calculate the average temperature of each side for the axial

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locations along a subassembly duct.

TO 3.3 The first two comparisons calculated with the testprogram gave the same results as the theoretical or NUBOW-3D results. Due to lacking detailed quantitative figures from NUBOW-3D calculations, it was not yet possible to perform a comparison of the results of the third example which is a real three dimensional problem. But it was learned from ANL experts via direct contacts, that the results seem to agree. An exact comparison is only possible, if better contacts to ANL can be reestablished.

TO 3.4 The program with friction, but using a friction coefficient $\mu$ = o,yields the same results as the program without friction. At the conference "Optimisation of sodium-cooled fast reactors" held in London 1977, the programs SABOW and CRAMP were presented by the Nuclear Power Corporation; both are three dimensional core restraint programs but CRAMP is the more recent code. CRAMP calculates subassembly bowing with the effects of temperature- and swelling gradients, irradiation enhanced creep and friction. Therefore a comparison would be very interesting.

## 7. Next Steps

TO 3.1 The programming of DDT will be completed.

TO 3.2 After finishing DDT, a second modul DDAB will be written to calculate the bowing of differential swelling and irradiation enhanced creep.

TO 3.3 Testing and description of the programs DDT and DDAB.

TO 3.4 Some tests will be done, to see how the testcode is running with friction and to check the results.

TO 3.5 Consideration will be done with respect to torsion of subassemblies and a program will be written.

TO 3.6 This is planned for 1979.

TO 3.7 Considerations of the kind of mechanical duct models. Especially the variation of duct stiffness with loading conditions will be looked into.

TO 3.8 The information exchange with the USA being nearly blocked it is becoming more and more difficult to obtain new test samples for verification and qualification of the programs

DDT and DDAB. Due to the complexity of the input data set real comparisons are only possible, if a close contact with ANL or NPC (or UKAEA) can be maintained.

## 8. Relation with other Projects

## 9. References

R. Menssen

DDT - a 3-dimensional Program for the Analysis of Bowed Reactor Cores 4th SMIRT-Conference Aug. 77, San Francisco

## 1o. Degree of Availability of the Reports

The report is published in the proceedings of the 4th International Conference on Structural Mechanics in Reactor Technology as report D 2/4.

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.7.77 - 31.12.77 | 20 | RS 260 |

| Vorhaben/Project Title | |
|---|---|
| Untersuchungen zur langzeitigen thermischen Gefügestabilität des Stahles 10 CrMoNiNb 9 10<br><br>Investigation on long term stability of steel 10 CrMoNiNb 9 10 | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>INTERATOM<br>Bergisch Gladbach 1<br>OE 8320 |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.7.77 | 31.12.80 | Mr. Banasch (Coord.) |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | 31.12.77 | 51.060,-- DM |

1. General Aim

Determination of parameters influencing the mechanical properties in the long time high temperature range of the ferritic steel 10 CrMoNiNb 9 10 which has been chosen as a structural material for sodium cooled reactor systems.

2. Particular Objectives

- Investigation of the recrystallization behaviour.
- Influence of recrystallization on mechanical properties.
- Evaluation of the results from RS 261 by Mannesmann concerning the importance for the project RS 271 "low cycle fatigue behaviour of steel 10 CrMoNiNb 9 10".

3. Research Program

3.1 Specification, procurement and characterization of test materials for the projects RS 261 and 271.

3.2 Coordination of the projects "thermal stability" and "low cycle fatigue behaviour".

3.3 Evaluation of the results of the program "thermal stability" with regard to the low cycle fatigue strength.

3.4 Evaluation of the results concerning the definition of design values.

4. Experimental Facilities, Computer Codes

Tests are performed by Mannesmann. The present objective includes the activities according to point 3.

5. Progress to Date

Specification of test materials in cooperation with Mannesmann. Production of test materials (6 melts) by Mannesmann.

6. Results

Results of the test program are not yet available.

7. Next Steps

Production of further test materials (weld metal 10 CrMoNiNb 9 10), start of long time exposures, performance of tensile and impact tests.

8. Relation with other Projects

This project is closely connected with the projects RS 261 and 271.

9. References

-

10. Degree of Availability of the Reports

-

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| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.4.77 - 31.12.77 | 20 | RS 0261 |

| Vorhaben/Project Title | Land/Country |
|---|---|
| Untersuchungen zur langzeitigen thermischen Stabilität des Gefüges des Stahles 8 CrMoNiNb 9 10<br><br>Study on the long-term stability of the microstructure of the steel 8 CrMoNiNb 9 10 at elevated temperatures | FRG<br>**Fördernde Institution/Sponsor**<br>BMFT<br>**Auftragnehmer/Contractor**<br>Mannesmann-Forschungsinstitut, Duisburg<br><br>Abt. Metallkunde |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.4.1977 | 31.12.1980 | Dr. H. Fabritius |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | December 1977 | 417.499,-- DM |

1. General Aim

Determination of the parameters that influence the strength of the steel 8 CrMoNiNb 9 10 during long term exposure at elevated temperatures. This steel is used for components in sodium cooled fast breeding reactors.

2. Particular Objectives

The effect of the degree of stabilization (Nb/C) on recovery and recristallization shall be studied as well as the influence of coldworking.

3. Research Program

3.1 Materials: see Table 1

3.2 Rolling of the Materials:

The casts 1 - 6 are hot rolled to plats 22 mm in thickness

3.3 Heat treatment:

Cast 1: 1020 °C 30 min/oil and
720 °C 1 h/air cooling

Casts 2 - 5: 1020 °C 30 min/air cooling and
720 °C 1 h/air cooling

Cast 6: 950 °C 30 min/oil and
720 °C 1 h/air cooling

Material 7: 700 °C 5 h/air cooling

3.4 Pre-treatment of the materials:

20 % cold rolling of one half of the materials 1 - 6 (version A). Pre-annealing 20 h at 700 °C of parts of the materials 5 and 7 (material 5 in both conditions, heat treated and heat treated and cold worked) (version V).

3.5 Cutting of squares for annealing according to 3.6 of the materials according to 3.3 and 3.4.

3.6 Annealing of the squares according to the schedule in table 2.

3.7 Tensile tests are carried out at 20 °C and 550 °C on every specimen, annealed according to 3.6. The initial conditions according to 3.3 and 3.4 are tested with double specimens.

3.8 The impact toughness (Charpy-V-notch, transverse): The $A_V$-T-curve and the toughness at 550 °C is determined with 18 specimens for every annealing condition. The conditions tested are: initial conditions according to 3.3 and 3.4 and two selected annealing times at every temperature.

3.9 Metallographic examination of the microstructure and hardness-measurement at all annealing conditions.

3.10 Electron microscopy and residual analysis (chemical and X-ray) of selected specimens.

4. <u>Experimental Facilities</u>

| | |
|---|---|
| to 3.1 | Existing laboratory and industrial equipment. |
| to 3.2 | " " " " " |
| to 3.3 | " " " " " |
| to 3.4 | Existing laboratory equipment |
| to 3.5 | " " " |
| to 3.6 | Partly existing electrically heated furnaces, partly furnaces to be provided. |
| to 3.7 | Existing laboratory equipment |
| to 3.8 | " " " |
| to 3.9 | " " " |
| to 3.10 | " " " |

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

| | |
|---|---|
| to 3.1 | Finished, except material 7 |
| to 3.2 | Finished |
| to 3.3 | " , except material 7 |
| to 3.4 | " , " " 7 |
| to 3.5 | " , " " 1 and 7 |
| to 3.6 | None |
| to 3.7 | Tensile tests at RT and 550 °C on materials 5,5A and 6 in the initial condition. |

| | |
|---|---|
| to 3.8 | Impact tests on the materials 5,5A, 6 and 6A in the initial condition. |
| to 3.9 | None |
| to 3.10 | None<br>The work is in time. |

| | |
|---|---|
| 6. | <u>Results</u> |
| to 3.1 | Finished, except material 7 |
| to 3.2 | " |
| to 3.3 | " , except material 7 |
| to 3.4 | " , " " 5, 5A, 7 |
| to 3.5 | " , " " 1 and 7 |
| to 3.6 | No results |
| to 3.7 | The mechanical properties of materials 5, 5A and 6 at RT and 550 °C in the initial condition are present |
| to 3.8 | The impact values of the materials 5, 5A, 6 and 6A in the initial condition are present. |
| to 3.9 | No results |
| to 3.10 | " " |

| | |
|---|---|
| 7. | <u>Next Steps</u> |
| to 3.1 | Production of material 7 |
| to 3.2 | None |
| to 3.3 | Heat treatment of material 7 |
| to 3.4 | Pre-annealing of materials 5, 5A and 7 |
| to 3.5 | Cutting of the squares from materials 1 and 7 |
| to 3.6 | Start of the annealing treatment |
| to 3.7 | Testing of the materials in the initial condition and after short annealing treatments |
| to 3.8 | Testing of the materials in the initial condition |
| to 3.9 | Testing in the same extend as with 3.7 |
| to 3.10 | None |

8. <u>Relation with other Projects</u>

Relation with the Project RS 02 605 (Interatom)

9. References

None

10. Degree of Availability of the Reports

-

Table 1: characterization of the materials

| No. | material denomination according to DIN | remarcs |
|---|---|---|
| 1 | similar 8 CrMoNi 9 10* | 0,01 - 0,02 % C, Nb : C = 0 |
| 2 | 8 CrMoNiNb 9 10* | about 0,6 % Nb, Nb : C ≈ 10 |
| 3 | 8 CrMoNiNb 9 10* | 0,7 - 0,8 % Nb, Nb : C ≈ 12,5 |
| 4 | 8 CrMoNiNb 9 10* | 0,9 - 1,0 % Nb Nb : C ≈ 16 |
| 5 | 8 CrMoNiNb 9 10** | high degree of stabilization |
| 6 | 10 CrMo 9 10** | C ≤ 0,10 %, Nb : C = 0 |
| 7 | similar 8 CrMoNiNb 9 10*** | usual analysis |

* 1000-kg-laboratory casts
** commercial material
*** weld metal deposit

<u>Table 2:</u> annealing schedule to 3.6

| ϑ [°C] | annealing time in h | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| 20 | 0 | | | | | | | | | |
| 750 | 1 | 3 | 10 | 30 | 100 | 300 | 1000 | | | |
| 725* | - | 3 | 10 | 30 | 100 | 300 | 1000 | | | |
| 700 | - | 3 | 10 | 30 | 100 | 300 | 1000 | 3000 | 10000 | 30000 | |
| 650 | - | - | - | 30 | 100 | 300 | 1000 | 3000 | 10000 | 30000 | 50000 |
| 600 | - | - | - | - | 100 | 300 | 1000 | 3000 | 10000 | 30000 | 50000 |
| 550** | - | - | - | - | 100 | 300 | 1000 | 3000 | 10000 | 30000 | 50000 |

* Except with material 5 and 5A

** The pre-annealed specimens of the materials 5, 5A and 7 (3.4) are annealed only at 550 °C.

1653

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.7.77 - 31.12.77 | 20 | RS 271 |

| Vorhaben/Project Title | |
|---|---|
| Ermüdungsverhalten des Stahles 10 CrMoNiNb 9 10 unter Berücksichtigung von Haltezeiten<br><br>Low cycle fatigue behaviour of steel 10 CrMoNiNb 9 10 in consideration of hold time | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: INTERATOM<br>Bergisch Gladbach 1<br>OE 8320 |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1.7.77 | 31.12.80 | Mr. Banasch (Coord.) |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| Continuing | 31.12.77 | 884.247,-- DM |

## 1. General Aim

Determination of the low cycle fatigue behaviour of steel 10 CrMoNiNb 9 10 and of the influence of hold times for the derivation of design values in the strength analysis for sodium cooled reactor plants.

## 2. Particular Objectives

- Determination of the low cycle fatigue behaviour at 550°C in different aging conditions.
- Investigation of the hold time effect on the life time in low cycle fatigue test.
- Use and evaluation of damage accumulation rules.
- Determination of the influence of microstructure on the low cycle fatigue behaviour.

## 3. Research Program

3.1 Pretests to the low cycle fatigue behaviour

3.2 Preconditioning of the specimens

3.3 LCF-tests without hold time

3.4 LCF-tests with hold time

3.5 Creep tests for evaluation of the hold time effect

3.6 Relaxation tests to define the effective stress during hold time.

3.7 Use of damage accumulation laws to evaluate the test results.

3.8 Evaluation of structural changes at high temperatures on the low cycle fatigue behaviour.

3.9 Definition of design values.

4. Experimental Facilities, Computer Codes

The low cycle fatigue tests are performed with a servo-hydraulic universal testing machine with axial strain measurement and strain-control in the "closed loop system".
For the relaxation tests one testing machine is available.

5. Progress to Date

In cooperation with Mannesmann seven test materials were specified with the following basic chemical composition:

| % C | % Si | % Mn | % P | % S | % Cr | % Ni | % Mo | % Al | % N |
|---|---|---|---|---|---|---|---|---|---|
| 0,05/<br>0,07 | 0,15/<br>0,5 | 0,40/<br>0,80 | 0,010/<br>0,015 | 0,010/<br>0,015 | 2,0/<br>2,5 | 0,60/<br>0,70 | 0,9/<br>1,1 | 0,010/<br>0,020 | 0,010/<br>0,015 |

These heats mainly differ in the Nb-content which was fixed as follows:

| Mat.No | Nb-content |
|---|---|
| 1 | no niobium; C = 0.01 - 0.02 % |
| 2 | low stabilized (Nb = 0.6 %; Nb : C = 10 : 1) |
| 3 | middle degree of stabilization (Nb = 0.7 - 0.8 %; Nb : C = 12.5 : 1) |
| 4 | high stabilization (Nb = 0.9 - 1.0 %; Nb : C = 16 : 1) |
| 5 | high stabilized commercial heat |
| 6 | steel 10 CrMo 9 10 (no Nb), commercial heat |
| 7 | weld metal, Nb-alloyed |

The test materials 1 - 6 were produced by Mannesmann. Due to the shortage of material from heat no. 5 this melt cannot be taken for low cycle fatigue testing. Instead of material 5 a new melt (original SNR steam generator material) was ordered by INTERATOM. The delivery of the test materials is expected in January 1978.

## 6. Results

Test results are not available for the reasons mentioned above.

## 7. Next Steps

Start of the low cycle fatigue testing on material in the as-received condition.
Further steps see point 3.

## 8. Relation with Other Projects

This project is closely connected with the projects RS 260 and RS 261.

## 9. References

-

## 10. Degree of Availability of the Reports

-

1657

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1. 7. - 31. 12. 1977 | 20 | RS 259 |

| Vorhaben/Project Title | |
|---|---|
| Untersuchung von Na-Meßverfahren auf ihre Anwendbarkeit in NaK-Kreisläufen<br><br>Test of Sodium Instrumentation and Measurement Methods for Application in NaK-Loops | Land/Country: FRG<br>Fördernde Institution/Sponsor: BMFT<br>Auftragnehmer/Contractor: Technische Universität Berlin<br>Institut für Kerntechnik |

| Arbeitsbeginn/Initiated | Arbeitsende/Completed | Leiter des Vorhabens/Project Leader |
|---|---|---|
| 1. 7. 1977 | 30. 6. 1979 | Prof.Dr.-Ing.U.Wesser |
| **Stand der Arbeiten/Status** | **Berichtsdatum/Last Updating** | **Bewilligte Mittel/Funds** |
| continuing | Dec. 31, 1977 | 310.900,-- DM |

## 1. General Aim

Measurement methods and liquid metal cleaning techniques will be tested concerning their applicability in NaK-loops. The measurement methods were developed for water and partially employed successfully. Experiences with liquid metal cleaning techniques exist for applications in Na- and K-loops. The measurement and cleaning methods should be tested in conditions existing in a core-melting-prevention system.

## 2. Particular Objectives

The tests of measurement methods and liquid metal cleanings are planned especially in NaK-loops. In the starting phase the operational security of the test-rig is to be tested, parameters are to be measured in stationary states. Nonstationary operating conditions request fast variations of temperature in the range up to 500 °C. This situation is expected in a coremelting-prevention system. Determination of attainable temperature-transients is the aim of the proceeding work. Tests with fluidlevel detecting systems and flow meters may show temperature influence concerning measurement accuracy of these instruments. Information about onset and formation of cavitation may be provided by measuring instruments, mounted at the outside. Operational security of liquid-metal-loops is dependent on the purity of the liquid metal. For this aim purchasable instruments for determination of oxid concentration and cleaning methods are tested.

## 3. Research Program

The research program is divided into six interconnected activities.

3.1 Operational tests NaK-loop

3.2 Operational tests NaK-K-loop
3.3 Investigation of methods for oxid concentration measuring and NaK-cleaning
3.4 Test of flow meters
3.5 Test of fluid level monitoring systems
3.6 Test of detection methods for cavitation

## 4. Experimental Facilities

The test rig has two closed loops. One of them consisting of a NaK-loop, the other of a K-loop. Both loops may be operated separately. Secondary rigs include systems for cover-gas and cool-air-providing, vacuum stand, cleaning system for argon and filling systems. The test rig is remote-controlled with projected connections to the data processing system.

## 5. Progress to Date

5.1 Operational tests NaK-loop
The preparations for starting the NaK-loop have been accomplished.

5.2 -

5.3 Investigation of methods for oxid concentration measuring and NaK-cleaning
Installation of a pluggingmeter for oxid concentration measuring and a cold trap for NaK-cleaning.

5.4 Test of flow meters
In the NaK-loop an EM-flowmeter was installed.

5.5 Test of detection methods for cavitation.
Design and construction of a sound-transducer-system. Operational test of this system.

## 6. Results

-

## 7. Next Steps

Starting of the NaK-loop and completion of the K-loop. Operational capability of the test rig is to be checked and the area of parameters is to be determined. The built-in and the still to be installed measuring instruments will be checked and operated.

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8. Relation with Other Projects

-

9. References

-

10. Degree of Availability of the Reports

-

| Berichtszeitraum/Period | Klassifikation/Classification | Kennzeichen/Project Number |
|---|---|---|
| 1.1.77 - 31.12.77 | 20 | RS 305 |
| Vorhaben/Project Title<br>Investigations on the interactions of pressure waves and components in liquid filled systems<br><br>Untersuchung der Wechselwirkung zwischen Druckwellen und Bauteilen in flüssigkeitsgefüllten Systemen | | Land/Country<br>FRG<br>Fördernde Institution/Sponsor<br>BMFT<br>Auftragnehmer/Contractor<br>INTERATOM<br>Bergisch-Gladbach 1<br><br>OE 8210, 7110 |
| Arbeitsbeginn/Initiated<br>1.11.77 | Arbeitsende/Completed<br>28.2.81 | Leiter des Vorhabens/Project Leader<br>H.Banasch(Koord.) |
| Stand der Arbeiten/Status<br>continuing | Berichtsdatum/Last Updating<br>31.12.77 | Bewilligte Mittel/Funds<br>1.722.679,-- DM |

## 1. General aim

LMFBR safety analysis shows a number of plant failures by which strong overpressure transients in the system occur. In the design of components the hydraulic loads due to the resulting pressure waves have to be taken into considerations.

As an "improved status of the art" the description of pressure-time histories inside of an apparatus in which the accident takes place is already possible and proofed. But there is still a great lack of knowledge in the mathematical-physical description of the transmission of pressure waves through a system. This knowledge is necessary in order to describe the behaviour of a component which is pressurized by a pressure wave transmitted from another one.

## 2. Particular objectives

The particular aim is the development of experimentally proofen computer codes which allow a satisfactorily description of the interactions between pressure waves and components. Special attention will be payed on the elastic-plastic behaviour of components during highly hypothetical events.

## 3. Research program

At INTERATOM there are three different computer codes for the description of pressure waves under design. These are: HEINKO/C, ROPLAST/2 and MEKKA/1.

The ranges of validity of these codes have to be verified and qualified as well by experiments.

## 4. Experimental facilities, computer codes

The experimental set up consists of a pipe system of 100 mm nominal diameter. Pressure waves of different amplitudes and steepnesses will be generated by gas explosions or deflagrating explosives. Transients up to 35 bar/ms and maximum amplitudes up to 120 bar are envisaged. The system will operate at room temperature, with water as the pressure transmitting liquid. The resulting load functions will be picked up by piezoelectric pressure transducers and strain gages. The experimental results have to be compared with the results of calculations done with the different computer codes.

Three computer codes at different stages of development are available at INTERATOM:

- HEINKO/C, a one dimensional program for the description of pressure waves. The feed back from the tube wall material to the pressure is approximated by a change of the velocity of sound in the liquid.
- ROPLAST/2, describes the elastic-plastic behaviour of the tube wall in interaction between tube wall material and pressure wave. Similar to HEINKO, the hydrodynamics are one dimensional.
- MEKKA/1, a three dimensional program for the calculation of pressure wave scattering effects inside a component such as i.e. an elbow.

## 5. Progress to date

The time schedule has been revised to a start point of 1.11.77. In detail the following activities have been started.

- Design of a spherical pressure wave generator and first connections to vendors.
- Design of a platform to support the test installations, first connection to vendors.
- A first preliminary design of some test components.
- Evaluation of a suitable instrumentation for fast pressure measurements, to complete the already available equipment.

6. Results

none

7. Next steps

The next steps are given by the time schedule as follows:

- Design, fabrication and delivery of test equipment
- theoretical preparation of computer programs
- test of the pressure wave generator
- start of experiments

8. Relations to other projects

none

9. References

none

10. Degree of avialability of reports

none.