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

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EUR 6?

7. CONTAINMENT AND ASSOCIATED SYSTEMS

Titre	Pays :
Etude du comportement du puits de cuve des réacteurs PWR 900 MWe en cas de rupture de cuve	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais)	Organisme exécuteur :
Primary shield wall behaviour of PWR's in case of pressure vessel rupture	CEA - DEMT
	Responsable : R.AVET FLANCARD (DSN-SETSSR) ROCHE (DEMT)
Date de démarrage : 01/01/75 Date prévue d'achèvement 31/12/80	Scientifiques :
Etat actuel : en cours Dernière mise à jour : 03/01/77	

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).

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 est susceptible d'influer l'élaboration des règles et des guides pour la conception du puits de cuve.

Installations expérimentales et programme :

L'étude et la mise au point de l'installation expérimentale ont été confiés 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 les puits de cuve).

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

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Etat de l'étude :

1) Avancement à ce jour :

- Environ quarante essais à petite échelle ont été réalisés.

- Parallèlement à ces essais, lancement d'une étude d'avant-projet à l'échelle 1/20 compte tenu des orientations définies par la sûreté.

2) Résultats essentiels :

- La rupture des maquettes est atteinte pour des pressions non uniformes et très localisées au tout début de la détente du circuit d'eau pressurisée.

- Les premiers essais permettant de dégager les points suivants :
  - Les pics de pression en début de détente constituent le chargement le plus sévère pour le puits de cuve.
  - A l'échelle des essais, les exutoires n'ont aucune influence sur le niveau des pics.
  - Le ferraillage des maquettes en béton armé est insuffisant pour éviter la création de projectiles secondaires.

Prochaines étapes :

Nouvelles séries d'essais à petite échelle avec maquettes comportant davantage d'aciers passifs, et maquettes précontraintes.

Etude d'avant projet au 1/20.

Titre PIEGEAGE DE L'IODE DANS LES BETONS	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) IODINE - TRAPPING IN CONCRETE	Organisme exécuteur : CEA/DTECH/SECS/STA STA-SACLAY Responsable : R.AVET FLANCARD (DSN-SETSSR) CONTRE (DTECH-STA)
Date de démarrage : 1/01/75 Date prévue d'achèvement 31/12/78 Etat actuel : en cours Dernière mise à jour : 03/01/77	Scientifiques :

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.

Objectifs particuliers :

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

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

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

.../...

Etat de l'étude :

I) Avancement à ce jour :

- L'adaptation du banc de mesure a nécessité l'étude et la réalisation :

  - d'un ensemble chromatographique pour mesurer en continu les quantités d'iode qui auront diffusé à travers l'éprouvette de béton

  - d'un dispositif d'injection des iodes

- Des essais préliminaires ont été également effectués pour vérifier que les joints d'étanchéité prévus sur le banc d'essai ne fausseront pas les résultats (Problèmes du piégeage préférentiel des composés iodin et de diffusion à travers le joint).

2) Résultats essentiels :

- Pas de résultats à ce jour sur la diffusion de l'iode dans le béton.

- A ce stade on peut noter néammoins les points suivants :

  - les joints d'étanchéité sont satisfaisants

  - le chromatographe et le dispositif d'analyse séquentielle automatique sont opérationnels

  - la construction du dispositif d'injection d'iode est en cours.

Prochaines étapes :

- Montage des dispositifs de mesure et d'injection d'iode à SACLAY

- Premiers essais avec de l'iode stable (ICH3)

- Etude des problèmes posés par l'analyse de l'iode moléculaire stable

Documents de référence :

- Note DSN/SETS 75-08 du 6-0I-75

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 Completed : 1975 Status : completed 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 Operational : June 1977	Project leader E.STUBBE 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.

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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

8. 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 Completed : April 73 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 Completed : July 1975 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 the simulation 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

Classification: 7.1	
Title 1 (Original Language): Untersuchung der Vorgänge in einem mehrfach unterteilten Containment beim Bruch einer Kühlmittelleitung wassergekühlter Reaktoren (RS 50 - I.1.4 , Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Battelle-Institut e.V., Ffm.
Title 2 (English): Investigation of the Phenomena Occurring within a Multi-Compartment Containment after Rupture of the Primary Cooling Circuit in Water-Cooled Reactors	Project Leader: Dr. T.F. Kanzleiter
Initiated (Date): May 4/14, 1971 Status: Continuing Completed (Date): June 30, 1978 Last Updating (Date): December 31, 1976	

1. General Aim

The objective of the present 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. Further LOCA experiments.

4. Experimental Facilities, Computer Codes

The experimental facilities consist essentially of

- a special model containment (approx. 600 m^3, 6 bar),
- a model coolant circuit (approx. 6 m^3, 140 bar, 300 ^oC),
- measuring instruments (approximately 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, several computer codes are used by external institutions. Some of these codes are being used for licensing procedures, others have been newly developped in connection with this research program.

5. Progress to Date

Ad 3.1: The first series of LOCA experiments (Nos. C1 to C16) with single- and double-end water line ruptures in different compartments of the PWR model containment was finished in spring 1976.

Ad 3.2: A new series of basic experiments (D series) has been specified and prepared for starting in January 1977. These experiments may be characterized as follows:

- pressure vessel 5.3 m^3, inside diameter 0.78 m, height 11.2 m, initial water level 5.0 m (286 ^oC).

- Depressurization via a steam line (140 mm in diameter) connected to the pressure vessel at its top end.

Initial conditions: 70 bar, 286 °C
Expected outflow: for approx. 1.5 s saturated steam, subsequently a steam-water mixture.

- Directing the saturated steam or the steam-water mixture into a 40-m^3 compartment and from there, via a series of adjoining compartments, to the dome space. The total containment volume is about 630 m^3.

- Overflow cross sections between the containment compartments of uniform diameter (750 mm), mainly designed as sharpedged orifices.

Series D will include an experiment which has been suggested by the Gesellschaft für Reaktorsicherheit for international comparison with theoretical results (Containment Analysis Standard Problem).

6. Results

Ad 3.1: Comparisons between the results of the C experiments and model calculations have been and are still being made by several research institutions in the Federal Republic of Germany and in the USA on the basis of different computer codes.

Comparisons of the experimental results with the results obtained by the computer codes used in the licencing procedure always showed a calculatory overestimate of the actual situation, with respect to both the absolute pressures and the differential pressures between compartments situated close to the site of rupture. At high differential pressures between the compartment where rupture took place and the adjoining compartment, deviations even exceeding a factor of 2 were observed in some cases. The calculations of the differential pressures for compartments located far from the site of rupture are highly erroneous. These differential pressures are, however, always so small that they do not affect the design of the plant. It was not possible to improve the computational results substantially even by appropriate variation of the somewhat uncertain input parameters.

Based on the above findings various institutions investigated the weak points of existing computer codes and improved them. G. Mansfeld (1) presented the computational results of the new multi-node code COFLOW, which show a much better agreement with the experimental results than conventional calculation methods. The COFLOW code differs from the conventional multi-node codes, e.g. ZOCO VI of LRA, by

- subdividing long compartments into several control volumes,
- considering the flow velocities within the control volumes.

7. Next Steps

Ad 3.1: Evaluation of the results will be continued and reports will be prepared.

Ad 3.2: The experiments of the D series will be carried out.

Ad 3.3 and 3.4: Specification and preparation of further experiments.

8. Relation to Other Projects

9. References

(1) G. Mansfeld, Simulation of Dynamic Pressure Differences in Full-Pressure Containments after a Loss-of-Coolant Accident, Comparison between Theoretical and Experimental Results. Laboratorium für Reaktorregelung und Anlagensicherung Garching. Paper presented at the IAEA Specialist Meeting on Thermo-Hydraulic Consequences of Loss-of-Coolant Accidents Inside and Outside the Containment. Cologne, FRG, 7-8 December 1976.

(2-5) Quarterly Reports in the Series "IRS-Forschungsberichte" (in German)

IRS-F-30 January to March 1976
IRS-F-31 April to June 1976
IRS-F-33 July to September 1976
IRS-F-34 October to December 1976

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

(7) BF-RS50-21-3 "Fehleranalyse für das Meßwerterfassungs- und -verarbeitungssystem der Containmentversuchsanlage" April 1976

(8) BF-RS50-31-4 "Aufbereitung der auf Magnetband dargestellten rohen Meßdaten zu physikalischen Größen" April 1976

(9) BF-RS50-32-C9 "Vorläufiger Versuchsbericht C9", March 1976

(10) BF-RS50-32-C10 "Vorläufiger Versuchsbericht C10", April 1976

(11) BF-RS50-32-C11 "Vorläufiger Versuchsbericht C11", Sept. 1976

(12) BF-RS50-32-C12 "Vorläufiger Versuchsbericht C12", Sept. 1976

(13) BF-RS50-32-C13 "Vorläufiger Versuchsbericht C13", July 1976

(14) BF-RS50-32-C15 "Vorläufiger Versuchsbericht C15", July 1976

(15) BF-RS50-32-C16 "Vorläufiger Versuchsbericht C16", Nov. 1976

(16) BF-RS50-62-4 "Experimentelle Bestimmung der instationären Wärmeübergangszahl in einem Containment während eines Kühlmittelverlustunfalls", June 1976

10. Degree of Availability of the Reports

Reports are available through

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

Documents (7) to (15) can be made available only by special agreement.

Classification: 7.1	
Title 1 (Original Language): Untersuchungen über die Auswirkungen des Ausströmens von Dampf- und Dampf-Wasser-Gemischen aus Rohrleitungslecks (RS 93 - I.1.4, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Reaction and Impingement Forces on Components and Structures Caused by a Two Phase Jet Discharged at Primary Pipe Breaks	Project Leader: Dr. Riedle
Initiated (Date): anuary 1973 atus: Completed Completed (Date): October 1975 Last Updating (Date): December 1975	

General Aim and Particular Objectives

In the safety analysis of a loss of coolant accident it is necessary to calculate the mechanical loading caused by the pressurized water discharging through the break. The discharging two-phase jet applies thrust on the supporting structures of the primary loop components and pipe restraints. The neighbouring walls and structures have to withstand the impact pressure caused by the impingement of the jet.

Experimental Facilities and Research Program

This program deals with quasi-steady state experiments which were carried out to investigate on a reduced scale a guillotine break and a slot break. During the guillotine break tests the discharging jet is directed on to a baffle plate, during the slot break tests two jets are directed against each other and are diverted by the simulated bottom of the ruptured pipe. The discharged fluid is saturated water; in several cases experiments with one-phase fluid were carried out.

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The test parameter were varied in the following way:

discharge diameter D,	guillotine break:	5, 10, 25, 50, 65 mm
discharge diameter D,	slot break:	5, 10 mm
distance pipe outlet-baffle plate A	guillotine break:	0,25 to 10 D
distance between the pipe outlets A	slot break:	0,20 to 14 D
stagnation pressure:		10 to 100 bars

The instrumentation supplies following informations:

critical discharge rate,
thermohydraulic state at the pipe outlet,
pressure distribution on the baffle plate,

the contour of discharging two-phase jets as well as thrust forces.

Project Status / Progress to Date / Essential Results

The investigations concerning the critical flow of saturated pressurized water have shown the following results:

1. The critical flow rate measured agrees well with theoretical calculations of diverse models from literature. The hydraulic resistance of the pipe, the diameter of the circular outlet cross section and, connected with it, the flow regime are of particular importance. For small outlet dimensions the calculation with the Moody model (flow regime: annular flow) seems to be successful, while for large outlet a model with homogeneous flow conditions should be preferred (e. g. that by Linzer). These facts were confirmed by the measured results.

2. The thermohydraulic state at the outlet cannot be determined exactly by the measurement of the outlet pressure and temperature and by mathematical estimation of the steam content. These data do not allow exact conclusions about flow regimes actually occuring, boiling delay and thermodynamic non-equilibrium. In particular the measurement of the pressure a few millimetres upstream of the

outlet plane is subject to considerable errors.

3. The pressure distributions resulting from the impact of the two-phase jets on the baffle plates are similar to normal distribution curves. For all diameters tested it was possible to show the radial pressure distribution as a two-parameter-function $f\ (\frac{r}{D}, \frac{A}{D})$ if the local pressure on the plate was related to the maximal value in the jet centre. They are similar to corresponding normal distribution curves at the impact of one-phase jets - only the parameter distance between outlet and the baffle plate differs by factor of 6 to 10. The radial expansion of two-phase jets, disturbed by the baffle plates, and that of undisturbed two-phase jets are almost the same - they differ by factor of 5 from the corresponding one-phase jets. Finally, it must be mentioned, that the two-phase jet has no conical jet core and that it loses its effectiveness after a distance of about 5 D. In the case of a one-phase jet the core has a length of 5 D and a total range of 30 to 40 D.

4. As to the jet load and thus to the design of the structures subject to the impact of two-phase jets it is necessary to mention the differences between the individual break shapes. In the case of the guillotine break a thrust force acts on the outlet pipe and an equal and opposite force acts on the baffle plate. For a slot break the jet causes both thrust forces and forces on the remaining metal joint. The thrust forces are the same for both break types. For the outlet of saturated pressurized water with large outlet diameters they can be sufficiently described by an impulse balance at the outlet, assuming homogeneous flow distribution.

The conditions for annular flow must be taken into account for small outlet diameters. The method of considering the force effect by means of the thrust factor, already introduced in literature is especially suitable to show the influence of parameters on the forces. Beside the value of the pipe resistance coefficient, these are the stagnation pressure, the outlet diameter and therefore the flow regime. The radial forces appearing in slot break geometries reach their maximum value when the inlet cross sections are equal to the break outlet cross section. The radial forces compared with the axial thrust forces show different results for the geometries investigated. For tests on geometry I the radial force is always smaller than twice the axial thrust force. When diverting two jets by 90 ^{o} without loss it is possible to reach maximal $R_L / (2\ R_Q) = 1$ for the ratio of radial force to the sum of the axial thrust forces. For tests on geometry II, however, the radial forces assume values that can be higher than twice the axial thrust force, $R_L / (2\ R_Q) > 1$.

This is due to the fact that - in case of outlet cross section and inlet cross sections being equal - at geometry I only low pressure due to the expansion in all directions perpendicular to the pipe bottom becomes effective. In the case of geometry II a considerable additional force (due to the pressure which at A/D = 0,61 is larger than the critical pressure) exists on a surface, which for this kind of break simulation can become larger than the remaining metal joint.

Next Steps

Work has been completed on this project.

Relation with Other Projects

RS 16 Investigation of the Phenomena Involved in the Depressurisation of Water-Cooled Reactors

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

Reference Documents/Degree of Availability

W. Kastner, R. Eichler, K. Riedle

Experimentelle Untersuchungen zu Kräften kritischer Zweiphasenstrahlen bei Quer- und Längsrissen von Rohrleitungen

Abschlußbericht zum Förderungsvorhaben BMFT RS 93,

KWU-Erlangen, (Oktober 1975)

Company Confidential

Classification: 7.1	
Title 1 (Original Language): Sicherheitsexperimente im Kernkraftwerk Marviken, Schweden (RS 33 - I.1.4., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GKSS, Geesthacht
Title 2 (english): Joint Reactor Safety Experiments in the Power Station of Marviken, Sweden	Project Leader: Franke
Initiated (Date): December 1971 Completed (Date): March 1975 Status: finished Last Updating (Date): December 1975	

1. General Aim

The Marviken experiments are carried out to investigate the behaviour of BWR containment with pressure-suppression system after a loss of coolant accident. The results are used to check analytical models for the design of containments and pressure-suppression systems.

The following institutions and organisations are participating in the program:

- Aktiebolaget Atomenergi, Sweden
- Atomenergikommissionen, Denmark
- Ministry of Trade and Industry, Finnland
- Institut for Atomenergi, Norway
- Gesellschaft für Kernenergieverwertung in Schiffbau und Schiffahrt mbH, Federal Republic of Germany.

In March 1973 the United States Atomic Energy Commission (US AEC), in May 1974 the Japan Atomic Energy Research Institute (JAERI) and in June 1974 the Comissarat à l'Energie Atomique (CEA) joint the project.

2. Particular Objectives

There were 20 blowdown experiments scheduled, 16 of which have been preliminarily defined in regard to the initial parameters.

The experiments were carried out under operational conditions, for which the reactor originally had been laid out (50 bar, 264 ^{o}K).Steam and feed water pipe ruptures with fracture areas up to 710 or 280 cm^2 respectively, were scheduled. For the investigation of the pool behaviour its initial temperature was varied between 25 and 60 ^{o}C.

3. Experimental Facilities and Program

See Report IRS-F 24 (1974) p. 136

4. Project Status

4.1 Progress up to Date

The preparation of the containment for the tests and the conduction of preliminary experiments to test the facility, including the instrumentation and the data recording system, were carried out in the phase 1.12.1971 - 31.9.1972. In the range 1.10.1972 - 14.5.1973 16 blow-downs were performed. After termination of the experimental phase the evaluation and reporting of the results were started. Till to the end of March 1975 all reports were completed and the project terminated.

4.2 Essential Results

4.2.1. Containment reponse to loss of coolant accident

4.2.1.1. Basis for the computer code testing

It may first be stated that the experiments have provided information which can be used for the verification of computer codes for calculating the pressure build-up and temperature rise in a pressure-suppression containment, which was actually the main objective fo the series of blowdown experiments.

Dry containment analytical models may be tested against the short-term pressure transients in the drywell compartment, before the onset of venting to the wetwell compartment.

4.2.1.2. The project computations

The limited testing of computer codes for containment response analysis whithin the project has shown that these can reproduce the containment pressure and temperature conditions fairly well. The current opinion that computer codes give a conservative prediction of the containment pressure response was further strengthened, although it is apparent that nonconservative features exist in the analytical models. Consequently, there are some points in the models which can be improved so as to yield more realistic calculations and thereby more reliable determinations of the safety margins.

4.2.1.3. The Tests as demonstrations

The sixteen simulated blowdown tests, carried out under different conditions, have repeatedly demonstrated in full scale the ability of a water pool based pressure-suppression containment to receive large quantities of mass and energy at varying rates, condense the steam and control the containment pressurization, as intended. The tests have, in that sense, confirmed the applicability of pressure-suppression systems,as previously stated on the basis of e.g. the small-scale Humboldt Bay and Bodega Bay tests.

The occurrence of minor damage, which in a real accident situation could have been a potential risk, indicated the importance of carefully ensuring the integrity of the containment and vital safety systems; in this context it is necessary to point out that the assessment and licensing of the Marviken containment systems had never been completed.

4.2.1.4. The presence of pressure oscillations

A specific phenomenon demonstrated in these tests was the presence of pressure oscillations of significant amplitudes, not only in the wetwell water pool but also in the vent pipe system and in the drywell compartment.

Since the oscillations impose additional loadings on structures in the containment, it seems necessary to gain some insight into their nature or cause. This does not seem possible, however, on the basis of the limited information from this series of tests; the data acquisition system was not designed for the recording of pressure oscillations.

4.2.2. Jodine and Xenon experiments

4.2.2.1. Leak testing of the containment structures

The absolute method and a radioactive tracer isotope method using ^{133}Xe were applied in two leakage rate tests under static conditions. From the investigation it may be concluded that:

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The leakage of ^{133}Xe was somewhat higher than that of dry air, the difference being 30 - 50 %.
The absolute method is to be preferred for a proof leakage rate test. This test is carried out infrequently, so that the long time required for the test is not a serious drawback. The isotope method however, is very promising, and might well be applied for intermittent controls of containment tightness as an alternative to the absolute method. The application of the isotope method is limited to cases where a vessel can be completely surrounded by a limited volume. It must be possible to take representative samples of this volume, preferably by ventilation and subsequent sampling of the contained air.

4.2.2.2. Containment leakage during accident conditions

The leakage rate in runs with elemental iodine is apparently c. 500 times smaller than in runs with methyl iodide. The reason for this is that iodine is removed from the air, probably by deposition on the containment walls and washout. Another contributing reason may be that leakage paths for elemental iodine are blocked after blowdown. A small fraction of the iodine is converted into organic iodine compounds and this fraction may be responsible for the leakage.

The leakage rate of methyl iodide is 2 - 3 times lower than that of xenon if the activities are injected before or during the blowdown. This difference ist, in part, accounted for by removal of the iodine from the air space as a result of retentions in the wetwell pool. Spray cooling also removes some methyl iodide.

The leakage rate of xenon determined under static conditions was about five times higher after all the blowdown runs than before the runs. The reason for this may be an increased leakage through the structure and/or the conduits through the walls. New conduits were introduced during the experimental period. No tendency was found towards an increasing leakage rate as the number of experiments performed increased.

The leakage rate under blowdown conditions was higher than that determined under static conditions both before and after all blowdown runs, the leakage rate of xenon increased with time in parallel with a decrease in the total pressure in the containment. The reason for the increase in the leakage rate is not clear. It may be connected with the effects of condensation and evaporation of a water film in the leakage paths as steam and heat penetrate into the containment walls, and/or a decreased tightness of the containment structure. A timedependent distribution of xenon in the drywell atmosphere might also contribute to an increasing leakage.

4.2.3. Component Tests

The component tests were carried out in order to find out how the different components stood up to the atmosphere under blowdown conditions Most of the components were left inside the PS containment between the blowdowns, and for some days before and several days after each blowdown they were exposed to an atmosphere with a high humidity and an increased temperature.

In addition to certain components which were selected and installed for testing, some of the components originally installed were also included in these component tests. These tests involved the registration of signals from measuring devices and the operation of valve actuators during the blowdowns.

The following groups of components were tested:
Electrical equipment(Measuring devices, Switches, Installation material, Electric motos, Electric valve actuators)
Mechanical equipment (Pneumatic valve actuators, Hydraulic valve actuators, Heat insulation)
Building materials (Surfacers and paints for concrete).

The aim of the tests was not to obtain quantitative results. The tests were principally of the go/no-go type, and the investigations and data collecting were carried out accordingly. In some cases it was possible to obtain quantitative information from the reported results; however, the intention was not to evaluate in the sense that the cause of a malfunction would be discussed in detail. For more information about the results, reference should be made to the complete test reports.

5. Next Steps

In order to get more detailed information about the pressure oscillations, a new project with specially instrumentated experiments is founded (RS 33a).

6. Relation with other Projects

RS 50: Investigation of the phenomena occuring with a multicompartment containment after rupture of the primary cooling circuit in water-cooled reactors.
Batelle Institute, Frankfurt, 1971 till 1976.

RS 33a: Joint Reactor Safety Experiments in the Power Station of Marviken, Sweden Part 2 - Oscillationexperiments.
Gesellschaft für Kernenergieverwertung in Schiffbau und Schifffahrt mbH, Geesthacht, 1975 till 1977.

7. Reference Documents

1. Reactor Safety Experiments in the PS-Containment of the Marviken Power Station, H.G. Thorén et al., AB Atomenergi, Studswik/Sweden July 1971. (written in english)

2. Annual Report A 72 — IRS-F-12 written in english
   Annual Report A 73 — IRS-F-18 written in english
   Annual Report A 74 — IRS-F-24 written in english

3. Intermediate Report December 1971 till June 1973 — IRS-F-16 written in german

4. Full-Scale Containment Experiments performed in the Marviken Power Station, H.G. Thorén et al., AB Atomenergi, Studsvik/Sweden 1973. (written in english)

5. About 15 MXA-Reports from the project, table 1, list of MXA-reports, in IRS-F-24 p. 140-141. (written in english)

8. Degree of Availability

References 1 - 4 are free available. Reference 5 ist available only to the contracting parties and free available in about 1 year.

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Classification: 7.1	
Title 1 (Original Language): Sicherheitsexperimente im Kernkraftwerk Marviken, Schweden. Teil 2 - Schwingungsuntersuchungen (RS 33 A - I.1.4., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GKSS, Geesthacht
Title 2 (english): Joint Reactor Safety Experiments in the Power Station of Marviken, Sweden. Part II - Oscillation Experiments	Project Leader: W. Franke
Initiated (Date): February 1975 Completed (Date): March 1977 Status: continuing Last Updating (Date): December 1975	

1. General Aim

In February 1975 a second agreement for full scale containment experiments in the Marviken power plant was signed between Denmark, Norway, Germany, Finland, Sweden, France, Japan and USA. The containment response tests are mainly carried out to investigate the pressure oscillation behaviour of the containment with pressure-suppression system after a loss of coolant accident and to make measurements of parameters thought to be important for the understanding of these phenomena. After the preparation phase (1.2.75 - 30.1.76) the Blowdowns shall be carried out during the experimental phase (1.2.76 - 30.9.76). The evaluation and reporting stage shall be started on 1.10.1976.

2. Particular Objectives

There were 8 Blowdown experiments (Nr. 17 till 24) scheduled, all of them have been preliminarily defined in regard to the initial parameters.

The experiments were carried out under operational conditions, for which the reactor originally had been laid out (50 bar, 264 ^{o}K). Pipe ruptures with fracture areas up to 900 cm^2 and Flowrates up to 2000 kg/sec were scheduled. For the investigation of the pool behaviour the submergence depth (0,5 - 2,8 m) and ventarea (4 - 2 m^2) is varied.

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3.1 Experimental Facilities

The experimental facility is the decommissioned reactor of Marviken, Sweden. This facility has been designed and constructed as a 140 MWe direct circuit boiling heavy water reactor. Soon after completion it had been decommissioned and offers the opportunity to conduct full scale safety experiments.

3.2 Research Program

The experimental program i.e. the choice of testparameters, is done due to the following reasons: .

Blowdown 17. This is a close simulation of Blowdown 10 in the earlier series which had indicated significant pressure oscillations. The test should reproduce those conditions and thereby provide a data base for instrumentation check-out.

Blowdown 18. This is similar to Blowdown 17 but with a reduced vent flow area. The purpose is to get an increased vent mass flux. The blocking of the vent pipes should be arranged so as to provide a simple symmetrical and compact pattern. The arrangements will be compatible with the new pool geometry in Blowdown 21.

Blowdown 19. This test will utilize prepurging of the drywell air in order to investigate the influence on oscillations of noncondensibles in the vent flow.

Blowdown 20. This test will be run with a reduced submergence depth associated with a reduced pool water mass. A significant change of submergence depth is wanted and for geometrical reasons can only be obtained by reducing the pool water volume. The lower temperature indicated at the start of the prepurging is meant to compensate for the decreased pool mass.

Blowdown 21. This test will be carried out with the pool volume reduced to about three quarter of the normal volume. The reduction will be achieved by introducing an internal structure forming a new pool around the unblocked vent pipes and thus reducing the pool cross section. The smaller pool mass will result in an increased temperature rise. The reduced pool mass may also have an effect on the pressure oscillations.

Blowdown 22. This test should show the effect of air on pressure oscillations as compared to Blowdown 21. The dynamic loads on thestructures above the pool can be compard to the results from Blowdowns 17 and 18.

Blowdowns 23 and 24. These tests intended to show effects on the pressure oscillations due to changes in vent system geometry. The specific changes should be determined on the basis of further theoretical studies.

4. Project Status

4.1. Progress up to Date

The preparation of the containment for the tests and the conduction of preliminary experiments to test the facility, including the instrumentation and the data recording system, were carried out in the phase 1.2.1975 - 31.12.1975.

4.2. Essential Results

As to the end of the reported period no experiments were carried out at this place the instrumentation installed (during the year 1975) shall be described.
The main part of measurements consists of the pressure, differential pressure and temperature measurements. The semistatic measurements of these quantities will be recorded by a VARIAN 620/L process computer on digital tape while the so-called dynamic measurements will be recorded by a Pulse Code Modulation (PCM)-system. In addition, the pressure in the pressure vessel, drywell and wetwell and the differential pressure in the pressure vessel - one channel for each - are recorded by pen-recorders. These latter measurements are primarily for controlling purposes in the control room.
The water levels in the wetwell pool and vent pipes are measured with level probes of spark plug type. The signals from these measurements are recorded by PCM-system. In measuring the water level in the pressure vessel a special type of level probes are used and the recording is made on a light beam recorder.

The impact load in the wetwell will be investigated by using nine accelerometers and three strain gauges all of which are recorded on PCM-system.

In the wetwell the behaviour of the water is observed via two TV-cameras. At the same time the sound in the wetwell is recorded by a microphone. Another microphone is placed near the room 122 (break room) in the drywell.

The flow related measurements in the break pipe and the downcomers will be performed using the radio tracer method and the infrared-absorption-technique.

5. Next Steps

It is expected to start the experimental phase in February 1976 and carry out all 8 experiments till end of September 1976.

6. Relation with other Projects

RS 50: Investigation of the phenomena occuring with a multicompartment containment after rupture of the primary cooling circuit in water-cooled reactors.
Battelle Institute, Frankfurt, 1971 till 1977.

7. Reference Documents

Commentary on the proposal for The Marviken II containment response tests, AB-Atomenergi Studsvik 1974.

8. Degree of Availability

The above mentioned reference is available only to the contracting parties.

	$p_{R,0}$ /bar/			ϑ_{Pool} /K/			$\dot{m}/A$ /kg m^{-2} s^{-1}/		
Parameter	2,00	2,25	2,50	30	40	50	75	107	125
ν /Hz/	4,44	4,48	4,50	4,50	4,35	4,32	4,25	4,50	4,71
$2\,\Delta p_1$ /bar/	0,35	0,30	0,25	0,25	0,20	0,15	0,30	0,25	0,10
$2\,\Delta p_2$ /bar/	0,80	0,60	0,45	0,45	0,30	0,25	0,50	0,45	0,20
$\overline{R}$ /m/	0,39	0,39	0,40	0,40	0,41	0,42	0,34	0,40	0,41
$2\Delta R$ /m/	0,23	0,21	0,11	0,11	0,09	0,08	0,19	0,11	0,06

Table 1: Parameter variations for the MARVIKEN geometry.

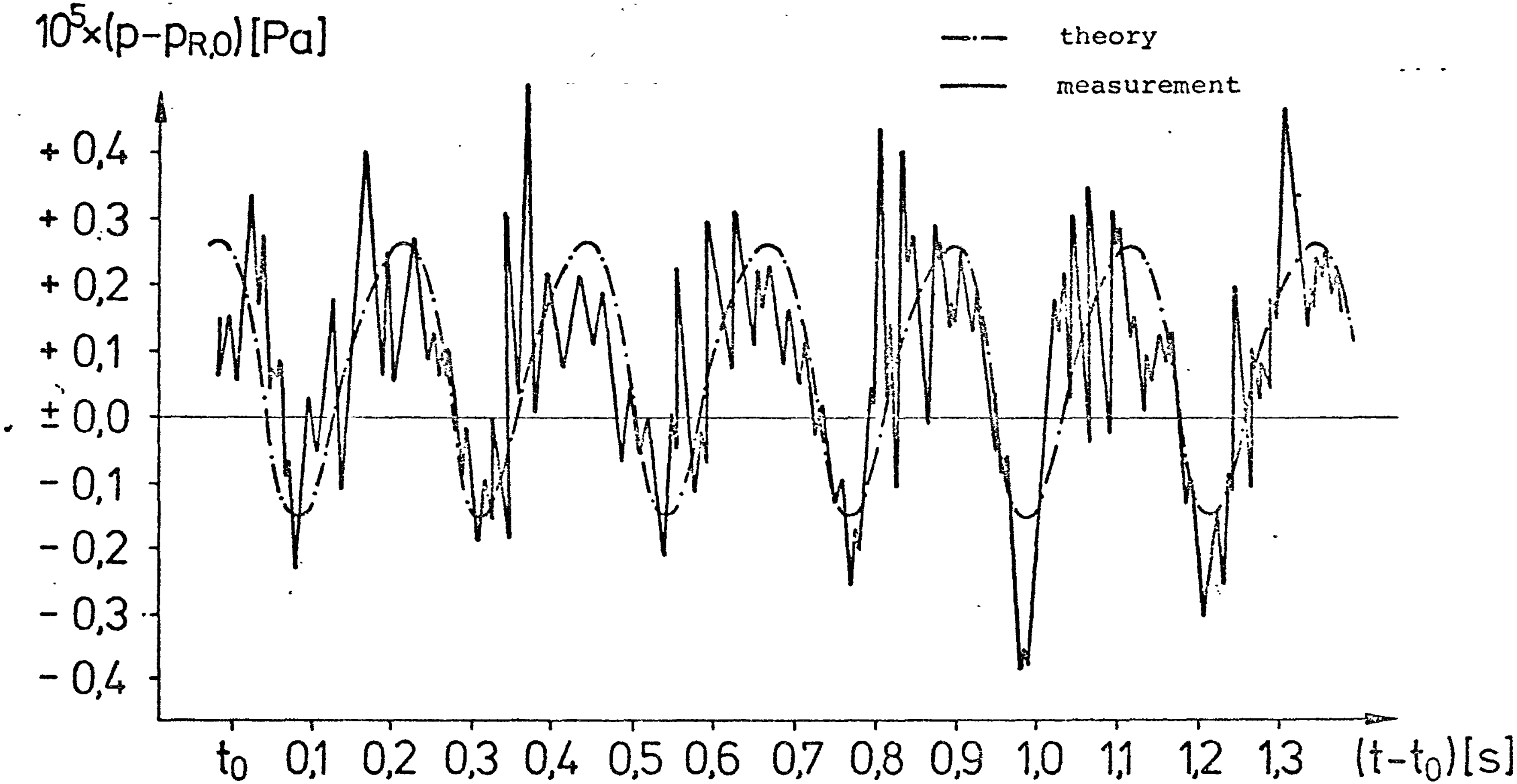

Fig. 1: Comparison between the theoretical and experimental pressure transients in the wetwell during Marviken-13 blowdown.

Classification: 7.1	
Title 1 (Original Language): Sicherheitsexperimente im Kernkraftwerk Marviken, Schweden Teil 2 - Schwingungsuntersuchungen (RS 33 A - 1.1.4 Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GKSS
Title 2 (English): Joint Reactor Safety Experiments in the Power Station Marviken, Sweden. Part 2 - Oscillation-experiments	**Project Leader:** Franke
Initiated (Date): February 1975 **Status:** Experimental phase finished **Completed (Date):** March 1977 **Last Updating (Date):** December 1976	

1. General Aim

In February 1975 a second agreement for full scale containment experiments in the Marviken power plant was signed between Denmark, Norway, Germany, Finland, Sweden, France, Japan, USA and the Netherlands. The containment response tests are mainly carried out to investigate the pressure oscillation behaviour of the containment with pressure-suppression system after a loss of coolant accident and to make measurements of parameters thought to be important for the understanding of these phenomena.

2. Particular Objectives

There were 8 Blowdown experiments (Nr. 17 till 24) scheduled, all of them have been preliminarily defined in regard to the initial parameters. One Blowdown (Nr. 25) with one ventpipe in a single-cell has been added.

The experiments were carried out under operational conditions, for which the reactor originally had been laid out (50 bar, 264 ^{o}K). Pipe ruptures with fracture areas up to 900 cm^2 and Flowrates up to 100 kg/cm^2 s were scheduled. For the investigation of the pool behaviour the submergence depth (0,5 - 2,3 m) and ventarea (4 - 2 m^2) were varied.

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ing stage has started.

6. Results

Blowdown 17: This test showed a similar oscillation behaviour as Blowdown 1o of the earlier series and gave an instrumentation check.

Blowdown 18: Due to reducing the vent flow area a higher vent Massflux is achieved (100 kg/cm^2 sec.) and also a higher Watersurge. The oscillations were similar to Blowdown 17. The test gave additional dates for impact loads of the internal structures.

Blowdown 19: Due to the prepurging the results of this test show the behaviour of condensation without air:There was no Watersurge. The harmonic oscillation was superposed by high frequency peaks, due to collapsing steambubbles. These peaks were observed only local; i.e. the vent pipes are not coupled.

Blowdown 2o: The test was again without Watersurge. The reduced submergence depth gave smaller peaks compared with Blowdown 19, due to a smaller water depth. Due to the smaller watermass at the end of this test the pool temperature was 80 ^{o}C.

Blowdown 21: Due to the reducing of the pool volume at the end of this test the pool temperature was 104 ^{o} C. The oscillation amplitudes were smaller than those at Blowdown 18. An essential result was that also in the corner volumes of the drywell were blown free of air during Blowdown totally.

Blowdown 22: At this Blowdown the same oscillation behaviour as in Blowdown 19 was achieved. When reducing the Massflux from 100 to 4o kg/cm^{2}sec. the amplitudes of the pressure oscillation decreased from 1 to o.1 bar.

Blowdown 23: This test gave a verification of Blowdown 22. Due to the slower Massflux variation from 4o kg/cm^{2}sec. to 0 the oscillation amplitudes were lower.

Blowdown 24: Due to the changes in the vent system geometry the observed frequences of the pressure oscillation changed from 4 to 3.6 Hz and from 30 to 27 Hz.

Blowdown 25: The observed harmonic pressure oscillations in the pool and in the single cell were about 4, 8, 10 and 26 Hz with 1.5 to 2 times larger amplitudes in the single cell. The peak amplitudes in the single cell were also double as high as in the pool. The slopes of the pressure peaks were the same as observed in other single cell experiments.

7. Next Steps

The evaluation and reporting stage shall be terminated at the end of the first quarter 1977.

8. Relation with other Projects

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9. Reference Documents

1. D. Slaughterbeck, L. Ericson, H. G. Thorên:
   Nuclear Safety Experiments in the Marviken Power Station.
   Nuclear Safety Magazine (under printing)(Language English).

2. Marviken Interim Reports: 9.2.1 9.2.8 Blowdown 17 bis 24, 9.2.9 Measurement Systems, 9.2.10 Data Accuracy. (All reports written in English.)

10. Degree of Availability

The above under pos. 8.2 mentioned references are available only to the contracting parties.

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Classification: 7.1	
Title 1 (Original Language): Entwicklung und Verifizierung von Rechenprogrammen zur Beschreibung von Containment-Problemen (ATT 085 A - I.1.4., Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (English): Development and Verification of Codes Simulating Containment Problems	Project Leader: **Dr. H. Karwat**
Initiated (Date): Completed (Date): Status: continuing Last Updating (Date): December 1976	

1. General Aim

General aim of this work is the application of existing codes to verify results of relevant experiments and to improve existing codes which simulate effects associated with the pressurization of containment systems and its long time behaviour with respect to pressure, temperature and hydrogen distribution of the loss-of-coolant accident.

2. Particular Objectives

During the reporting period the work concentrated on pre- and recalculations of containment experiments carried out at the Battelle-Institute in Frankfurt, the improvement of codes simulating pressure suppression system effects and on a code simulating hydrogen distribution within containment systems resulting fron radiolytic decomposition of water during the emergency core cooling process.

3. Experimental Facilities and Program

Experimental work carried out in connection with the pre- and recalculation mentioned above is reported unter the headline "Investigation of the phenomena occurring within a multi-compartment containment after a rupture of the primary cooling circuit in water-cooled reactors" (RS 50) by the Battelle-Institute, Frankfurt.

4. Project Status

4.1 Progress to Date

A new lumped-parameter model for the simulation of full pressure containment systems has been developed. The new model COFLOW takes into account kinetic energy within the momentum and the energy equation and has been successfully used to recalculate the results of blowdown experiments carried out at the Battelle-Institute, Frankfurt. Discrepancies observed between experiment and analysis using the code ZOCO-VI have been considerably reduced with the code COFLOW.

During the reporting period work has been started to simulate the behaviour of a pressure suppression system under different loading conditions. A new model DRASYS is in development which is supposed to simulate integral as well as differential effects of pressure suppression systems. The basic concept consists of a lumped parameter model in which two types of typical nodes describing the drywell and the wetwell, can be used in any arbitrary connection. The two phases vapour and water are considered to be in thermodynamic equilibrium within the drywell. Within the wet well nonequilibrium processes are assumed to exist if interphases are formed between water and steam or steam/air mixtures during condensation. Significant effects like vent clearing, pool swelling, condensation oscillations and chugging should be taken into account. The code DRASYS will be of modular structure and will incorporate subprograms like KSWING for the simulation of separate effects expected during the fluiddynamic transient.

Work on the code RALOC simulating hydrogen distribution within a containment system has been continued and resulted in a working version which was used to perform parametric studies describing different important parameters for hydrogen distribution.

A proposal has been filed for an experimental program to study the basic phenomena of hydrogen distribution within a subdivided model containment. It is proposed to use the Battelle model containment for studies to the above mentioned problem. The parametric studies have shown a certain necessity for such experimental activities to limit the range of parameters influencing the calculated results.

4.2 Essential Results

The analytical verification of containment experiments carried out in the Battelle facilities in Frankfurt has been continued. Both codes, ZOCO-VI as well as COFLOW have been used. COFLOW showed much better agreement between experimental and analytical results in specific in those areas where deviations have been observed due to kinetic energy effects. In connection with the next tests planned to be performed at the Battelle-Institut in Frankfurt (RS 50) precalculations have been performed for a series of tests (D-series) which is characterized by a release of pure steam during the initial phase of the blowdown. The purpose of these tests is to eliminate in certain experimental periods water/steam phase separation effects.

5. Next Steps

Development work with the aim to simulate fluiddynamic transients in full pressure containment systems is considered to be terminated. Main efforts will be directed towards the verification of experimental results by use of existing codes like ZOCO-VI or COFLOW. Work in connection with a comprehensive description of pressure suppression systems will continue within the next year. Work in connection with simulation of hydrogen distribution will continue with respect to numerical improvements of the code and analytical work in connection with planning and verification of relevant experimental activities.

6. Relation with Other Projects

Strong relations exist between code verification work and the experimen-

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tal work carried out by the Battelle-Institute, Frankfurt (RS 50), and the HDR program.

7. Reference Documents

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8. Degree of Availability

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Classification: 7.1	
Title 1 (Original Language): Auslegung, Vorausberechnung und Auswertung der HDR-Blowdown Experimente, sowie Weiterentwicklung und Verifizierung fluid-strukturdynamischer Codes zur Beanspruchung von RDB-Einbauten beim Blowdown. (PNS 4221/4223-I.1.4, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** **ORGANIZATION:** GfK/PNS
Title 2 (English): 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.	**Project Leader:** Dr. Krieg Dr. Schlechtendahl
Initiated (Date): Oct. 74 **Status:** Continuing **Completed (Date):** 1980 **Last Updating (Date):** Dec. 1976	

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 Test apparatus for HDR natural convection

Ad 3.2 Codes currently applied: YAQUIR(fluid), STRUDL/Dynal (structure)
Codes under development: FLUST, SOLA-DF (fluid), CYLDY2 (structure)
STRUYA(YAQUIR + CYLDY2), FLUX (coupled fluid + structure)

Ad 3.3 Facility for fast pressure relief of a vessel. Shallow water table for analogy experiments

5. + 6. Progress to Date and Results

Ad 3.1 Detail design drawings have been provided for the complete test core barrel including the upper and lower core barrel flange. A detailed (nonlinear) stress analysis for the upper flange was carried through and a riging diagram has been established. Peak stresses added by the blowdown load are below 100 kp/cm^2. Furthermore the stability of the HDR pressure vessel and its support under blowdown loading could be shown. The heating and cooling system for establishing the temperature field in the vessel was analysed with respect to stability and heat dissipation. For experimental verification natural convection tests were defined for the HDR facility and a smaller scale test rig was designed.

Predictive calculations were performed and documented for the reference test No 3.

Ad 3.2 The fluid code YAQUIR has become a production program. Test calculations were carried out for various cases including a comparison with results of the REXCO as well as the PWINCAD code and with potential flow theory. Difference equatior for the drift flux model have been established. All administrative and system routines for the systems code FLUST have been programmed and tested.

After the code CYLDY1 for the structural dynamics of the core barrel had failed to provide satisfying results, major changes of the solution method led to an improved code version CYLDY2. Test calculations with CYLDY2 which follows a semi-analytical concept (superposition of eigenfunctions) indicate considerable advantages in spatial resolution and computer time in comparison to the finite element code STRUDL/DYNAL. The applied loadings were non-axisymmetric, transient pressure peaks. Fluid structure coupling methods have been derived. A modified YAQUIR version (STRUYA) has been sucessfully coupled with simple structural models (vessel, beam). Coupling with CYLDY2 is underway. Eigenfrequencies of the complete system (including the fluid inside the core barrel) have been obtained with FLUX under the assumption of an incompressible fluid.

Ad 3.3 Manufactoring drawings have been prepared for a fast pressure relief facility to be used for small scale experiments in coupled fluid-structural dynamics. Test conceptions are developed. Shallow water analogy experiments were performed in various geometries indicating the usefulness of this technique for code verification.

7. Next Steps

Ad 3.1 Predictive calculations will be performed with YAQUIR (fluid) and with STRUDL/DYNAL and CYLDY2 (structure) for the significant test runs at the HDR-facility. The pressure field for the structural computation will be taken from YAQUIR and WAMMOD. In addition, calculations using the STRUYA-CYLDY2 coupling will be performed. Special structural vibration tests will be performed for the HDR.

Ad 3.2 Code development will proceed for FLUST, SOLA-DF and FLUX. In particular the influence of different grid schemes upon the results will be investigated. CYLDY2 and STRUDL/DYNAL will be compared. Extension of CYLDY to include nonlinear material properties will be considered.

Ad 3.3 The apparatus for fast pressure relief will be fabricated and first small scale tests will be performed. A test program for investigation of two phase flow with pronounced acceleration are going to be prepared. Shallow water experiments will be performed in HDR like geometry.

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, B. Laursen: Auslegung des Kernmantels und der Kernmanteleinspannung für die DWR-Blowdown-Versuche an HDR-Reaktor. Transiente Analyse: Jan. 76

R. Krieg: Obere Abschätzung für die horizontale Verschiebung des HDR-Reaktordruckbehälters, hervorgerufen durch innere Kräfte bei den geplanten Blowdown-Versuchen, Mai 76

G. Hailfinger, R. Krieg: Elastisch-Plastische Spannungs- und Verformungsanalyse für die Kernmanteleinspannung bei den DWR-Blowdown-Versuchen am HDR-Reaktor, Okt. 76

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U. Schumann: Zur Kopplung unterschiedlich diskretisierter Fluid- und Strukturdynamik (June 76)

G. Enderle: Entwurf eines zweidimensionalen Modells für den Fluiddynamikcode FLUST1 (July 76)

E.G. Schlechtendahl: Überlegungen zur Datenstruktur dreidimensionaler Modelle im Code FLUST unter topologischen und geometrischen Gesichtspunkten (Sept. 76)

K. Stölting: Vorausberechnung der HDR-Blowdown-Untersuchungen (Fall 3) mit dem Rechencode YAQUIR (Nov. 76)

K. Stölting: Vergleich mehrerer Blowdown-Berechnungen mittels normierter Druckentlastungs- und Ausströmungsverläufe (Dez. 76)

10. Degree of Availability of the Reports

The reports listed under item 9 are unpublished.

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Classification: 7.1	
Title 1 (Original Language): Meßtechnische Erfassung und Auswertung des dynamischen Verhaltens der Versuchseinbauten im Reaktordruckbehälter (RDB) des HDR im Rahmen der HDR-Blowdown-Versuche (PNS 4222 - I.1.4, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: ORGANIZATION: GFK/PNS
Title 2 (English): Experimental Data Aquisition and Processing of the Dynamic Behavior of the Pressure Vessel Test Internals in the HDR-Blowdown-Experiments	Project Leader: K.D.Appelt
Initiated (Date): 1.1.1975 Status: Continuing Completed (Date): 1978 Last Updating (Date): December 1976	

1. General Aim

Methods and codes are presently available which allow LWR reactor design in such a way that a blowdown accident due to failure of one of the primary coolant circuits can be safely contained. However, in the case of such a blowdown the degree is not yet known up to which the load carrying potential of the plant will actually be utilized. To investigate the safety margins in detail, the conservative models must be replaced by more realistic methods and codes which will have to be verified by appropriate experiments.

2. Particular Objectives

The dynamic stresses and deformations of the vessel internals must be investigated in particular. To get realistic results, the phenomenon of fluid-structure interaction has to be included, i.e. the feedback of structural flexibility upon the pressure field imposed.
Consequently, in appropriate experiments the variables of state in the fluid as well as the dynamic reactions of the loaded structures must be recorded simultaneously by measurement technology. It is the objective of this task to measure the structure response (strains, accelerations, displacements) in the planned blowdown tests to be performed in the HDR facility and to evaluate the experimental data. Experience gathered in other similar projects have shown that the instrumentation used must

satisfy very stringent quality requirements if an excessive number of failures and a negative influence on the experimental results, respectively, is to be avoided.

3. Research Program

The experimental program includes developing an appropriate instrumentation for large-scale experiments, testing this instrumentation and making it available for the scheduled experiments.

4. Experimental Facilities, Computer Codes

Experiments allowing to provide an adequate instrumentation without creating scaling problems are most important for the investigations described above. Therefore, the (out-of-service-) HDR-reactor, which had been converted into an almost full-scale model of a PWR-reactor, is used as an appropriate experimental facility. Details can be found in the annual report of the PNS 4221 project. In the laboratory an autoclave and shaker system is used for the development and testing of the instrumentation.

5. Progress to Date

The autoclave was completed and accepted by the Technical Inspectorate (TÜV). The associated electromagnetic shaker has been delivered. Mounting has started on a vibration insulated concrete block built for this purpose.

The test program for the examination of the prototype measurement transducers has been specified in detail and the necessary auxiliary test equipments were designed and are being fabricated. A first series of laboratory basic tests with three prototype displacement transducers were carried out. Determinations included the dependency of zero, sensitivity and linearity on the temperature in a furnace at atmospheric pressure. In HDR the interferences were investigated using different types of measurement cable.

A concept of instrumentation was elaborated for the measurements planned in addition on the thermal stratification in the HDR pressure vessel and the specifications were defined which are required for the invitation of tenders and placing the order for this measurement equipment.

Computer code "DYMECO"

was developed to correct the dynamic measurement errors occurring during. blowdown tests. Testing of this program was started.

6. Results

The laboratory basic tests so far performed on three prototype displacement transducers exhibited an error behavior which was largely within the tolerance range of the specification. However, these results do not yet permit a conclusion to be drawn about the service life and the quality of measurement transducers since the decisive tests are still to be made.

The feasibility in principle of the measurement chains was determined which allow to perform the measurements with the required time behavior ($\Delta t_{Ph} \leqq 500\ \mu s$).

Measurements of the electromagnetical background noise in HDR and the shielded capability of different types of measurement cable, produced results which do not suggest major difficulties to be expected.

7. Next Steps

After the operation will have started of the autoclave test system, the continuing tests with the prototype transducers will be carried on under the test program. This includes tests under transient temperature and pressure conditions without mechanical loads as well as dynamic tests performed both in air and in water at elevated pressure and temperature. To the extent available, the whole original measurement chain will be used in these tests so as to obtain reliable information about the real dynamic behavior. At the end of the test series, the prototype transducers will be subjected to high acceleration loads allowing experimental determination of the mechanical ultimate loadability. The test results will be the most important aspect to be taken into account in the selection of the definite HDR core shell instrumentation.

8. Relation with Other Projects

The data on the structure response under blowdown conditions, which were found in the experiment and subsequently processed, will be made available to the PNS 4223 project for future use.

9. References

PNS-Arbeitsbericht Nr. 47/75 (August 1975 +)

1st semiannular Progress-Report PNS 1/1976 ++)

2nd semiannular Progress-Report PNS 2/1976 ++)

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8. Degree of Availability

+) Restricted distribution

++) free

Project Nuclear Safety, Gesellschaft für Kernforschung Karlsruhe.

Classification: 7.1

Title 1 (Original Language): Weiterentwicklung und Verifizierung gekoppelter fluid- und strukturdynamischer Codes zur Analyse der dynamischen Spannungen und Verformungen von RDB-Einbauten bei Kühlmittelverlustunfällen in LWR (PNS 4223 - I.1.4, Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Development and Verification of Coupled Fluid-Structural-Dynamic Codes for Stress and Deformation Analysis of Reactor Vessel Internals under Blowdown Loading	**Project Leader:** Dr. Krieg, Dr. Schlechtendahl
Initiated Date): [illegible]. 1974	**Completed (Date):** 1979
Status: continuing	**Last Updating (Date):** Dec. 1975

1. General aim

Up to date methods and codes are available to design LWR-reactors in such a way, that a blowdown-accident due to failure of one of the primary coolant circuits can be safely contained. However, in the case of such a blowdown it is not known so far, up to which degree the load carrying potential of the plant will actually be utilized. Hence, on present knowledge a further development to probabilistic safety analysis is not yet possible. To investigate the safety margins in detail more realistic methods and codes verified by appropriate experiments rather than the well known conservative models must be developed.

2. Particular objectives

In particular to assess the proper function of both the shutdown-system and the emergency-cooling-system after blowdown, the dynamic stresses and deformations of the vessel internals must be investigated. To get realistic results the phenomenon of fluid-structure interaction, i.e. the feedback of structural flexibility upon the imposed pressure field has to be included. That means, in experimental investigations the thickness of relevant structural members should be relatively small as to allow for appropriate flexibilities, while with respect to theoretical investigations the development of computer codes for coupled fluid and structural dynamics is mandatory.

3.1 Experimental facilities

Most important for investigations described before are experiments which allow for an adequate instrumentation without creating scaling problems. Therefore the (out-of-service-)HDR-reactor, which has been changed to an almost full-scale model of a PWR-reactor, is used as an appropriate experimental facility. Details can be found in the annual report for project PNS 4221.

Furthermore a series of small-scale experiments for investigation of special blowdown phenomena are going to be planned.

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3.2 Research program

The experimental results will be compared with computer results for stress and deformation analysis of reactor vessel internals under blowdown loading. Appropriate computer codes which take account for the effect of coupled fluid and structural dynamics are under development. For analysis of the fluid dynamics within the annular space between reactor vessel and core barrel the finite difference technique will be used. Basically these codes will be two-dimensional, but having special features to allow for dynamic variations of the thickness within the third dimension. The so-called singularity method is under discussion as an alternative method especially for the fluid region within the core barrel. For analysis of dynamic stresses and deformations of the core barrel care must be taken to obtain reasonable computer times which allow several time steps with iterations in the case of fluid structural coupling. Therefore, a method will be used which is based on analytical solutions of a shell with dynamic loading.

4. Project status - Progress to date - Essential results

The YAQUI program developed in Los Alamos (LASL) for calculation of two-dimensional transient flows was used as a basis for the fluid part. It was adapted to PL/1 and made available as a REGENT subsystem. Trial computations had been successfull. Also special features for non-rectangular boundaries and variable thickness of the flow channels are built in. In addition, top down design was started for another fluid dynamic code FLUSTO1 which is more flexible than YAQUI for complex geometries. Also progress was made on the method of singularity which is in discussion for analysis of the fluid region within the core barrel. A shell model (static, linear-elastic, without bending resistance) was formulated and verified. It was found that bending resistance must be taken into account. Therefore, another shell model was formulated which included bending resistance and inertia effects. Comparisions between this model and STRUDL/DYNAL computations yielded agreement with respect to natural frequencies and modes of vibrations. The development of a computer program on this basis is under way.

5. Next steps

The most difficult problem is the coupling of the models for fluid and structural dynamics. Depending on the effects of coupling upon stability significant modifications of the particular codes may be necessary.

6. Relations with other projects

This research program is based on the RS16 program, where blowdowns of a vessel with a relatively small diameter are investigated. Furthermore, as mentioned before, parallel to this theoretical program, full scale blowdown experiments will run on the HDR-reactor under project PNS 4221.

7. Reference documents

/¯1_7 R. Krieg: Ein halbanalytisches Verfahren zur Spannungs- und Verformungsanalyse für eine linear elastische Kreiszylinder-Schale unter nicht-rotationssymmetrischer (spiegelsymmetrischer) dynamischer Belastung normal zur Schalenfläche.

8. Degree of availability

Restricted distribution.

Classification: 7.1	
Title 1 (Original Language): Erstellung eines Auswerteprogramms für instationäre Wärmeübergangsmessungen im RS 50-Containment (RS 233 - I.1.4, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Battelle-Institut e.V., Ffm.
Title 2 (English): Development of Computer Code for the Evaluation of Measurements of Unsteady Heat Transfer in the RS-50 Containment	Project Leader: B. Böhm
Initiated (Date): Oct. 1, 1976 Status: Continuing Completed (Date): March 31, 1977 Last Updating (Date): December 31, 1976	

1. General Aim

For the evaluation of the heat-transfer investigations carried out within the project RS 50, a computer code is to be developed which uses the temperatures measured at different depths of the containment walls to calculate the heat transfer coefficient as a function of time. The evaluation is based on the storage of heat, i.e. the complete temperature profile (including the deeper regions with unchanged temperature) is used. The computer code is aimed at accelerating the evaluation and preventing the increase in the error bandwidth which occurs when the graphic method is applied.

2. Particular Objectives

3. Research Program

The investigations cover the following steps:

3.1. Development of the computer code: data transfer, data smoothing and extrapolation, heat transfer relations, graphic representation.

3.2. Error investigation: the effects of measuring inaccuracy (noise, superposition of noise on the measured signals) and inaccuracy of calculation (smoothing, extrapolation) are investigated.

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3.3. Test computations and comparison of the results with those of other evaluations.

3.4. Final report with detailed description of the computer code.

4. Experimental Facilities, Computer Codes

The computer code to be used for the automatic evaluation of the measured temperature values is divided into two main parts. The first part deals with the smoothing or correction of the input data by means of the following mathematical procedures:

- addition of constants,
- piecewise simple linear regression,
- interpolation by means of cubic splines.

The smoothed curves can subsequently be evaluated with the aid of line printer plots and stored on disk or tape. In the second part of the computer code the heat transfer coefficient is determined as a function of time. This calls for differentiation of the temperature values, which is effected again by means of cubic splines. Square and linear extrapolations finally supply the necessary estimates of the surface temperatures and heat flow densities; the intermediate results can also be shown in graphic form. The final result is the graphic and tabular presentation of the heat transfer coefficients for the two extrapolation methods as a function of time.

The computer code is written in FORTRAN; it is being tested on the digital computer B6700 of Battelle-Institut.

5. Progress to Date

- Establishment of a data connection between the RS-50 process computer and the B6700 of the Battelle computer center.

- Development of the complete computer code and performance of the first tests with given test data.

6. Results

7. Next Steps

The work to be performed in the next report period will concentrate on

- tests of the computer code developed,
- evaluation of some selected experimental results and their comparison with graphically evaluated results,
- improvement of the computer code, and
- preparation of the final report.

8. Relation with Other Projects

The computer code developed for evaluating the RS-50 experiments is to be used likewise for the containment experiments under project RS 123.

9. References

The evaluation method has been described already in the report on research project RS 50 (Battelle report No. BF-RS 50-62-4).

10. Degree of Availability of the Reports

The above-cited report is available upon request from

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

Classification: 7.1	
Title 1 (Original Language): Auswirkungen des Ausströmens von Dampf-Wasser-Gemischen aus Rohrleitungslecks (RS 93 A - I.1.4, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Influence of water-vapour mixtures streaming from pipe leaks	Project Leader: E. Weber
Initiated (Date): 1. 2. 76 — Completed (Date): 31. 7. 77 Status: Continuing — Last Updating (Date): 31. 12. 76	

1. General Aim

The theoretical models for the calculation of the critical flow rates have to be checked for the influence of subcooling and instationary start-up procedures after streaming from pipe leaks (pipe diameter 800 mm).

2. Particular Objectives

When the calculation models have been confirmed, the repulsion and bound forces of two-phase flows can be determined.

3. Research Program

3.1 Experimental investigations on the influence of subcooling on the quasistationary flow

3.2 Theoretical study on the difference between dynamic start-up phase and quasistationary flow with respect to flow rates and forces

3.3 Theoretical study on the transfer of results from small test-stands on reactor dimensions.

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4. Experimental Facilities

The experimental tests will be carried out with an existing test stand of 10 mm pipes. Some accompanying tests will be run on a big stand, in order to get informations on the influence of layer streaming cross sections.

5. Progress to Date

A literature study was started for the two theoretical studies.

For comparison some calculations have been carried out with respect to the difference between dynamic start-up phase and quasistationary streaming from a leak. The results were compared with data from the literature.

The study on the transfer of test results on reactor dimensions was completed.

The teststand (10 mm diameter) was prepared for the experimental investigations. The quick-closing valve, the pressurizer and the instrumentation have been checked.

6. Results

The stream rates depend on the geometrical details of the opening and the size of the pressurizer.

The transfer of the results on reactor dimensions is only possible, when thermodynamic equilibrium is reached in the two-phase flow. When the pressurizers are similar and the flow conditions on the end are comparable, test results for diameters $>$ 50 mm are transferable. The calculation method of Moody is successful.

The theoretical methods, which assume thermodynamic equilibrium at homogeneous phase distribution, have shown good agreement with experimental results from the literature.

Tests with a pipe shorter than 300 mm are not suited for this model, because thermodynamic nonequilibrium occurs.

In the primary circuits of a PWR the thermodynamic equilibrium exists in most cases. As this model is independent from the pipe diameter, the results are transferable on reactor dimensions.

7. Next Steps

Completing of the two theoretical studies
Commissioning of the teststand.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Classification: 7.1	
Title 1 (Original Language): Meßsystem zur Analyse des Dampf-Wasser-Luft-Gemisches in Containment-Überströmöffnungen (RS 195 - I.1.4., Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Battelle-Institut e.V., Ffm.
Title 2 (English): Measurement System to Analyze the Steam-Water-Air Mixture in Containment Overflow Openings	Project Leader: W. Zirnig
Initiated (Date): February 15, 1976 Status: Continuing Completed (Date): May 31, 1977 Last Updating (Date): December 31, 1976	

1. General Aim

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

2. Particular Objectives

The measurement 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 measurements 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 ray absorption densitometer.

3.2. Selection and calibration of the densitometer and error analysis for the whole measurement 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 measurement 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 order 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: Two housing variants were designed and evaluated which are to reduce the ambient temperature of beta detector and preamplifier by water and air cooling to 40 °C. Appropriate measuring windows are provided at the front side so that the beta radiation to be measured is attenuated only insignificantly.

Ad 3.2: A beta ray densitometer was selected and supplied. The delivery of the instrument was substantially delayed.

Ad 3.3: For the measurement of pressure, a piezoresistive transducer was selected and evaluated under laboratory conditions. The measurement of temperature will be carried out by means of fast-response sheathed thermocouples.

Ad 3.4: The experimental facility described in Section 4 was constructed, put into operation, and utilized for the first experiments to evaluate the protective housings and measuring instruments.

6. Results

Ad 3.1: The two housing variants differ essentially in the design of the measuring window. One variant is an open housing in which the beta detector is protected from exposure to steam by air flowing in the opposite direction, in the other variant the front side is covered by a thin steel foil. According to the results of the first experiments, both housings can be designed such that they fulfill their cooling function. The covered window is to be preferred because it does not exert any influence on the steam-water-air stream to be measured.

Ad 3.2: -

Ad 3.3: The laboratory investigations confirm the manufacturer's statement that the piezoresistive pressure transducer enables very accurate static and dynamic pressure measurements to be made at temperatures of up to 175 °C. High measuring accuracy is maintained under both thermal shock and thermal long-time loading.

Ad 3.4: In the first experiments 150 °C and 5 bars were achieved in the low-pressure vessel of the experimental facility.

7. Next Steps

Ad 3.1: Further experiments relating to the design of the housing.

Ad 3.2: Upon delivery of the complete measuring chain, approximately early in February 1977, start-up and calibration.

Ad 3.3: The preliminary investigations into the selection of pressure and temperature transducers will be completed by tests in the experimental facility.

Ad 3.4: Evaluation of the whole measurement system is now scheduled for the end of March 1977.

8. Relation with Other Projects

The complete measurement system is planned to be used in LOCA experiments under project RS 50.

9. References

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10. Degree of Availability of the Reports

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Classification: 7.1	
Title 1 (Original Language): Untersuchung und Entwicklung von Systemen zur Begrenzung der Wasserstoffkonzentration im Sicherheitsbehälter von Siedewasserreaktoren. (RS 223 I.1.4, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Frankfurt
Title 2 (English): Investigation and development of systems limitating the H_2-concentration in the BWR containment.	**Project Leader:** H. Queiser
Initiated (Date): 1. 7.1976 **Status:** Continuing **Completed (Date):** 30. 4.1977 **Last Updating (Date):** 30. 6.1976	

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-distribution:
Theoretical investigation and in case evalution of measurement results about the H_2-distribution in the atmosphere and its influence by convection, diffussion effects and steam condensation.

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 regenration-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 containments is made prior to entering the reaction container by means of a heated tube and/or injection device.

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

5. Progress to Date

Tests have been done in order to investigate the suitable of the catalyst under PWR conditions (the catalyst contaminated with boric acid). In addition, the effect of methyl iodine on the activity of the catalyst was evaluated.

Both tests series have been completed.

Based on the currently available experimental test results, a specification and a flow diagram for the hydrogen concentration control system have been conceived.

6. Results

Experiences gained from pre-tests have been utilized in the selection of process temperatures, catalyst evading and catalyst deactivation characteristics that are relevant to operational techniques. No significant reduction of the recombination effect was observed despite of partial contaminant deposits on the catalyst.

Further the tests conducted at a lower temperature level show a distinct dependency of the recombination rate on the processing temperature.

7. Next Steps

Test series currently under preparation are expected to verify the effects caused by oil mists on the catalyst activity. Subsequently it is planned to apply all possible contaminants to the catalyst in order to obtain reference experiences on resistance to aging (catalyst operating life) and the possibilities of catalyst regeration.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Classification 7.1. 7.2

Title 1	COUNTRY Denmark SPONSOR Risø ORGANIZATION Risø
Title 2 MACON. A Containment Multiroom Transient Analysis Code.	Project leader: Vagn S. Pejtersen
Initiated: 1975 Completed: Status: Progressing Last updating:	Scientists: Margit 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 and 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 nonequilibrium 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.

The blowdown mass flow rate and enthalpy may be calculated by the program or given as input.

The program can model from one to five rooms.

3. Experimental facilities and programme

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

4. Project status

A one-room version has been tested. A sparse technique for solution of the equation system has been used with a speeding up effect on the calculation time. The multiroom code is now been tested.

5. Next steps

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

Calculations with available data (see point 3).

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 1	COUNTRY Denmark
	SPONSOR DAEC Risø
	ORGANIZATION DAEC Risø
Title 2 CONTAC-II. A containment transient analysis code	Project leader: Aksel Olsen
Initiated: January 1969 Completed: Feb. 70 Status: In use Last updating: March 1971	Scientists: N. Kjær-Pedersen 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.

3. 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 calculation has been undertaken.

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, is expected to be ready in early 1975.

6. Relation with other projects

A multiroom containment code, CONTAC-M, is now under development.

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 Intergration
Risø, RD-Memo nr. 46 (1972)

8. Degree of availability

Available

Classification 7.1, 7.2

Title 1	COUNTRY Denmark SPONSOR Risø ORGANIZATION Risø
Title 2 CONTAC-III. A containment transient analysis code	Project leader: V.S. Pejtersen
Initiated: Completed: Mar. 75 Status: Testing phase Last updating: March 1975	Scientists: V.S. Pejtersen F. Cortzen K. Ladekarl 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 67oo 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 and to walls are incorporated(as one-dimensional heatconducting discretized structures, two for each node). A revised version of the vent-flow correlation is incorporated.

3. Experimental facilities and programme

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

4. Project status

5. Next steps

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

6. Relation with other projets

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 Intergration
Risø, RD-Memo nr. 46 (1972)

8. Degree of availability

Available.

Classification 7.1, 7.2

Title 1	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 CONTAC-M: A Containment Multiroom Transient Analysis Code	Project leader: Aksel Olsen
Initiated: 1974 Completed: Status: Progressing Last updating:	Scientists: K.L. Thomsen P. Hansen

1. General aim

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

2. Particular objectives

The code, CONTAC-M, is under development in FORTRAN IV for the Burroughs 6700 computer.
It is to calculate transient pressures and temperatures in a multiroom full pressure or pressure suppression containment following a loss-of-coolant accident. The blow down mass flow rate and enthalpy may be calculated by the code or be supplied as input. Heat transfer to or from structures and walls is to be included as well as the effect from core spray, drywell spray, wetwell spray and wetwell pool cooling. Some effects from non-thermal equilibrium is to be included.

3. Experimental facilities and programme

The code is to be tested against available experimental data, as for instance the Marviken data.

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4. <u>Project status</u>

The formulation of some physical equations has not yet been decided, while various possible mathematical solution methods are still being considered.

5. <u>Next steps</u>

Further work along the lines indicated.

6. <u>Relation with other projects</u>

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

7. <u>Reference documents</u>

No reports available yet.

8. <u>Degree of availability</u>

	Classification 7.1/10.4/11.2/11.4
Title 1 BEAMDYN - Program til lineær 3-dimensionel statisk og dynamisk analyse af bjælkestrukturer.	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2 BEAMDYN - Program for the linear 3-dimensional static and dynamic analysis of beam type structures.	Project leader Per Lundsager Scientists:
Initiated (date) November 1975 — Completed: (date) 1976 Status: progressing — Last updating (date) February 1976	Per Lundsager

1. General aim

To develope a computer program for linear 3-dimensional static and dynamic analysis of beam type structures.

2. Particular objectives

2.1. To make simplified analysis of Nuclear Structures, especially for aircraft impact and earthquake loadings.

2.2. To investigate the properties of various FEM techniques e.g. time integration methods, eigenvalue extraction methods etc.

3. Experimental facilities and programme

None

4. Project status

4.1. A program layout has been made which allows for easy change or extension of essential program features. This is done by extensive use of subprograms.

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4.2. The static and direct time integration parts have been programmed and are being tested. The modal analysis part is being programmed.

4.3. At this stage standard IBM/SSP routines are used where possible. Newmark β time integration is used with $\beta = 0.25$ in order to obtain unconditional stability.

4.4. The program is coded for an IBM 370/165 computer.

5. Next steps

Completion and test of the program in its initial configuration. Further steps have not been decided yet.

6. Relations with other projects

No specific relations for the time being, but the project is of general relevance to Risø's activities in structural analysis.

7. Reference documents

None until now.

8. Degree of availability

Project information will be freely available.

Titre	Pays :
Conséquences d'un LOCA sur l'enceinte : Etude de l'écoulement du brouillard entre les casemates d'une enceinte de confinement : programme REBECA.	FRANCE
	Organisme directeur : CEA
Titre (anglais) Mist flow between subcompartments of a containment : REBECA project.	Organisme exécuteur : CEA/DRE
	Responsable : M.GINIER
Date de démarrage : 01/01/75 Date prévue d'achèvement : 31/12/81 Etat actuel : En cours Dernière mise à jour : 21/01/77	Scientifiques :

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.

Objectifs particuliers :

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

Installations expérimentales et programme :

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

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

Etat de l'étude :

1) Avancement à ce jour :

Etude des composants de l'installation : étude du mélangeur au moyen de l'expérience TUYERE.

Prochaines étapes :

Construction de l'installation

Titre	Pays :
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.	FRANCE
	Organisme directeur : CEA-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 : M. COURTAUD (STT)
Date de démarrage : 01/01/76 Date prévue d'achèvement :31/12/78 Etat actuel : en cours Dernière mise à jour : 21/01/77	Scientifiques :

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.

Objectifs particuliers :

Développement de modèles physiques pour l'interprétation des expériences.

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.

Etat de l'étude :

1) Avancement à ce jour :

Construction de l'installation.

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	CLASSIFICATION 7.1	
TITLE 1	CALCUL DE LA PRESSION DANS L'ENCEINTE DE CONFINEMENT LORS D'UNE RUPTURE DE TUYAUTERIE PRIMAIRE (PWR).	COUNTRY FRANCE SPONSOR E.D.F. SEPTEN ORGANIZATION E.D.F.
TITLE 2	CALCULATION OF THE PRESSURE IN THE CONTAINMENT, FOLLOWING A LOSS OF COOLANT ACCIDENT (PWR).	Project Leader E.D.F./SEPTEN/T
Initiated 1973	Completed 1974	Scientists 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 recirculation

CATEM 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.

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IV - PROJECT STATUS

4.1 - Progress to date

Codes élaborés.

4.2 - Essential Results

Calcul des enceintes des tranches PWR françaises.

V - NEXT STEPS

Améliorations éventuelles des corrélations utilisées dans les codes à l'aide de résultats expérimentaux.

VI - RELATION WITH OTHER PROJECTS

VII - REFERENCE DOCUMENTS

- note E.D.F./SEPTEN TC 74.46 "Equations de base et utilisation du code CATEM 6"
- note E.D.F./SEPTEN TC 74.48 "PAREO 5 : modèle mathématique pour l'étude de l'évolution des pressions et températures dans une enceinte de réacteurs PWR après PDR".

VIII - DEGREE OF AVAILABILITY

- rapports § 7 : disponibles.

La communication des codes ou leur utilisation pour les études appliquées doivent faire l'objet de contrats cas par cas.

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

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

PROJECT TITLE : Development of a code for transient analysis	CLASSIFICATION 7.1
SPONSORING COUNTRY : ITALY	ORGANISATION : NIRA-Cirene
DATE INITIATED : 1974 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: NETHERLANDS. SPONSOR: ECN ORGANIZATION: ECN
TITLE: (ENGLISH LANGUAGE): CHARME-IM Blowdown jet impingement process	PROJECTLEADER: Speelman, J.E.
INITIATED: 1976 LAST UPDATING: April 1977	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 : In discussion

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: -

ENERGIEONDERZOEK CENTRUM NEDERLAND		CLASSIFICATION: 7.1
TITLE: CHARME-DIS Berekening van het thermohydraulische proces in de uitstroomleiding van veiligheidskleppen.		COUNTRY: NETHERLANDS. SPONSOR: ECN ORGANIZATION: ECN
TITLE: (ENGLISH LANGUAGE): CHARME-DIS Determination of thermohydraulic process in safety relief discharges pipes		PROJECTLEADER: Speelman, J.E.
INITIATED: June 1976	LAST UPDATING: April 1977	SCIENTISTS: Putten, A.P.W.M. van der Bogaard, J.P.A. van den 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 preliminary calculational method has been developed.

Next steps

Development of special subroutines to calculate shockwaves.

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

Reference documents: -

Degree of availability: Not yet applicable

Budget: -

Personnel: -

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

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TNO - IWECO	CLASSIFICATION: $\boxed{3.2}$/ 3.3 7.1.
TITLE: Responsieberekeningen voor reactorgebouw	COUNTRY: NETHERLANDS SPONSOR: Ministry of Social Affairs ORGANIZATION: TNO-IWECO
TITLE (ENGLISH LANGUAGE): Dynamic Response of Reactor structures (building and containment)	PROJECTLEADER: Meyers
INITIATED: June 1974 COMPLETED: end 1976 STATUS: progressing LAST UPDATING: February 1976	SCIENTISTS: Van Beek Geertsema

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TNO - IWECO	CLASSIFICATION: 3.2/3.3/7.1
TITLE: Responsieberekeningen voor reactorgebouw	COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs ORGANIZATION: TNO-IWECO
TITLE: (ENGLISH LANGUAGE): Dynamic response of reactor structures (building and containment)	PROJECTLEADER: Meijers
INITIATED: June 1974 / LAST UPDATING: April 1977	SCIENTISTS: Van Beek Geertsema
STATUS: Progressing / COMPLETED: 1978	

	Classification 7.1
Title 1 EXPERIMENTAL WORK TO EXAMINE THE EFFECT OF MISSILE IMPACT ON REACTOR CONTAINMENT STRUCTURES	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION FOULNESS
Title 2	Project Leader T P O'BRIEN
Initiated 1976 Completed : 1978 Status : Last updating	Scientists:

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.

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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.

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Classification

7.1

Title 1	COUNTRY UNITED KINGDOM
INTERNAL MISSILE EFFECTS	SPONSOR UKAEA
	ORGANIZATION SRD CULCHETH
Title 2	Project Leader 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.

Classification: 7.2		
Title 1 (Original Language): Versuche zum Wärmeübergang bei Eiskondensation (RS 67 - I. 1.4, Jahresbericht A 76)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Technische Universität München
Title 2 (English): Experiments for Heat Transfer of Ice-Condensation		Project Leader: Prof. Dr.-Ing. U. Grigull
Initiated (Date): 1.03.1973	Completed (Date): 28.02.1977	
Status: Continuing	Last Updating (Date): 31.12.1976	

1. General aim

A leak in the primary loop of a reactor results in a pressure increase in the reactor containment due to outflow of water-steam mixture. If the steam can be condensed e.g. on an ice surface the pressure increase is nearly avoided and therefore the danger of contamination minimized.

2. Particulat objectives

This research program will investigate the heat- and mass transfer at the condensation of steam on a vertical ice-surface.

3. Research Program

Experiments will be made with steam flowing upwards and downwards with varying velocities to establish analytic means for calculating ice-condensors.

4. Experimental Facilities, Computer Codes

See Annual Report IRS - F - 24. General computer codes are not used.

5. Progress to Date

The experiments with steam flowing upwards have been completed. Tests with steam flowing slowly downwards are currently under way.

6. Results

If the steam flows slowly upwards (20 m/s) a slug of water drops swings up and down. The melting rate is 1 to 2 mm/s.

If the steam flows very slowly upwards (< 10 m/s) the water film flows downwards nearly undisturbed by the steam. The melting rate decreases to 0,3 mm/s.

The same melting rate was measured if the steam flows slowly downwards (20 m/s), the water film is accelerated by the steam. The melting rate inceases to 1.3 mm/s.

Currently test-runs are made with steam flowing downwards with moderate velocity.

7. Next Steps

Experiments with steam flowing downwards at a high velocity.

8. Relation with Other Projects

See Annual Report IRS - F - 24.

9. References

- Quarterly Reports (German) in the series "IRS - Forschungsberichte" Oct. 1975 - Sept. 1976: IRS - F - 30, 31, 33.
- Annual Report (English) in the series "IRS - Forschungsberichte" Jan. 1975 - Dec. 1975: IRS - F - 29.

10. Degree of availability of the Reports

The reports in the series "IRS - Forschungsberichte" are obtainable from the Gesellschaft für Reaktorsicherheit, Köln.

Classification: 7.2	
Title 1 (Original Language): Theoretische Arbeiten zum Druckabbausystem (Kondensation IV, Teil 1) (RS 78 B - I.1.4, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Theoretical Investigations of the Pressure Suppression System (Condensation IV, Part 1)	Project Leader: Dr. Sobottka
Initiated (Date): 15. 10. 74 Status: Completed	Completed (Date): 31. 3. 76 Last Updating (Date): 31. 12. 76

1. General Aim and 2. Particular Objectives

Development and Evaluation of theoretical models to describe special events in the pressure suppression system (symmetric and unsymmetric pool swell and pressure oscillations) during a loss of coolant accident. Special data evaluation for the problems mentioned by using existing experiments.

3. Research Program

3.1 Investigation of water pool swell

3.2 Pressure oscillations

4. Experimental Facilities

No experimental facilities necessary.

5. Progress to Date

Pool Swell:

The existing model was improved by the effect of the air/vapour mixture in dry well before clearing the vent pipes and a more realistic pressure transient in wet well (air compression during pool swell).

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Pressure Oscillations:

The model KSWING IV was improved by a model, which describes the collapse of a vapour bubble at the vent pipe outlet, the entering of the water into the vent pipe and the forming of a new bubble. The whole process is called chugging.

6. Results

Pool Swell

The calculated pool swell height and dynamic air bubble oscillation are in good agreement with experimental results. Furthermore an amplification factor for the average pool swell which describes the maximum local water height was evaluated from experiments data.

Pressure Oscillations

Calculations for the Marviken-I-test No. 4 with chugging effects gave a frequency of 3,7 Hz which agreed with the measured value. The calculated pressure amplitude was $\pm$ 0,2 bar, the measurend value at the bottom of the wet well was $\pm$ 0,08 bar (must be lower because of the geometrical distance!). The water height in the vent pipe was theoretically expected to be 0,1 m; the experiment showed that the thermocouple 0,3 m above vent pipe outlet was not reached by pool water. This result shows, that the theoretical model describes the condensation process sufficient well.

Furthermore the dynamic pressure of Marviken 13 test which is supposed to be a blow down with pure steam condensation was calculated in good agreement with the experiment.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

RS 78 D   Kondensation V Part 1

RS 78 E   Kondensation V Part 2

9. References

Dres. Sobottka, Antony-Spies
"Kondensations- und Freiblaseversuche (Kond. IV, Teil 1)"
Abschlußbericht Förderungsvorhaben RS 78 B
Kraftwerk Union Aktiengesellschaft, Erlangen (Nov. 1976)

10. Degree of Availability

The report is classified Company Confidential.

Classification: 7.2		
Title 1 (Original Language): Dynamische Beanspruchung von LWR-Druckabbausystemen (PNS 4211 - I.1.4., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Dynamic Load of LWR Pressure Suppression Systems		Project Leader: R.A. Müller
Initiated (Date): January 1972 Status: continuing	Completed (Date): December 1976 Last Updating (Date): December 1975	

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 characteristics and intensity of which is not yet sufficiently known in all cases. The aim consists in enlarging knowledge of the extended steady-state condensation process so that the dynamic loads applied to the components can be determined. The implementation of this task is governed by the availability of suitable test facilities.

Particular Objectives

Evaluation of the intensity and characteristics of pressure pulsations taking place during condensation in the water pool (essential parameters: mass flow density, pipe diameter, number of pipes, water temperature, pressure level and air content of the steam).

Influence of the drywell on the condensation process.

Stresses and dynamic behavior of containment structures under the influence of condensation loads.

Project Status

Progress to Date

In the containment experiments performed in the Marviken Nuclear Power Station in 1972 and 1973 higher pressure oscillations were observed to occur in the pressure suppression system, above all in tests Nos. 10 and 13, which seemed to indicate possible system resonances. To explore these problems, the natural frequencies of the Marviken plant were calculated. A model was used for the calculations, which takes into account coupling of the three most important single systems,namely blowdown channel, header and vent pipe. The results are strongly dependet on the velocity of sound which in turn is determined mainly by the steam moisture.

For realistic values of the velocity of sound the first three natural frequencies calculated by this model come close to the measured values to be derived from the test data. Since also the excitating frequency from the water pool, which is caused by condensation, lies within the range of the first and second natural frequencies, conditions could actually have been passed during blowdown under which the steam and water volume fraction, respectively, in the drywell has generated such a sound velocity and hence natural frequency close to the frequency range of excitation that resonance took place. This would explain the greater pressure amplitudes observed.

The development work in the laboratory scale of the infrared measurement technique was terminated and an offer submitted to the Marviken II Project. After the order for one measuring position had been received, the measuring equipment was conceived, designed and manufactured. The supporting frames and the supply lines were installed in Marviken while the optical source and receiver components as well as the electric and electronic supply units were pressembled and tested in the workshop. This development work was carried out together with Institut für Thermische Strömungsmaschinenen (ITS) of Karlsruhe University.

As a supplement to the tests with the pressure relief nozzles performed in the Brunsbüttel Nuclear Power Station in October 1974 further tests were made in June 1975 with a vent pipe of 600 mm Ø. The measured values recorded by IRE in the two test series were evaluated. In all tests standing pressure waves having low amplitudes were generated in the annular water pool in addition to the local events taking place.

A transient calculation was performed in the studies of the dynamic behavior of the containment structure. For this purpose, a pressure field in terms of time and space was interpolated from one test of the first Brunsbüttel test series applied as a load to the model of calculation. The displacement with respect to time of interesting points of the containment was determined. The actual displacements were assayed by double integration of the measured accelerations and thus comparable with the calculated results. The agreement is not yet satisfactory.

Within a new series of condensation tests involving single and multipipe arrangements and carried out by KWU in Karlstein (previously Großwelzheim) measurements were performed in a total of 11 tests. These measurements related to the pressure development with respect to time at 15 points of measurement arranged on three coordinates at right angles to each other. The origin of the coordinates was the expected bubble center. The evaluation first preferably concerned the tests having a mass flow density of 16 kg/m^2 sec since this value had been used to investigate all the configurations (pipe diameter, number of pipes).

Moreover, first parameter calculations were started within the period of reporting, using a computer program which describes the thermohydraulic events of the condensation process. In this program vent pipes operating in parallel are treated and significant coupling mechanisms on the steam and water sides are taken into account.

Essential Results

The computations are well as the observations made during the Marviken I tests support the assumption that events resembling resonances might taken place in pressure suppression systems under specific conditions of geometry and operation, which might lead to pressure pulsation amplification.

According to the results of laboratory tests use of the infrared absorption measurement technique is promissory for determination of the moisture content of a two-phase flow in blowdown tests and for determination of the velocity of the liquid phase.

In toroidal water pools of pressure suppression systems standing pressure waves having low amplitudes are generated in addition to the local events caused by the individual vent pipes.

The high pressure peaks generated during condensation of air-free steam and superimposed to the usual pressure pulsations with smaller amplitudes have a very short base width of about 0,15 msec. Therefore, their significance with respect to current structural components is generally low.

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Next Steps

Final assembly work, commissioning and use in future tests of the infrared measurement equipment delivered to Marviken.

The calculations will be continued relative to the dynamic behavior of the containment structure. Analysis of further test results is to yield the reasons of the as yet unsatisfactory agreement between the calculated and experimental displacements so that an improvement of the calculation model can possible by achieved.

The parameter calculations concerning the condensation at one and several parallel pipes are being continued, partly by use of improved programs.

Reference documents

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Reaktortagung, Nürnberg, 8. - 11. April 1975,
Deutsches Atomforum e.V., Kerntechnische Ges. im Dt. Atomforum e.V.,
Leopoldshafen 1975

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3. Internat. Conference on Structural Mechanics in Reactor Technology,
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Computation of the Dynamic Behavior of a Nuclear Reactor Containment with TOPOLOGY and STRUDL-DYNAL,
15. Worldwide ICES Users Group Conference, London, Sept. 25-26, 1975

R.A. Müller, E.G. Schlechtendahl,
Dynamische Belastungen und Strukturverhalten von Reaktorsicherheitsbehältern mit Druckabbausystemen,
KFK-Nachrichten, 7 (1975) No. 3, S. 12-21

J. Kadlec, R.A. Müller,
Dynamic Loading of Containment during Blowdown. Review of Experimental Data from Marviken and Brunsbüttel,

B. Göller, B. Laursen, E.G. Schlechtendahl,
Results of Calculation of the Dynamic Behavior of Pressure Suppression System during Blowdown
Extreme Load Conditions and Limit Analysis Procedures for Structural Reactor Safeguards and Containment Structures, Berlin, Sept. 8 - 11, 1975

Degree of Availability

Unrestricted Distribution

Classification: 7.2	
Title 1 (Original Language): Dynamische Beanspruchung von LWR-Druckabbausystemen (PNS 4211 - I.1.4, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: - ORGANIZATION: GFK
Title 2 (English): Dynamic Load of Pressure Suppression Systems	Project Leader: R.A. Müller
Initiated (Date): Jan. 1, 1972 Status: Continuing Completed (Date): Last Updating (Date): Dec. 1976	

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 characteristics and intensity of which are not yet sufficiently known in all cases. The aim consists in enlarging knowledge of the extended steady-state condensation process so that the dynamic loads applied to the components can be determined. The implementation of this task is governed by the availability of suitable test facilities and by the requirements of the licensing procedures under way for boiling water reactors.

2. Particular Objectives

Experimental evaluation as well as analytical and numerical modeling of the intensity and characteristics of pressure pulsations taking place during condensation in the water pool (essential parameters: mass flow density, water temperature, pressure level, air content of the steam, pipe diameter, number of pipes, drywell volume and shape).

Dynamic behavior and stresses of containment shell structures under the influence of condensation loads.

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3. Research Program

The program includes

- gathering of experimental data on the pressure pulsations occurring during the process of condensation and on dynamic system responses of the vessel structure;
- interpretation and verification by calculations of data measured, using available computer programs or computer programs to be developed;
- computation of real pressure suppression systems and development of the required computer programs.

4. Experimental Facilities, Computer Codes

Measurements have been performed in the following experimental facilities:

- Marviken Nuclear Power Station (Sweden),
- Brunsbüttel Nuclear Power Station,
- Philippsburg Nuclear Power Station,
- Karlstein Large Tank Test Facility,
- GKM Test Facility of Mannheim.

The following computer codes are used:

a) the KONDAS program for the calculation of the process of condensation;

b) the EIGVAL (Time Data Analyzer) program for the calculation of the acoustic resonances of the drywell;

c) some special codes for the calculation of the dynamical shell behavior.

5. Progress to Date

The infrared measurement equipment for the Marviken II Project was installed in the plant and started operation. The instrument was successfully used in all nine blowdown tests. It served to measure the composition of the mixture and the flow rate of the liquid phase in one of the four blowdown channels. The data measured were evaluated and documented in the Marviken reports.

By contrast with Marviken I, the recording of measured data in the Marviken II tests allows to establish a correlation between the values measured in the water pool of the wetwell and in the vent system. An evaluation was made for test no. 19.

Within the framework of a new test series carried out in the conventional power station "Großkraftwerk Mannheim" control measurements were performed on behalf of TÜV Baden (Technical Inspectorate). The test rig used in these tests was so designed that it maps in a representative way a selected section of the real facility (a so-called "single cell"). Besides the temperature, these supplementing measurements related to the pressure prevailing at the walls of the wetwell and in the drywell with a high time resolution. The data measured were evaluated and the results transmitted to TÜV Baden. No noticeable deviations were found between the values measured by the two participating groups.

Also on behalf and for the account of TÜV Baden, Mannheim, further control measurements were performed in the Philippsburg Nuclear Power Station during the non-nuclear hot tests with the pressure relief nozzles. The wall loads and the wall reactions of the reactor safety containment and the wetwell, respectively, were measured at selected points in 86 tests in total. The data measured and processed were directly transmitted to TÜV Baden.

As a whole, quite comprehensive experimental data sets are now available on the intensity and characteristic of pressure pulsations generated in pressure suppression systems (MARVIKEN I and II multipipe experiments; single and multipipe experiments in the Karlstein Large Tank Facility; Größkraftwerk Mannheim experiments; tests with the relief nozzles in the Brunsbüttel and Philippsburg Nuclear Power Station, test with one vent pipe, 600 mm Ø , in the Brunsbüttel Nuclear Power Station).

The parameter calculations of the condensation process were continued. This work was performed on behalf and for the account of TÜV Baden using the KONDAS computer program developed for this purpose. This code considers the thermohydraulic effects of the condensation process (including buoyancy, inertia, surface roughness and the detachment of steam bubbles). Furthermore, vent pipes operating in parallel are treated and significant coupling mechanisms on the steam and water side are taken into account.

The computation on the dynamic behavior of the containment structure performed last year using the STRUDL-DYNAL finite element program did not yield a satisfactory agreement with the deflections measured in Brunsbüttel. Therefore, some analytical studies were first made.

For the inner cylinder of the Brunsbüttel wetwell the dynamic behavior was derived from the Donnel's theory of shells. The Flügge's theory of shells was used to calculate the spherical shell. For the cylinder the mass forces in the radial direction were taken into account whilst the mass forces in the radial as well as in the tangential directions were considered for the spherical shell. The deflection was determine for the spherical shell, using modular superposition, for the case of a distributed pressure load.

6. Results

The velocity measurements of the liquid phase using the correlation technique, which were performed in Marviken with the help of the infra-red absorption technique yielded but little differences with respect to the vapor velocity measured by the radiotracer technique for the flow zone covered by the measurement equipment (150 and 300 mm Ø, respectively). Consequently, there is only little slip between the liquid and the vapor phases at the location of measurement. Also the measured moisture content can be considered to be very reliable. By contrast, the total liquid mass flow determined by conversion to the full channel cross section (1.2 m Ø) deviates heavily from the integral mass flow determined by other methods, with the highest deviation found for the small measurement cross section of 150 mm Ø. A moisture content differing in the radial directions as well as the liquid film on the channel walls not recorded by the infra-red measurement technique can be considered as the reasons.

The previous evaluation of the Marviken test 19 showed that the pressure variations measured in two of the four parallel blowdown channels are equal in phase; the same holds for the two instrumented vent pipes connected in parallel. Also the comparison of pressure signals measured in the water pool with that measured in the vent system revealed the strong correlation of these values.

The tests performed with the pressure relief nozzles at the Philippsburg Nuclear Power Stations essentially confirmed the data measured in similar tests performed at the Brunsbüttel Nuclear Power Station in 1974.

Using the KONDAS computer program it was possible to compute in a reliable manner both the low frequency events of pulsating condensation, i.e. the formation of vapor bubbles, and bubble collapsing and to make predictions about the wall loads occurring in a multiple-tube assembly. Comparison with the experimental data, on the whole, yielded satisfactory agreements. It also appeared from the computations that for the multiple tube assembly coupling on the vapor side via the common drywell of all vent pipes leads to a global synchronization and general similarity of the condensation processes taking place on the individual tubes. By contrast, coupling on the water side prevents the collapsing processes to take place at the same time at the individual nozzles. Therefore, the pressure phenomena loading the wall structure of the pressure suppression system occur in the facility in the form of "time windows" of some hundred milliseconds duration, whilst the period of repetition of the globally synchronous bubble formation event lasts about 1 to 2 seconds.

The analytical investigations of the dynamic behavior of the containment structure showed that a variety of vibrational modes (some hundreds) up to high orders have to be taken into account to determine a dynamic response, since the eigenfrequencies are very close to each other. The requirement of considering such high orders must be known also for the load which in experimental load evaluation, would call for a correspondingly high number of measurement positions located closely to each other.

In case of the spherical shell it appeared that noticeable radial deflections occur only in the loaded range: however, this is not in line with former finite element computations although it is supported by the experimental findings.

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7. <u>Next Steps</u>

The parameter calculations concerning the condensation at one and several parallel pipes are being continued. The behavior of the spherical shell at transient pressure load and the stresses in the shell will be calculated in detail. Also attempts will be made to couple the water moved along with the structure.

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

A cooperation and an exchange of experience has been established with the research projects supported by the government and pursued simultaneously on the same topic by the manufacturer of the German boiling water reactor facilities.

9. <u>References</u>

1st Semiannular Progress Report PNS 1/1976

2nd Semiannular Progress Report PNS 2/1976

Appelt, K.D.; Cramer, M.; Eberle,F.; Kadlec, J.; Klima, P.; Lang, G.
Untersuchung der Druckpulsationen im Druckabbausystem und der dynamischen Reaktion des Sicherheitsbehälters im Rahmen der Entlastungsventilversuche im Kernkraftwerk BRUNSBÜTTEL
KFK-Ext. 08/75-05 (January 76)

Barschdorff, D.;, Wolf, E.
Infrarotmeßtechnik für Zweiphasen-Mehrkomponenten-Massenströme.
KTG-Fachtagung "Experimentiertechnik auf dem Gebiet der Reaktor-Fluiddynamik", Techn. Univ. Berlin, March 10-12, 1976

Schnauder, H.; Cramer, M.
Kondensationsversuche in der KWU-Großbehälteranlage in Karlstein
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Kadlec, J.; Müller R.A.
Dynamic loading of containment during blowdown: review of experimental data from Marviken and Brunsbüttel.
Nuclear Engineering and Design 38 (1976) p. 143-58

Class, G.; Kadlec, I.
Survey of the behavior of BWR pressure suppression systems during the condensation phase of LOCA
Joint Meeting of the American Nuclear Society, the Atomic Industrial Forum and the European Nuclear Society,
Washington, D.C., November 15-19 (1976)

10. Degree of Availability

Unrestricted distribution.

Classification: 7.2	
Title 1 (Original Language): Kondensation IV, Teil 2 (RS 78 C - I.1.4 - Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Karlstein
Title 2 (english): Condensation IV, Part 2	Project Leader: Dr. Simon
Initiated (Date): 1. 8. 75 Status: Completed	Completed (Date): 31. 3. 76 Last Updating (Date): 31. 12. 76

1. General Aim and 2. Particular Objectives

The pressure peaks and the loads on the structures in the water of the condensation chamber of the BWR-pressure suppression system were investigated on the basic of test results. Particularly work was concentrated on harmonic pressure oszillations and the forces in the struts of the condensation chamber walls.

3. Research Program

3.1 Evaluation of the model tank tests

3.2 Evaluation of large tank tests

4. Experimental Facilities

Use of Company proprietary facilities (model tank, large tank).

5. Progress to Date

The model tank tests with vents of 80 and 160 mm were evaluated. The evaluation of the results of the large tank tests with vents of 600 and 300 mm diameter were continued.

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6. Results

Model Tank Tests

With multi-vent arrangements a reduction of the amplitudes of the pressure peaks was reached. The harmonic amplitudes remain constant when the vent/pool area ratio is held constant.

Several mitigation devices were tested. With these devices the condensation was more homogeneous and the pressure peaks were reduced. The best results showed geometries which distributed the vent cross section on several smaller cross sections. Three types had good chances:

- the slit vent with closed bottom
- the perforated vent with closed bottom
- the step vent bundle.

Large Tank Tests

The evaluation of the strut loads of the vent tests with 300 mm diameter was completed. Maximum values were 75 kN (for temperatures of the water > 60 °C). The maximum strut load was independent from the number of vents.

The statistical mean value of the strut load was 11,1 kN. The maximum of the forces on the struts was in the direction of half the angle between two adjacent struts.

The statistic evaluation was completed for the pressure oszillations. The maximum amplitudes were in the mass flow range between 15 and 20 kg/m^2s. The amplitudes were lower at higher or lower mass flow rates for both vent diameters (300 and 600 mm diameter).

7. Next Steps

The work has been completed.

8. Relation with Other Projects

RS 78 D Condensation V, Part 1

RS 78 E Condensation V, Part 2

9. References

Hoffmann, Werner
"Kondensation von Dampf in einer Wasservorlage:
Untersuchung der bei der Mischkondensation auftretenden Druckschwingungen und der Kräfte in den Streben des Kondensationsrohrverbandes, Teilvorhaben A: Druckbelastungen
Teilvorhaben B: Strebenbelastungen"

Abschlußbericht BMFT RS 78 C, Kraftwerk Union Aktiengesellschaft (Okt. 1976)

10. Degree of Availability

The report is company confidential.

Classification: 7.2	
Title 1 (Original Language): Kondensation V, Teil 1 (RS 78 D - I.1.4, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Karlstein
Title 2 (English): Condensation V, Part 1	**Project Leader:** Dr. Simon
Initiated (Date): 1. 4. 76 **Status:** Continuing **Completed (Date):** 30. 9. 77 **Last updating (Date):** 31. 12. 76	

1. General Aim

The condensation loads were investigated with back pressure, with a well defined reservoir volume, at different containment stiffnesses and 600 mm vent diameter, which correspond to the reality of the 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 volumina 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, wich represents the condensation chamber stiffness

3.3 Visuell observations of the phenomena

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

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4. <u>Test Facilities</u>

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

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

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

The first 20 tests were run with a transient vapour mass flow rate which was adepted to the different simulated accidents. The init l temperature was 30, 45 and 60 °C. The chamber wall was preheated, the input length of the pipes into the water was 2,0 and 2,8 m.

When the wall stiffness had been changed, the tests Nr. 21 - 34 were run. Some tests were varied by the exchange of the air in the pressure chamber volume with vapour. The pressure in the testchamber was lower for these cases.
The test results were evaluated.

6. <u>Results</u>

The test 1 - 20 showed that the submergence of the vent into the water had an influence on the loads. The maximum wall pressure was 1,1 bar, the mean value 0,1 bar. The mean strut forces were lower than 10 kN.
The variation of the back pressure had no significant influence on the results.

With the first wall the stiffness of KKB was verified (test 1 - 20), the second wall corresponded to KKP (test 21 - 34). The frequency analysis of the pressure data showed resonances at 8 and 12 Hz.

7. <u>Next Steps</u>

Analyses shall be continued. Further tests shall implement conditions which are not in conformity with real possible blow-down-conditions.

8. Relation with Other Projects

RS 78 C   Condensation IV, Part 2
RS 78 E   Condensation V, Part 2

9. References

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10. Degree of Availability

Classification: 7.2	
Title 1 (Original Language): Kondensation V, Teil 2 (RS 78 E - I.1.4; Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Karlstein
Title 2 (English): Condensation V, Part 2	Project Leader: Dr. Simon
Initiated (Date): 1. 4. 76 Status: Continuing Completed (Date): 30. 9. 77 Last Updating (Date): 31. 12. 76	

1. General Aim

Attest of the single cell theory and determination of the condensation load at different water rooms.

2. Particular Objectives

The earlier model tank experiments (Condensation IV) were continued, the evaluation of the results was carried out with statistical viewpoints.

3. Research Program

3.1 Build-up of the test stand

3.2 Tests with two troughs (single cells)

3.3 Main test with other troughs up to 8 vent-couples

3.4 Evaluation of the results.

4. Test Facilities

Two model troughs with different size and defined wall stiffness 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

The test stand was completed. The trough with two vents of 80 mm diameter was variable in the reservoir volume and pipe length. The bottom and three sides were made of concrete. The fourth side was movable and made of steel in order to vary the stiffness.

6. Results

The tests proved that the stiffness has a noticeable influence on the condensation behaviour. Extension oscillations on the movable wall and pressure course in the water were synchronized, the frequency of the harmonic oscillations was 9 Hz.

The tests confirmed the previous results. Remarkable were the needle-like pressure peaks of the pressure-gauges on the concrete walls, whereas no high frequent pressure pulses were observed on the steel walls.

7. Next Steps

The results will be analysed. Troughs with more than two up to 8 vents will be built and tested.

8. Relation with Other Projects

RS 78 C	Condensation IV, Part 2
RS 78 D	Condensation V, Part 1

9. References

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10. Degree of Availability

Classification 7.1, 7.2

Title 1	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 CONTAC-II. A containment transient analysis code	Project leader: Aksel Olsen
Initiated: January 1969 Completed: Feb. 70 Status: In use Last updating: March 1971	Scientists: N. Kjær-Pedersen V.S. Pejtersen

Classification $\boxed{7.1}$, 7.2	
Title 1	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 CONTAC-M: A Containment Multiroom Transient Analysis Code	Project leader: Aksel Olsen
Initiated: 1974 Completed: Status: Progressing Last updating:	Scientists: K.L. Thomsen P. Hansen

Classification $\boxed{7.1}$, 7.2	
Title 1	COUNTRY Denmark SPONSOR Risø ORGANIZATION Risø
Title 2 CONTAC-III. A containment transient analysis code	Project leader: V.S. Pejtersen
Initiated: Completed: Mar. 75 Status: Testing phase Last updating: March 1975	Scientists: V.S. Pejtersen F. Cortzen K. Ladekarl Thomsen

Classification $\boxed{7.1}$ 7.2	
Title 1	COUNTRY Denmark SPONSOR Risø ORGANIZATION Risø
Title 2 MACON. A Containment Multiroom Transient Analysis Code.	Project leader: Vagn S. Pejtersen
Initiated: 1975 Completed: Status: Progressing Last updating:	Scientists: Margit B. Andersen

Title 1 (Original language)	Classification
Heat transfer in pressure suppression	7.2
Title 2 (English)	Country ITALY Sponsor / Organisation CNEN
Date initiated 1975 Date completed 1977 Last updating April 1977	Project Leader 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 m^3/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) SOPRE 1 - Ricerca sul comportamento del sistema di contenimento a soppressione di pressione in caso di LOCA.	Classification 7.2
Title 2 (English) SOPRE 1 - Research on the behaviour of pressure suppression containment system after LOCA.	Country ITALY Sponsor CNR - CNEN Organisation University of Pisa
Date initiated 1974 Date completed 1977 (first phase) Last updating January 1977	Project Leader 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 : SOPRE 1 - Research on the behaviour of pressure suppression containment system after LOCA	CLASSIFICATION 7.2
SPONSORING COUNTRY : ITALY	ORGANISATION : UNIVERSITY OF PISA
DATE INITIATED : 1974 (actual phase) 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.

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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.

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

. DESCRIPTION

Research Program
The program has the aim of investigating, using analytical tools, thermo-fluidynamic 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

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

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

1009

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

140-21-71/4111-01

Classification : 7-3

1011

Titre	Pays : FRANCE
Inhibition de la radiolyse des solutions d'arrosage en cas de LOCA.	Organisme directeur : CEA
Titre (anglais) Inhibition of spray solutions radiolysis in a LOCA accident.	Organisme exécuteur : CEA/DRA Responsable : M.ROTH (DRA)
Date de démarrage : 01/01/73 Date prévue d'achèvement :31/12/77 Etat actuel : en cours Dernière mise à jour : 15/11/76	Scientifiques : M.ROZENBERG

Objectif général :

Protection des centrales nucléaires à eau légère contre le risque d'explosion dans l'enceinte de confinement après un accident de base par perte de caloporteur.

Objectifs particuliers :

Suppression ou diminution de l'hydrogène formé après LOCA, par réaction métal-eau, corrosion et surtout radiolyse. Les solutions d'arrosage destinées à retenir principalement les iodes de fission accélérent la formation d'hydrogène radiolytique, rendant rapidement l'atmosphère de confinement inflammable. L'utilisation d'additifs chimiques peut ralentir suffisamment la décomposition radiolytique au point d'écarter tout danger d'inflammabilité dans l'enceinte de confinement. L'étude est consacrée à la recherche de l'additif le plus efficace.

Installations expérimentales et programme :

On dispose d'un irradiateur GAMMA de 10 000 Ci de CS 137. Divers additifs potentiellement efficaces sont testés au point de vue de leur influence :
Sur le rendement radiolytique en phase liquide,
Sur la vitesse de formation en phase gazeuse d'hydrogène dans un milieu diphasique statique,
Sur la vitesse de formation en phase gazeuse d'hydrogène dans un milieu diphasique à recirculation.

.../...

1012

Etat de l'étude :

1) Avancement à ce jour :

L'étude de l'influence des additifs sur le rendement radiolytique en phase liquide et sur le dégagement gazeux en milieu diphasique statique est terminée.

2) Résultats essentiels :

Certains additifs peuvent diminuer jusqu'à 60 % le rendement net radiolytique de formation d'hydrogène en phase liquide. Cette efficacité n'est pas toujours confirmée par les expériences de dégagement gazeux dans un milieu diphasique statique. D'autres additifs ont été trouvés qui retardent ou limitent le dégagement d'hydrogène.

Prochaines étapes :

Il s'agit de confirmer l'effet limitatif ou retardateur des additifs, potentiellement efficaces dans des expériences avec dégagement, dans un circuit avec recirculation d'eau.

Documents de référence :

"Génération d'hydrogène par décomposition radiolytique des solutions d'arrosage capables d'absorber l'iode de fission - influence de certains additifs", O.PAOLI, J.ROZENBERG - Rapport DRA/SAECNI-75-387.

7013

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/TEGAEN Scientists MM. BUREAU BERGE
Dated février 1975 Completed 1975 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.

...

014

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.

1015.

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

1016

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)

1047

NETHERLANDS ENERGY RESEARCH FOUNDATION (ECN)		CLASSIFICATION: 7.4
TITLE: Advies inzake periodieke kontrole van insluitvaten op lekkages		COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs ORGANIZATION: ECN
TITLE: (ENGLISH LANGUAGE): Advice on periodic supervision on leakage tightness of containments		PROJECTLEADER: H.J. van Grol
INITIATED: June 1974	LAST UPDATING: end 1977	SCIENTISTS: J.W.H. van den Bergh H. Pruimboom
STATUS: progressing	COMPLETED: March 1977	

1. 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 inclusing 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 Answers to the Questionnaire on Reactor Containment Leak-tightness" February 24, 1977

5. Next steps

Discussion with authorities and others on basis of the draft reports in order to finish the work.

6. Relation with other projects: -

7. Reference documents: See under 4

8. Degree of availability

The reports will be translated in the English 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

Classification: 8.	
Title 1 (Original Language): Untersuchungen über die Zuverlässigkeit von Druck- und Differenzdruck-Meßumformern unter GaU-Bedingungen (RS 110 - I.1.4, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Investigation of the Behaviour of Pressure and Differential Pressure Transducers under the Influence of an MCA	Project Leader: K. Riemann
Initiated (Date): Oct. 1973 / Completed (Date): Dec. 1975	
Status: Applied for prolongation / Last Updating (Date): 31. 12. 1975	

General Aim and Particular Objectives

Testing of pressure and differential pressure transducers suitable for operation at temperatures up to 150 °C and pressures of 5 bar.

Experimental Facilities and Program

In order to increase reliability under operation and MCA the primary pressure and differential pressure transducers shall be arranged within the safety shell.

Until now data of the reliability of suitable temperature and pressure transducers are only available from the producers. Before they are installed in the reactor, it is indispensable to test "wet" transducers being on the market under simulated MCA-conditions (ϑ = 150 °C, p = 5 bar). If the transducers do not satisfy this conditions it is intended to weld in automatic tube rupture fuses in the pulse pipes. Several types are to be tested and eventually improved.

Project Status/Progress to Date

Tests with pressure and differential pressure transducers of various producers were conducted. Ranges on test were 0 - 210 bar and 0 - 10 bar.

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Project Status/Essential Results

Transducers of one producer have not satisfied the simulated MCA conditions. The instruments were destroyed by moisture or showed rather big measuring errors (up to 11 %). The tests showed that casting of the electric components does not secure operation under MCA conditions.

Transducers of another producer worked however quite well under simulated MCA conditions (N_2-H_2O vapour-atmosphere, of 150 °C and 5,8 bar overpressure). After cooling and adjusting the transducers reached the specified measuring accuracy. No destruction was found on the instrumentation and electronic components after inspection. Another MCA test was conducted at 180 °C for half an hour; the transducers operated without difficulties.

The transducers had an error of + 0,2 % to + 1,1 %, the differential transducers of + 0,7 % to + 2,5 %.

Next Steps

The equipment of the transducers, which have failed, will be improved (pressure tight boxes). After this, the MCA tests will be repeated.

Another producer has offered transducers for a test under MCA conditions.

Relation with Other Projects

see RS 110 - A 74

Reference Documents/Degree of Availability

No reports available.

Classification: 8	
Title 1 (Original Language): Einsatz von Prozeßrechnern in Leistungsreaktoren zur Verbesserung der Betriebssicherheit (ATT 085 A - II.4.4., Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (English): The Use of Process Computers in Power Reactors for Improvement of Operation Safety	Project Leader: Dr. H. Hoermann Dr. W. Ehrenberger
Initiated (Date): Completed (Date): Status: continuing Last Updating (Date): December 1976	

1. General Aim

(a)

Improvement of the operational safety through the use of computerized inspection and plant supervision techniques.

(b)

Development of methods and guidelines for the construction of safe process computer systems and for the verification of desired safety or reliability margins.

2. Particular Objectives

(a)

Development of a general concept for computerized inspection of engineered safety systems, in order to improve failure detection and localization with respect to time conditions and overall performance. Investigation of the capability of new methods for on-line analysis of plant disturbances. Development of methods to implement new concepts for incipient failure detection using process computers.

(b)

Investigation of hardware and software structures for safety related computer applications in nuclear power plants. Reliability and availability evaluation of such systems in comparison with hardwired solutions. Development of guidelines for the design, construction and implementation of computer systems with safety related applications; providing methods for the verification of their reliability or their correct performance.

3. Experimental Facilities and Program

(b)

For the tests performed on a computerized reactor protection system for BWRs an AEG 60-10 computer has been installed at the Laboratory.

4. Project Status

4.1 Progress to Date

(a)

Further work has been deferred in favour of more time-critical subjects.

(b)

A contribution was made to the VDI/VDE guidelines number 3553 on "Failure Detection and Localization in Process Computer Systems". The public interest in recommendations for the construction of safety related user programs /1, 2/ proved to be quite high.

Concerning the computerized protection system for the Brunsbüttel power plant (KKB) more than 10^5 additional test runs had been performed. It turned out that with regard to some points an obsolete version of the specification had been used and that the employed reaction mode for reactor pressure changes worked differently to a normal two out of three selection. The computer self supervision programs were improved, now providing a better computer availability. Other investigations helped to clarify the causes of faults which occurred intermittently in the computers installed at the plant. ADCs of that system were remeasured, faults in the VVT (process interface) and concerning the IBM type writer must be corrected.

Theoretical investigations dealt with the treatment of several hardware failure modes differing in the associated failure detection times. Additionally data depending failure detection methods were considered with respect to their effectiveness. In the software field redundant and diverse programming methods were investigated and the effects of common data areas on possibly distant program parts in operating systems were looked at. Statistical software verification dealt with the accuracy of fixing of boundaries, the use of supervising a program's test procedure and the results obtainable by test during plant operation.

The comparison of a hardwired and a computerized protection system was completed as far as purely safety related actions are concerned. The comparison proceeded with not entirely safety related actions. Some preliminary calculations have been made, but no significant results were obtained so far.

4.2 Essential Results

(a) -

(b)

The comparison of the two protection systems showed,that present state hardwired systems are in view of safety advantageous compared with computerized ones, but it seems to be feasible to bring computerized systems to an equally high safety level.

It is in general not sufficient to evaluate a program's reliability only out of the results gained by a supervision of the conventional testing procedure or some operating experience in the plant.

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Some formulae for taking into account different failure detection times for some hardware failures were derived. Data depending failure detection at least in some cases seems to be a valuable detection tool for dealing with component faults.

5. Next Steps

(a) -

(b)

The comparison of computerized and hardwired protection systems will continue. Protection measures which are not purely safety related and common mode failures will be considered.

It will be investigated, how far the methods used to verify self supervision programs for the 3rd computer generation's hardware can be employed for micro processors.

Some aspects of statistical verification of program correctness will be summarized, taking into account corresponding results of program analysis.

The safety-oriented work concerning protection limiting computer systems in development for PWRs will be intensified.

6. Relation with Other Projects

The project is related to the following activities:

- Project Nuclear Safety (GfK)
- Project Fast Breeder Reactor (GfK, IA)
- Project Process Control with DP-Systems (GfK)
- Halden Reactor Project (OECD)
- CNEN (Italy).

7. Reference Documents

/1/
W. Ehrenberger, J.R. Taylor
Software for Safety Related Systems
Paper presented at the Meeting of Purdue Europe, Zürich, April 1976

/2/
W. Ehrenberger, U. Voges
Programmierempfehlungen für sicherheitsrelevante Anwenderprogramme
Vortrag auf der Tagung des SIEMENS Anwenderkreises für Prozeßrechner
Ulm, April 1976

/3/
K.P. Volkmann, H. Hoermann, W. Ehrenberger
Statistical Test Data Selection for Reliability Evaluation of Process Computer Software
Paper presented at IAEA/NPPCI Specialists' Meeting on the Use of Computers for Protection Systems and Automatic Control, Neuherberg/München, May 1976

/4/
W. Ehrenberger, K. Okroy
A Basis for an Automatic Analysis of Sequential Process Computer Programs
Paper presented at the 6th European Workshop on Real Time Programming, Roquenfort, France, June 1976

/5/
W.E. Büttner
Sicherheitstechnischer Vergleich eines festverdrahteten dynamischen Reaktorschutzsystems. Teil I
MRR 161, Juli 1976

/6/
W. Ehrenberger, G. Rauch, K. Okroy
Program Analysis - A Method for the Verification of Software for the Control of a Nuclear Reactor
Paper presented at the 2nd International Conference on Software Engineering, San Francisco, California, 13. - 15. October 1976, IEEE Catalog No. 76CH1125-4c

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8. Degree of Availability

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

Classification: 8	
Title 1 (Original Language): Untersuchungen zur Funktionstüchtigkeit der Druckhaltersicherheitsventile und des Abblasetanks beim Abblasen von heißem Druckwasser (RS 240 - I.1.8, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Investigations on the functioning of the pressurizer safety valves and the relief tank during blowdown of hot pressurized water	Project Leader: H. Landgraf
Initiated (Date): 1. 12. 76 Status: Continuing	Completed (Date): 28. 2. 78 Last Updating (Date): 31. 12. 76

1. General Aim

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

2. Particular Objectives

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 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.

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3. Research Program

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

Detailed planning of the test sequence for test of pilot valves at the Erlangen valve test facility has been initiated. Hydraulic design calculations covering the planned test circuit are being performed. Discussions of technical details pertaining to the functional behaviour of the safety valves have been held with the manufacturer of the test valves, Messrs. Sempell. Preparatory work covering selection and ordering of the test components continues.

Results of the blowdown tests with regard to pressure build-up in the relief tank dome and the relief tank itself have been verified by the computer codes "DOMFREI" and "ABTAN".The computer program "ABTAN" is based on a calculational model which is identical to that used for the computer program "ABTAN" which is used for the relief tank design.

6. Results

Comparison of measured and calculated data show that the design pressure of the blowdown dome determined by the computer code "DOMFREI", is in relatively good agreement with the measured data.

7. Next Steps

Completion of the schedule of the test arrangement. Construction and manufacture of loop components, modification of supporting structure to valve test facility requirements. Ordering material and components from suppliers (tubing, flange connections, valves, insulation, compressor). Assembly of test arrangement.

A computer program of the pressure build-up in the relief tank which provides an improved analysis of the conditions present in the relief tank, will be developed.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Titre	Pays : FRANCE
Qualification de la chaine de mesure neutronique à grande dynamique.	Organisme directeur : CEA/DSN
Titre (anglais) Qualification of wide range nuclear instrumentation.	Organisme exécuteur : CEA/SES (Saclay)
	Responsable : M.LE MEUR (DSN-SETSSR)
Date de démarrage : 01/01/76 Date prévue d'achèvement : 31/12/77 Etat actuel : Etude en cours Dernière mise à jour : 19/11/76	Scientifiques : Y.PLAIGE (SES)

Objectif général :

Influence de l'activité du détecteur sur son fonctionnement à bas niveau.
Linéarité à haut niveau.
Recoupement des mesures impulsion-fluctuation.
Influence du rayonnement GAMMA dans la zone de recoupement.
Comportement de la chaine en régime d'accident.

Objectifs particuliers :

Influence de la longueur du cable sur la mesure. Possibilité d'adaption à des détecteurs lents.

Prochaines étapes :

Qualification de ces chaines pour leur utilisation dans le système de protection.

Documents de référence :

"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.

133-41-20/4114-10

Classification : 8

1033

Titre	Pays :
Essais des compteurs de démarrage des centrales PWR.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Test of proportional counters used in PWR reactors.	Organisme exécuteur : CEA/SES (Saclay)
	Responsable : M. LE MEUR (DSN-SETSSR)
Date de démarrage : 01/06/76 Date prévue d'achèvement : 31/12/78 Etat actuel : Etude en cours Dernière mise à jour : 22/11/76	Scientifiques : J.DUCHENE (SES)

Objectif général :

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

Objectifs particuliers :

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

Avancement à ce jour :

Approvisionnement des compteurs en cours de réalisation.

Prochaines étapes :

Essais de qualification d'une année sur TRITON.

133-41-30/4114-10

Classification : 8

1035

Titre	Pays :
Etude du comportement des câbles auto-extinguibles aux silicones lors d'un A.D.R.	FRANCE
	Organisme directeur : CEA
Titre (anglais) Study of method for aging qualification of silicone non flamme propagating cable.	Organisme exécuteur : CEA/DCA-SCAPR
	Responsable : M. LE MEUR
Date de démarrage : 01/06/76 Date prévue d'achèvement : 01/01/79 Etat actuel : Etude en cours Dernière mise à jour : 22/11/76	Scientifiques : M. LAIZIER

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.

Etat de l'étude :

Avancement à ce jour :

Etude bibliographique en cours.

Prochaines étapes :

Qualification des câbles auto-extinguibles aux silicones pour système de protection.

Relation avec d'autres études :

Etude mission 4 : Comportement des matériaux aux ambiances nucléaires.

Documents de référence :

"Multiplexage bas niveau - principes et application à la mesure des températures", Note SES/SAI-76-06, janvier 1976.
"Système de multiplexage pour mesure de température", Note SES/SAI 76-87, avril 1976.

.../...

133-41-41/4112-04

Classification : 8

1037

Titre Test en ligne des capteurs.	Pays : FRANCE
	Organisme directeur : CEA
Titre (anglais) On-line method for testing of instrumentation.	Organisme exécuteur : CEA/SES-SAI
	Responsable : M.DEISS (SAI)
Date de démarrage : 01/01/77 Date prévue d'achèvement : 01/12/78 Etat actuel : à lancer Dernière mise à jour : 15/02/77	Scientifiques :

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.

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.

Etat de l'étude :

Avancement à ce jour :

L'étude débute actuellement.

133-41-42/4130-01

: Classification : . 8

-1039

Titre	Pays :
Méthodes et appareillages d'essais pour le contrôle de l'instrumentation.	FRANCE
	Organisme directeur : CEA
Titre (anglais) Method and test device for instrumentation control.	Organisme exécuteur : CEA/SES (Saclay)
	Responsable : J.BUISSON (SES)
Date de démarrage : 01/01/74 Date prévue d'achèvement : 01/12/78 Etat actuel : en cours Dernière mise à jour : 15/02/77	Scientifiques :

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.

Objectifs particuliers :

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

Etat de l'étude :

1) Avancement à ce jour :

Fin de l'étude d'un appareillage prototype. Mise au point d'un appareillage plus sensible pour la détection des parasites.

2) Résultats essentiels :

Contrôle de la validité de la méthode de détection et mise en évidence de défauts sur des chaînes de mesure.

Prochaines étapes :

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

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Documents de référence :

"Méthode de mesure de l'immunité aux parasites d'un ensemble de mesure neutronique", J. BUISSON - Note SES/interne/SAI - 76/152.

"Méthode d'essais et de mesure de la sensibilité aux perturbations des chaînes de contrôle neutronique et de détection de rupture de gaine", J. BUISSON - Note SES/interne/SAI - 76/263.

"Contrôle de la susceptibilité aux perturbations des ensembles de démarrage de Fessenheim", J. BUISSON - Notes SES/interne/SAI - 76/99 et 76/211.

1041

133-41-43/4114-03

Classification : 8

Titre	Pays : FRANCE
Définition des essais de vieillissement sous irradiation.	Organisme directeur : CEA
Titre (anglais) Definition of aging qualification tests.	Organisme exécuteur : CEA/SES-Saclay
	Responsable : J.BUISSON (SES)
Date de démarrage : 01/04/77 Date prévue d'achèvement : 01/01/81 tat actuel : A lancer Dernière mise à jour : 23/03/77	Scientifiques :

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.

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 30 à 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 fait, 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.) effectués 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

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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.

133-41-44/4114-02

Classification : 8 1043

Titre	Pays : FRANCE
Critères de sûreté pour l'utilisation des systèmes séquentiels dans le système de protection.	Organisme directeur : CEA
Titre (anglais) Safety criteria for using microprocessor in the protection system.	Organisme exécuteur : CEA/SES-(Saclay) Responsable : G.ZWINGELSTEIN (SES)
Date de démarrage : 01/01/77 Date prévue d'achèvement 01/02/79 Etat actuel : à lancer Dernière mise à jour : 15/02/77	Scientifiques :

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.

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".

Etat de l'étude :

Avancement à ce jour :

L'étude débute actuellement.

Prochaines étapes :

Application d'une procédure de qualification à un exemple concret.

133-41-51/4115-06

Classification : 8
1045

Titre	Pays :
Appareillages de mesure utilisés dans les réacteurs nucléaires.	FRANCE
	Organisme directeur : CEA
Titre (anglais)	Organisme exécuteur : CEA/DRIS-LCRI
Measure instrumentation in nuclear reactors.	Responsable : M. BOUTEILLER (LCRI)
Date de démarrage : 01/01/75 Date prévue d'achèvement : 31/12/78 Etat actuel : en cours Dernière mise à jour : 15/02/77	Scientifiques :

Objectif général :

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

Objectifs particuliers :

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

Installations expérimentales et programme :

TRITON (sources).
Enceintes pour réaliser des ambiances d'accidents.

Etat de l'étude :

1) Avancement à ce jour :

Définition des caractéristiques des ambiances (gamma - neutrons).
Mesures effectuées à TRITON, à blanc, sans matériel.

2) Résultats essentiels :

Les méthodes sont définies.

Prochaines étapes :

Détermination d'ambiances de référence en accord avec les normes "Westinghouse".
Définir les conditions d'essais équivalentes à une durée de vie du matériel de 40 ans.

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1046

Etudier et exécuter les enceintes pour la réalisation d'ambiances d'accidents.
Effectuer les essais de matériels.

Documents de référence :

"Détermination du spectre neutronique dans la fosse NAIADE auprès du réacteur NEREIDE", A.CAPGRAS et C.CLEMENT - Note LMRI, décembre 1976.
"Absorbées photoniques élevées. Utilisation du film plastique "TAC" pour la mesure de doses", JP.SIMOEN - Note LMRI, janvier 1977.
"Etude, réalisation et étalonnage d'un dispositif de flux de neutrons", C.RONTEIX - Note LMRI, février 1977.

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 Sponsor: CNEN Organisation: CNEN
Date initiated 1/1/1975 Date completed in progress Last updating June 1976	Project Leader 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.

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1048

TITLE 1 (original language)	Classification
Diagnostica e Sicurezza tramite Analisi di Rumore	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 Sponsor CNEN Organisation CNEN
Date initiated 1/1/1975 Date completed in progress Last updating June 1977	Project Leader 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).

-1050

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.

1051

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 Sponsor CNEN Organisation CNEN
Date initiated 1/1/1976 Date completed in progress Last updating June 1977	Project Leader 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.

1052

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
Sistema di controllo di reattori con barre bifasi	8
Title 2 (English)	**Country** ITALY
Reactor control system by two-phase rods	**Sponsor** CNEN **Organisation** CISE
Date initiated 1965 **Date completed** 1976 **Last updating** April 1977	**Project Leader** 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.

1054

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

7. Degree of availability: to a limited extent

1055.

PROJECT TITLE : Reactor Safety Studies via Noise Analysis	CLASSIFICATION $\boxed{14}$ - 8
SPONSORING COUNTRY : Italy	ORGANISATION : C.N.E.N.
DATE INITIATED : 1.1.1975 DATE COMPLETED : .... (in progress)	PROJECT LEADER N. Pacilio, LFCR

PROJECT TITLE :	CLASSIFICATION
Statistical analysis of random signals for safety problems	$\boxed{10}$ - 14 - 8
SPONSORING COUNTRY : Italy	ORGANISATION : CNEN
DATE INITIATED : 1966 DATE COMPLETED : in progress	PROJECT LEADER : A. Federico

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Title 1 (Original language)	Classification
Dynamic studies for safety analysis	$\boxed{4}$ - 8
Title 2 (English)	Country ITALY Sponsor Organisation CNEN
Date initiated 1962 Date completed in progress Last updating April 1977	Project Leader M. Di Bartolomeo

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

9. OTHER SAFEGUARDS

Classification: 9	
Title 1 (Original Language): Untersuchungsprogramm zur Erprobung einer Berstsicherung für Reaktorkomponenten (RS 104 - I.1.7, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** KWU, Erlangen
Title 2 (English): Investigation Program for the Testing of a Fracture Safety Device Protection System for Reactor Components	**Project Leader:** Dr. Kopp
Initiated (Date): 5. 2. 73 **Status:** Completed **Completed (Date):** 31. 12. 76 **Last Updating (Date):** 31. 12. 76	

1. General Aim

Under this project, burst tests were to be carried out on burst-protected pipe cross-sections of various dimensions in order to prove the effectiveness and functional worthiness of a burst protection structure. The break behaviour of shielded main coolant pipes, phenomena subsequent to a fracture and the loading of the burst protection device system were to be determined.

2. Particular Objectives

The project was being used simultaneously for the technical estimation of phenomena and loading within the scope of the research project "Fracture Safety Device for Reactor Pressure Vessels".

3. Research Program

The investigation program encompassed the following works:

a) Preliminary Tests on Burst-Protection Structure Components
b) Preliminary Tests on Pipe Elements
c) Preliminary Tests on Pipe Elements being Protected by a Burst Protection Strucutre
d) Primary Experiments for Testing the Burst Protection Structure

4. Experimental Facilities

The tests were performed in a separate building in the Forschungszentrum Erlangen.

5. Progress to Date

The fracture tests were carried out with burst protected tubes of 354 mm diameter. One test was carried out with a non protected tube with a slit penetrating through the wall.

Two tests were performed with a tube which had a longitudinal slit of 1600 mm length. The mean residual wall thickness was 4,77 mm. Both tubes refused under well defined conditions. The tubes were instrumented with a various number of transducers for pressure, temperature, strain and acceleration.

6. Results

The fracture protected tube ruptured at a temperature of 300 °C and a pressure of about 160 bar. The instrumentation worked well, but after a few milliseconds it was destroyed by the shock forces. The cracks had grown 2,5 - 3 mm on the surface of the tube.

The non protected tube ruptured at a temperature of 300 °C and a pressure of about 250 bar. The whole crack length was about 400 mm after fracture.

The last tube ruptured at a temperature of 300 °C and a pressure of about 150 bar. This last test is not evaluated.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

RS 108 Fracture Safety Device for the Primary Circuit

9. References

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10. Degree of Availability

Classification: 9	
Title 1 (Original Language): Berstsicherheit für Primärkreislauf (RS 108 - I.1.7; Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Fracture Safety for the Primary Circuit	Project Leader: Dr. Dorner
Initiated (Date): 1. 8. 73 Status: Continuing Completed (Date): 30. 6. 78 Last Updating (Date): 31. 12. 76	

1. General Aim and 2. Particular Objectives

A concept of a fracture safety device for the primary circuit had to be developed, which offered chanches 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. Desing of a fracture safety device for the primary circuit (piping)

1066

4. Experimental Facilities

No experimental facilities were necessary.

5. Progress to Date

The fixing of the RPV in the fracture safety device was designed and calculated.

The missile shield construction of the steam generator was designed, the material will be the fine grain steel 17MnCrMo33.

The wall-thickness for the missile shield of the pressurizer was calculated. A preliminary fracture protection device for the pressurizer was designed.

For the support of the steam generators heat resistent elastomeric bearings, and an adhesive for the steel armor were chosen.

The concept for the reactor building was redesigned in order to get more room for the fracture protection cylinders.

6. Results

The RPV is fixed in the fracture protection system by 32 necked-down bolts (1100 mm long, 300 mm broad, 250 mm high), which swing into their position by pneumatic zylinders. The top faces are cylindrical, the adjacent bearings have the surface of a hollow cylinder with a radius of 1100 mm. The cladding must be hardened steel.

The fracture protection cylinders of the steam generator are designed for a tolerable stress of $0{,}7 \cdot \sigma_B$ in the case of fracture.

The fracture protection for the pressurizer can be constructed analogous to the steam generator. A problem is the change of the heater-elements.

The experimental tests with the heat resistant elastomeric bearings showed that VITON was not suited for the steam generator set-up. Pressure tests at a temperature of 150°C showed that the elastomere layers are destroyed in a rather short time. Silicone and EPDM caouchuc were much better, the long time behaviour is expected to be sufficient.

7. Next Steps

The next steps are in preparation.

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

Classification: 9	
Title 1 (Original Language): Berstsicherheit für Primärkreislauf (RS 108 - I.1.7, Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (english): Fracture Safety Device for the Primary Circuit	Project Leader: Dr. Dorner
Initiated (Date): Aug. 1973 Completed (Date): April 1976 Status: Continuing Last Updating (Date): December 1975	

General Aim and Particular Objectives

The concept of a fracture safety device for the primary circuit has to be developed, which offers changes for a better protection after MCA. This is relieved by components with simple geometry and small surfaces. Under this project investigations are 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.

Experimental Facilities and Research Program

The investigation program encompasses 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)

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Project Status/Progress to Date

Model tests on semi-axial pumps were completed. Analytical tests on the vibration performance of steam generator primary chamber components were made. Stress analysis of the steam generator primary components as well as determination of the eigen-frequencies are being conducted using the computer program STARDYN. Geometries of the semi-axial and axial pump components were reproduced for use in the program.

Burst protection work on the tubing which consists of single interlocking rings has been finalized for the time being.

An order to Messrs. Krupp was being prepared covering the development of a missile shield cylinder for steam generator burst protection. The cylinder will preferably be fabricated as multilayer construction. A change was made in the missile shield so that the multilayer rings were welded to one cylinder. Thus the cylinder can also absorb axial loads. Consequently, the axial rods can be neglected. Further advantage of this design is a simplified closure head of the missile shield achieved by means of shear wedges.

Major part of the work concerning construction and calculation of the hydraulic cylinder for tightening the steam generator to the RPV burst protection has been completed. Detailed calculations on the lower bottom plate of the steam generator missile shield resulted in stress values which were too high. It is assumed that this will require an increase of the height of the bottom plate from 1.20 m to 1.40 m. Centering of the steam generator with the bottom plate was prepared, which positions the steam generator by three legs welded to the bottom of the primary chamber, absorbing the forces and moments which might occur.

Calculations have been performed on the upper locking device of the RPV burst protection. The dimensions of the supporting flange have been optimized. Calculations showed that slight relative movements might occur between the two construction units due to the temperature differences in the supporting flange and RPV top during start-up and shut down. The RPV support construction with round springs, i. e. necked-down bolts has been optimized.

Since elastomer bearings seem to be extremely useful for the steam generator assembly, the following tests have been performed on full size bearings:

- pressure tests
- shear test
- shear break test
- pressure test with eccentric vertical stress
- temperature test
- accompanying tests (e. g. Shore-hardness)

In anticipation of the test results it can be said that the tests proved the suitability of the elastomer bearings.

Project Status/Essential Results

Use of the round spring rods for RPV support construction results in sufficient design safety factors. Like the turbine vane anchorage, the U-shaped springs, having a rectangular cross-section (100 x 30 mm), will be tightened to the spring legs by a lugged support. A lateral lift-up of the springs will be eliminated by a cross-piece attached laterally to the lugger. This construction, however, requires an immense amount of work and a very costly and exact manufacturing process.

The primary chamber of the steam generator will be provided with a thicker forged ring and attached nozzles (as have been used before). The main steam nozzle was moved up to the upper center and the steam generator cut down for the length of the inside steam guide tube, which is no longer needed.

The results of the tests performed on the elastomer bearings prove the suitability of the type bearing for steam generator assembly; they also indicate, however, that influence of temperature has to be examined in more detail. Construction and calculation of the hydraulic cylinder for tightening the steam generator to the RPV burst protection has been finalized.

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Next Steps

Stress calculation of the lower bottom (height: 1.40 m) of steam generator missile shield construction. Armored elastomer bearings are preferred with bridge construction and temperatures of about -20 to +30 °C. Since in the discussed construction temperatures are involved which will be about 100° higher, the influence of temperature has to be investigated in more detail. First of all one has to find out whether or not Chloropren-caouchuc, which is usually used for bearings for bridge constructions, will be the optimum elastomer for our purposes. It can be seen from the literature that different types of elastomer e. g. Silicone-caouchuc or caouchuc fluoride should used with increased temperatures.

The components of the steam generator primary chambers will be calculated with the STARDYN program.

Detailed tests on the reactor components and burst protections will be continued.

Relation with Other Projects

see RS 104

Reference Documents / Degree of Availability

No reports available.

Classification: 9	
Title 1 (Original Language): Untersuchung des Verhaltens des Wärmedämm-Kühl-Systems (WKS) der Berstsicherung (BS) für RDB bei dynamischer Belastung (RS 157 - I.1.7., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Dyckerhoff & Widmann AG, München
Title 2 (english): Investigation of the Heat-Insulting Cooling-System under Dynamic Load Conditions with a View to Fracture Safety Device Protection for Reactor Pressure Vessels	Project Leader: O. Seidl Dr. Bittner
Initiated (Date): 15.3.1975 Status: continuing	Completed (Date): 31.12.1975 Last Updating (Date): December 1975

1. General aim

In order to establish absolutely reliable characteristic data for the behavior of reactor pressure vessels during the fracturing process, a series of dynamic loading experiments was carried out.

2. Particular objectives

The experiments involve the heat-insulating cooling system. For this purpose a test set-up using lightweight insulation concrete (Bn 550) both with and without a special filling material (spherical steatit, Ø 20 mm) was constructed in model form on a scale of 1:1 and subsequently subjected to various dynamic loadings. The test Specimens were 500 and 260 mm in diameter, the test length 700 and 400 mm. Gaps of 20, 12, 6 and 0 mm were fixed in order to allow free acceleration paths.

3.1 Experimental facilities

It was possible to obtain time-deformation functions on the basis of this test project.

3.2 Research program

The load was applied dynamically by means of a system of spring-coupled masses, consisting of steel plates and 58 m-long DYWIDAG-prestressing bars (quality 110/135).

Since no experimental values for the dynamic behavior of concrete with this type of filling material were available, it was first necessary to establish limit values. These limit values were obtained by varying the specific unit pressures between the material causing the impact (steel plates) and the material sustaining the impact (lightweight insulation concrete). The specific unit pressures provided for in the test program were approximately 175 bar at the load application point, i.e. between the steel and the concrete, and 40 and 175 bar at the load removal point, i.e. from the filling material to the support structure. The loads applied ranged from 100 to 500 Mp.

The loads were applied dynamically by means of the prestressing bars and a hinged nut.

The test values were recorded with the aid of electronic measuring equipment.

The strains in the prestressed bars were determined by means of strain gages.

Carrier frequency and direct-current bridges were used for the measurements.

The dynamic test values were stored with a magnetic tape recorder (frequency range 0 - 15 kHz) and subsequently recorded via an ultra-violet printer (frequency range 0 - 13 kHz).

The load application time was 1 - 2 ms. The times required for a maximum of elastic and plastic deformations to occur range from 5 to 10 ms.

For each dynamic loading process with the associated accelerations there are at least three test recordings suitable for evaluation purposes.

4. Project status

The tests were carried out according to plan. The course of the impact through the test material was determined integrally between the load application point and the transmission of the load to the support structure with the aid of time and deformation measurements.

The recordings, in the form of ultra-violet diagrams, (time-deformation diagrams), provide exact information on the dynamic loading and unloading processes as a function of time.

The acceleration rates of the impact plate can be read off these diagrams at each point in the recordings by examining the variation in the tangent curve.

The paths of the prestressing bar ends - and consequently the movements of the impact plate as a function of time - can be determined by step-by-step integration of the measured values along the prestressing bar. The time-deformation function at the load application point is thus obtained. The measured values at the measuring cylinder indicate the load-time function.

The load-deformation function for the complete test is obtained by combining these two functions.

The final report contains a summary of the test values, including a series of detailed diagrams.

5. Next steps

It is our intention to extend the range of experiments to include the acceleration and deformation processes within the test specimens. To this end the test program is to be extended by incorporating a number of experiments which will be of considerable importance for calculation purposes. New improved electronic instrumentation will be used. Whereas previous experiments were invariably one-dimensional, plans have been made to expand the test setup to include two-dimensional loading. Further proposals are being investigated and adopted at the present time.

6. Relation with other projects

The tests described in this report proved characteristic data for the fracture calculations on reactor pressure vessels which are currently being carried out by the Kraftwerk Union, Erlangen.

7. Reference documents

Forschungsprogramm Reaktorsicherheit Abschlußbericht Förderungsvorhaben BMFT RS 53, Voruntersuchungen zum Programm Berstsicherheit für Reaktordruckbehälter KWU, Reaktortechnik Erlangen Februar 1974.

Berstfolgeschutz aus Spannbeton für Reaktor-Druck-Behälter von Druckwasserreaktoren 600 MWe, Forschungs- und Entwicklungsauftrag BMFT RS 108 - KWU BS 006D, Bericht 1975, von Fa. Dyckerhoff & Widmann AG, München.

Dynamische Belastungsversuche und Bestimmung des Stoßfaktors für die Spannglieder des Kernkraftwerkes der BASF, Dyckerhoff & Widmann AG, München 1974.

Abschlußbericht Nr. 1 "Spezielle Materialmodelle für PISCES-Berechnungen, KWU, Jan. 1975.

8. Degree of availability

Some of the tests are being recalculated by KWU Erlangen using the PISCES program in order to compare the accuracy of the PISCES calculation for the fracturing of reactors.

Classification: 9	
Title 1 (Original Language): Funktionsversuch zum Schwungradabfall, 2. Versuch (RS 168 - II.1.6., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** KWU, Erlangen
Title 2 (english): Functional Test of an Automatic Fly-Wheel Detach at High Pump Rotation Speed	**Project Leader:** H. Schäfer
Initiated (Date): 20. 6. 75 — **Completed (Date):** 31. 10. 75	
Status: Completed — **Last Updating (Date):** 31. 12. 75	

General Aim/Particular Objectives

In a functional test the detach process of a fly-wheel of a primary coolant pump was investigated in order to verify the precalculated rotation velocity where the fly-wheel should detach and to get informations of the detach-procedure and the slow down behaviour of the fly-wheel.

Experimental/Facilities/Research Program

A test equipment was built up by KWU, which imitates the original in all essential parts. As drive, a motor of a fly-wheel bunker was used.

Project Status/Progress to Date

The test equipment was built in a fly-wheel-bunker. The T.V. and measuring lines were installed. A test run was carried out and the measuring values as well as the T.V. pictures were recorded till the standstill of the fly-wheel. The results were evaluated and reportet to the IRS.

Project Status/Essential Results

The fly-wheel was detatched at the precalculated rotation velocity and was slowed down safety in the hold up equipment.
The analysis of the experimental data showed good agreement with the theoretical investigations.

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Next Steps

Work on this project has been completed.

Relation with Other Projects

No relations with other projects.

Reference Documents/Degree of Availability

No reports available.

Classification 9.1, 9.2, 9.3	
Title 1 Rekenmodel voor de simulatie van transiënten in kokendwaterreactoren	Country The Netherlands Organization KEMA
Title 2 Computercode for the simulation of transients in boiling water reactors	Projectleader R.M. van Kuijk
Initiated Completed 1977 Status Progressing	Scientist P. Kloeg

1. General aim

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

2. Particular objectives

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

3. Experimental facilities and programme

Computer code REBOR.

4. Project status

Operational for Dodewaard BWR.

5. Next steps

Other types of BWR's.
Comparisans with experiments in Dodewaard.

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6. Relation with other projects

Not applicable.

ference documents

Internal KEMA reports.

8. Degree of availability

Free on basis of exchange with other programmes.

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Classification $\boxed{9.1}$, 9.2, 9.3	
Title 1 Rekenmodel voor de simulatie van transiënten in kokendwaterreactoren	Country The Netherlands Organization KEMA
Title 2 Computercode for the simulation of transients in boiling water reactors	Projectleader R.M. van Kuijk
Initiated Completed 1977 Status Progressing	Scientist P. Kloeg

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Classification 9.1, 9.2, 9.3	
Title 1 Rekenmodel voor de simulatie van transiënten in kokendwaterreactoren	**Country** The Netherlands **Organization** KEMA
Title 2 Computercode for the simulation of transients in boiling water reactors	**Projectleader** R.M. van Kuijk
Initiated **Completed** 1977 **Status** Progressing	**Scientist** P. Kloeg

10. CORE AND PRIMARY CIRCUIT IN STEADY STATE CONDITIONS

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

Description : 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: early failure detection).

Facilities

- 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.

Reference documents

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

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

Classification: 10.1

Title 1 (Original Language):	COUNTRY: BRD
Anwendung statistischer Analysenverfahren in Leistungsreaktoren mit dem sicherheitstechnischen Ziel der Früherkennung von Schäden (RS 68 A - II.4.1, Jahresbericht A 76)	SPONSOR: BMFT ORGANIZATION: TU Hannover
Title 2 (English): Application of Statistical Analysis Methods in Power Reactors under the Safety Oriented Aspects of Early Fault Detection	Project Leader: Prof.Dr.-Ing. D.Stegemann Institut für Kerntechnik
Initiated (Date): 1.3.1975 — Completed (Date): 28.2.1977	
Status: Continuing — Last Updating (Date): December 1976	

1. General aim

The aim of these investigations is the development of an early fault detection system for power reactors, based on the analysis of the stochastic neutron flux fluctuations during operation and correlation with other available signals of relevant reactor parameters.

2. Particular Objectives

Noise measurements in both Boiling- and Pressurized Water Reactors in order to gain extensive experience about space-dependent reactivity effects caused by pressure-, vibration- or coolant density feedback.
Development of a new and reliable in-core measuring system, using "self-powered detectors", since in most cases a suitable instrumentation is not available.
Application of these improved devices for neutron flux distribution measurements with high spatial resolution and for reactor power noise measurements without disturbance of the normal plant operation.

3. Research Program

Continuous measurements at the PWR Obrigheim (KWO), concerning axial neutron flux distribution and local neutron noise, by means of a large number of self-powered neutron detectors.
Instrumentation of three new in-core assemblies for the 8th fuel cycle at KWO.

Development of improved self-powered-detectors and investigation of their signal components due to neutron and gamma fluxes.
Systematic investigations of the space dependent neutron flux fluctuations in the boiling water reactor Lingen (KWL), using different types of SPN-detectors, which could be inserted in a tube of the TIP-system and moved at any axial position in the core.
Computer analysis of the recorded stochastic signals and evaluation with respect to the identification of noise sources which might be relevant for safe reactor operation.

4. Experimental Facilities, Computer Codes

The data acquisition system consists of several axially and/or radially distributed in-core self-powered-detectors, connected to differential wide band amplifiers, analog frequency filters and an analog magnetic tape station.
In the KWO-PWR several instrumentation tubes, each with six Co-SPN-Detectors, are available.
For measurements in the KWL-BWR a three SPN-detector assembly was manufactured and inserted into a dry thimble of the TIP-System. This assembly could be vertically positioned in the core as desired.
Flux distribution measurements have been performed by using the DC-component of the detector signals while suppression of the stationary value and further amplification yields the fluctuating part which had been recorded on magnetic tape.
The collected data were processed later on to a digital computer via special frequency filters and analog-to-digital converter (ADC).
A sophisticated program system was developed for computer analysis of the stored signals in time and frequency domain.

5. Progress to Date

Four in-core probes for the KWO Pressurized Water Reactor have been installed in Juli 1976. Each probe contains 6 Cobalt-SPN-Detectors. Manufacture of 18 detectors and instrumentation of three assemblies was performed by the Institut für Kerntechnik during the first months of 1976. The 4th probe was instrumented with commercial detector types. The detector signals have been recorded and analysed with respect to flux distributions and local and global reactivity feedback effects. Systematic measurements of neutron flux distribution and space dependent

noise phenomena were carried out at the boiling water reactor Lingen (KWL), using a three-detector arrangement with Hafnium and Gadolinium detectors.
The data were stored on magnetic tape and prepared for an extensive computer analysis program.

6. Results

A large amount of data was obtained from the KWO-mueasurements in 4 different radial core positions and 6 vertical planes. The results of the computer analysis show remarkable structures in the neutron noise spectra, depending upon radial and vertical position of the detector, power distribution and thermohydraulic parameters. Up to 14 resonances could be found in the frequency range from 0 to 120 Hz.
For in-core measurements in the boiling water reactor Lingen a new detector type with Gadolinium emitter and two Hafnium detectors were applied. Because of the special design of the Hf-detectors extremly high neutron signals could be measured with an excellent spatial resolution of only 2,2 %. Also the Gd-SPN had the expected high signal level of similar magnitude. Neutron flux distribution and reactor noise measurements were performed in 30 vertical core positions.

7. Next Steps

Analysis and evaluation of the last experiments as well as comparison with earlier measurements at both Boiling Water and Pressurized Water Reactors will be completed in the next months. Experimental results and corresponding theoretical investigations are to be published in the final project report.

8. Relation with other Projects

See IRS-F-24

9. References

Quarterly reports V 76/1,.....V 76/4 (german)
Zwischenbericht zum Förderungsvorhaben RS 68A, Jan 1976 (german)
Bestimmung der Dampfblasengeschwindigkeit und des Dampfgehalts in SWR-Brennelementen mit statistischen Analysenverfahren
Atomkernenergie Vol. 27, No. 4, 1976

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10. Degree of Availability of the Reports

Reports are available through
Institut für Kerntechnik
Elbestr. 38 A
D-3000 Hannover 21

Classification: 10.1		
Title 1 (Original Language): Früherkennung von Fehlern und Schäden (ATT 085 A - II.4.1., Jahresbericht A 76)		COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (English): Incipient Failure Detection		Project Leader: Dr. D. Wach
Initiated (Date):	Completed (Date):	
Status: continuing	Last Updating (Date): December 1976	

1. General Aim

The general aim of this research work is the development of measuring methods which permit the on-line monitoring of the mechanical status of main components in the primary system of a power reactor plant. Special attention is paid to the fact that already existent signals can be used for this purpose and that the need of additional sensors is kept to a minimum.

2. Particular Objectives

The following objectives are envisaged:

- On-line monitoring of the vibrational behaviour of reactor vessel internals and of the primary circuit
- Detection and identification of malfunctions inside the reactor vessel
- Detection and location of loose parts in the primary system.

3. Experimental Facilities and Program

Not relevant.

4. Project Status

4.1 Progress to Date

Basic investigations in the last years on several nuclear power plants verified the original concept that indirect measuring methods can be or have to be, respectively, used for reactor on-line monitoring systems surveying the vibrational behaviour of reactor internals and diagnosing anomalies or malfunctions within the primary system (loose parts, cavitation, leackage, blockage). Safety reasons and the inaccessibility of the inside of the pressurized system for additional detectors require efforts to gain the relevant information from sensors outside the vessel and pipes or via secondary effects in other physical parameters being already measured (e.g. internal vibrations in neutron or pressure noise signals). To solve the main problems of indirect measuring methods, that means identification and separation of the signal components (sources) being representative for the individual internal vibrations or reactor malfunctions, correlation techniques are applicated.

Vibration and accelerometer signals of sensors attached from outside to the surface of the pressurized system, dynamic pressure signals in the pipes and neutron noise of incore and out-of-core ionisation chambers have been already proved as appropriate information carriers for monitoring systems.

As far as reactor vibrations are concerned the development of suited theoretical models describing the coupled multi-mass system of the mechanical structures and their exciting forces is the most important task. Identification methods and modern filter algorithms are applicated for optimal estimations of model parameters. The verification of the structure vibration models is gained by extensive vibration measurements during hot functional tests without and with the core performed in close collaboration with the industry. During these preoperational measurements also the correlations of component vibrations inside and outside the pressure vessel are determined empirically. The measured data are used as a reference state of the reactor for the on-line monitoring.

A proto-type vibration monitoring system and a loose parts monitoring system have been installed in the NECKARWESTHEIM PWR (GKN). Extensive reference measurements were performed during hot functional tests and during power operation.

Control of reactor vibrations and detection of loose parts is the aim of a monitoring system which has been realized using accelerometers at the BRUNSBÜTTEL BWR (KKB). A number of calibration, reference and operational measurements has been performed in this year.

Special attention was paid to the improvement of the theoretical models developed so far and the detailed interpretation of the experimental data, the PSDs and coherences of the available noise and vibration signals mentioned above.

Similar investigations, i.e. data interpretation and model development have been performed for the sodium cooled test reactor KNK-I. In preparing the KNK-II test program the vibration behaviour of instrumentation tubes has been precalculated using the finite element program SAP IV.

With the aim of improved signal interpretation the program STAMPO has been developed. It determines the complex transfer matrix of a structure model from the given mass-, stiffness- and damping matrices of the mechanical system and calculates the PSDs und CPSDs of the vibration signals from this transfer matrix and the PSD matrix of the exciting functions. In contrary to deterministic programs one is able now to vary the degrees of the coherence of the exciting functions at the different structure components. Further on as an important point in the principle of the control, (i.e. for the interpretation of deviations of the actual pattern from the reference pattern) in an advanced version of STAMPO sensitivity studies of model parameters can be performed.

4.2 Essential Results

The interpretation and identification of relevant signal sources in the available information carriers could be essentially improved. As a consequence of the experimental results in GKN the vibration behaviour of the loops including the pendular eigenfrequencies of the steam generators and main coolant pumps has to be considered when interpreting the PSDs of the lid screw absolute displacements. Peak-"triples" belong to differences in the stiffnes (lengths) of the three loops. In the same frequency coherence peaks of the GKN neutron noise which are additional to the well-known core barrel attenuation noise could be related to bending modes of the fuel elements. Investigations of in-core neutron noise of the STADE reactor confirmed these results.

Typical pattern of continous and burst-type sound signals of a BWR at various operational conditions have been studied and evaluated at KKB. Sound propagation phenomena were investigated from artificial and operational impacts. A paper about operational experiences with the loose parts monitoring system has been prepared for the Reaktortagung 1977. At the same meeting also a paper about the state of the developments concerning vibration monitoring systems for PWRs will be presented. Publications about the systems and the experimental results are listed in refs. /1-6; 8/. The application of the identification program SYSIFA to the pendulum model and the tape-stored measuring signals of the PWR BIBLIS A which resulted in a verification of the model and a optimal estimation of the system parameters is documented in /7/.

5. Next Steps

With the proto-type vibration monitoring system at GKN long time behaviour, normal operational influences and signal variations during the first fuel cycle will be investigated systematically (e.g. increasing of the neutron noise PSDs corresponding to the increasing operation time). Studies are planned for the processing of the data and deriving of alarm criteria indicating an abnormal reactor state or malfunctions inside the primary system. It will be investigated whether methods of pattern recognition and learning codes for the reference functions can be involved to solve these problems. The experience in internal vibration monitoring as gained at PWRs will be applicated also to BWRs. In this sense the installed external accelerometers at the BRUNSBÜTTEL BWR will be used for loose parts detection as well as for longtime vibration control in the low frequency range. Accompanying model development will be done using finite element codes.

The work performed on the KNK facility will be continued with the aim to develop diagnostic systems for the SNR fast breeder reactor.

6. Relation with Other Projects

See annual report 1974.

7. Reference Documents

/1/
On-line-Überwachung zur Früherkennung von Fehlern und Schäden in Leichtwasserreaktoren
MRR 153, November 1975

/2/
B. Raible, J. Moravek, D. Wach
Ein Körperschall-Überwachungssystem zum Melden und Orten loser Teile
Compacts of Reaktortagung, Düsseldorf, März/April 1976

/3/
D. Wach
Korrelationsanalyse in der Meßtechnik von Kernkraftwerken
Lecture at Technische Akademie Esslingen, March 1976, published in: Korrelationstechnik - ein neuer Zweig der industriellen Betriebsmeßtechnik, Lexika-Verlag, Grafenau, 1976

/4/
D. Wach
Ein neues Meßmodell der Neutronenflußkreuzleistungsdichte zur Identifizierung von Rauschquellen in Leistungsreaktoren
Thesis, Technische Universität München, Juli 1976

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/5/
P. Panagopoulos
Korrelationsuntersuchungen der Rauschsignale von Neutronenfluß-Incoredetektoren in einem Druckwasserreaktor. Interner Bericht
MRR-I-72, August 1976

/6/
D. Wach
On-line-Schwingungsmessung am Reaktordruckbehälter des KKW Brunsbüttel, Referenzmessung während des Vollast-Probebetriebs am 22. Oktober 1976. Vertraulicher Bericht
MRR-V-18, Oktober 1976

/7/
E. Sädtler
Identifikation des Schwingungsverhaltens und Systemparameter-Estimation von Druck- und Kernbehälter des KKW Biblis A
MRR 165, Oktober 1976

/8/
R. Sunder, B. Österle
Modellentwicklung zum Schwingungsverhalten des Reaktordruckbehälters und seiner Einbauten am Kernkraftwerk Biblis A. Teil I: Analyse von Kennwertmessungen. Interner Bericht
MRR-I-61, Januar 1976

8. Degree of Availability

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

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

Ref. /4/ is available through
Technische Universität München, D-8000 München 2, Arcisstraße 21

Classification: 10.1	
Title 1 (Original Language): In-Kern Meßdatenerfassung und -verarbeitung mit dem sicherheitstechnischen Ziel der Früherkennung von Schäden in Leichtwasserreaktoren (RS 232 - II.4.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: TU Hannover
Title 2 (English): In-Core-Data Acquisition and Processing with the Aim of Early Fault Detection in Light Water Reactors	Project Leader: Prof.Dr.-Ing. D.Stegemann Institut für Kerntechnik
Initiated (Date): 1.12.1976 Status: Continuing Completed (Date): 31.12.1979 Last Updating (Date): December 1976	

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 system is to be developed, that allows an early detection of certain faults during reactor operation and may support the operators to avoid dangerous operating conditions.

Performance of this project and construction of the system is planned in a way, that at a later point of time a combined acquisition and evaluation of neutron vibration and body-sound measurements should be possible.

3. Research Program

Development of a data acquisition system, using an analog magnetic tape station, which has to store the signals of in-core sensors in case of significant changes in the behaviour of nuclear and thermohydraulic parameters in the reactor core.

Development of a data processing unit for the on-line analysis of

1.100

stochastic signals, for evaluation of the characteristic functions and driving the tape recorder. Off-line analysis of the recorded signals and computation of relevant nuclear and thermohydraulic data like power distribution, local steam bubble velocity and local steam void content. Theoretical and experimental investigations of the propagation of neutron flux fluctuations.

Improvement of self-powered detectors with defined neutron and gamma sensitivities in order to support the experimental and theoretical analysis concerning space dependant effects.

4. Experimental Facilities, Computer Codes

Data acquisition system, consisting of in-core sensors (fission chambers, self-powered detectors), self-compensating low-noise amplifiers (developed at the Institut für Kerntechnik), signal distribution and multiplexing unit and analog magnet tape recorder with 14 tracks. Analyzer which computes characteristic functions and values from the digitized input signals and decides about the storage on magnetic tape. Computer analysis of in-core data by Fourier transform, using partly the existing analysis methods and developing further program modules for special evaluation problems.

5. Progress to Date

Development of a differential, self-compensating low-noise amplifier prototype.

Procuring of the following units:

- I/O-Extender for connection of the peripheric devices to the data acquisition system
- Analog-to-Digital-Converter and Multiplexer
- Digital plotter for graphical presentation of results
- Analog magnetic tape recorder with 14 tracks.

Combination of all devices to a complex data collection and processing system.

Adjustment of the electronics and first tests with respect to the special requirements.

Development of a driver for the ADC.

6. Results

Construction of a prototype amplifier and test in laboratory, completion of the data acquisition hardware, completion of the driver for the ADC and adaption to the computer.

7. Next Steps

Test of the prototype amplifier under realistic conditions at KWL.
Development of a micro-program for Fourier analysis and of display programs for plotter and coloured video display. Completion of a driver-program for the video display system and the tape recorder.
Check of the applicability of the existing computer analysis programs.
Records of in-core signals on magnetic tape; using them as input data for development and check of computer programs.
Application of special self-powered detectors and investigations of the gamma sensitivity.

8. Relation with other Projects

9. References

10. Degree of Availability of the Reports

Classification: 10.2	
Title 1 (Original Language): Dynamik großer leichtwassergekühlter Reaktoren (ATT 085 A - II.1.2 , Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (English): Dynamics of Large Water Reactors	Project Leader: Prof. Dr. Birkhofer Dr. Frisch Dr. Werner
Initiated (Date): / Completed (Date): Status: continuing / Last Updating (Date): December 1976	

1. General Aim

The general aim of this project is the development of reactor and plant dynamic models and computer programs, and reactor and plant transient analysis with respect to safety problems in large water reactors.

2. Particular Objectives

One main objective of this research project is the development of three-dimensional models of large reactor cores in order to analyze local effects during power transients. This requires the development of new computational methods in order to solve the problems with sufficient accuracy within the limits of computer memory space and computation time.

Another objective is the analysis of reactor plant transients (system analysis, analysis of anticipated transients with and without scram), requiring complex models of the entire plant as well as core models and safety systems.

3. Experimental Facilities and Program

The performance of experiments is not part of the project, however, the results of experiments and especially of light water reactor start-up tests are used for model verification purposes.

4. Project Status

The code development was concentrated on four groups of codes:

QUABOX/ CUBBOX	A series of programs for space-time-kinetics simulation using a new coarse mesh method for space discretisation and an implicit matrix decomposition time integration method.
QUABOX/ CUBBOX-Fluid-dynamics	Space dependent fluid dynamics model using coarse mesh methods.
ALMOS	A series of plant models for BWRs with one-dimensional solution of the hydrodynamic equations and optional a point-kinetics model or a space-dependent solution of the neutron kinetics equations. The model includes all important plant components and the safety system.
ALMOD	A plant model for PWRs similar to the BWR model ALMOS.

4.1 Progress to Date

QUABOX/CUBBOX /1/: Several high order approximations, which also take into account mixed partial derivatives, have been incorporated as user options into the code.

QUABOX/CUBBOX-Fluiddynamics: The method used in QUABOX/CUBBOX is also applied to multidimensional fluiddynamics equations. A one-dimensional test version has been implemented and tested.

An improved version of ALMOS was completed, tested and transferred to the IRS for application in licensing. Further development of ALMOS is concentrated on the simulation of small leaks /2/. ALMOS application included a very detailed ATWS study /3, 4, 11/ and the analysis of space dependent feedback effects in LWRs during fast pressure transients /9/. For code verification experimental results (start-up test at the Würgassen plant) have been post-calculated /5/.

The basic version of the PWR plant model ALMOD has been completed and tested. Further code development is the extension of the model to two-phase flow simulation in the primary coolant system. In addition programming of a two-loop model of ALMOD has been started /6, 12/. ALMOD application included ATWS calculations for a detailed analysis report /3/.

4.2 Essential Results

QUABOX/CUBBOX: The most accurate CUBBOX version permits to calculate power distributions in PWRs and BWRs with errors of fuel assembly powers below 2 %, employing a computational mesh corresponding to the fuel assembly structure. Computing time (CDC 7600 equivalent) for the most accurate version is 0.85 ms per mesh and time step (or iteration step) /7, 8, 10; ANS Topical Meeting M&C Division, Tuscon, March 1977/. The method is internationally recognized as being substantially superior to other coarse mesh methods, e.g. Finite Element Methods.

QUABOX/CUBBOX-Fluiddynamics: One-dimensional test calculations of artificial wave propagation problems, employing the basic quadratic (QUABOX) method show the same gain of efficiency relative to Finite Difference methods, that has been found in neutron kinetics applications /ANS Topical Meeting, see above/.

With respect to the SWR code ALMOS the most important result is the good agreement of transients compared to start-up test measurements and to the measured results of a loss-of-heat sink incident at the Würgassen plant /5/. Results of an ATWS study are documented in detail in /3/. The analysis has shown, that ALMOS is a proper tool for ATWS analysis.

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Within the PWR plant model development the completion of ALMOD is the most important result. First applications have shown that severe transients can be simulated with reasonable computation time.

5. Next Steps

Implementation of a two- and three-dimensional version of QUABOX/CUBBOX-Fluiddynamics.

Development of ALMOS-L for the analysis of small leaks in BWRs.

Programming and testing of an extended ALMOD version (two-phase flow, two parallel coolant systems), and detailed model verification of ALMOD.

6. Relation with Other Projects

Investigations with similar objectives with respect to core simulation are carried out at KWU (RS 26, RS 178).

Results of experimental projects are used for code development and verifications (RS 64, RB 455).

7. Reference Documents

/1/
S. Langenbuch, W. Maurer, W. Werner
QUABOX/CUBBOX - Ein Grobgitterverfahren zur Lösung der Neutronendiffusionsgleichungen. Programmbeschreibung
MRR-P-22, September 1976

/2/
P. Peternell
Erweiterung des Anlagenmodells ALMOS zur Simulation von kleinen Lecks.
Interner Bericht
MRR-I-83, Dezember 1976

/3/
W. Ullrich, W. Frisch u.a.
Untersuchungen von Betriebsstörungen bei Versagen der Reaktorschnell-abschaltung (ATWS) und anderer ausgewählter Sicherheitseinrichtungen
MRR 163, IRS-W-22, September 1976

/4/
P. Peternell
Einsatz des SWR-Transientenmodells ALMOS am Beispiel einer ATWS-Analyse
MRR 157, Mai 1976

/5/
S. Langenbuch, P. Peternell
Nachrechnungen von gemessenen Transienten mit dem Anlagenmodell ALMOS.
Interner Bericht
MRR-I-82, November 1976

/6/
W. Heider
Hydrodynamisches Modell für die Simulation paralleler Primärkühlkreis-läufe in einem DWR. Interner Bericht
MRR-I-75, Dezember 1975

/7/
M.R. Wagner, H. Finnemann, R.R. Lee, D.A. Meneley, B. Michelsen, I. Misfeldt, D.R. Vondy, W. Werner
Multidimensional LWR Benchmark Problems
Trans. Am. Nucl. Soc., 23, 221 (1976)

/8/
W. Werner, H. Finnemann, S. Langenbuch
Two- and Three-Dimensional BWR Kinetics Benchmark Problem
Trans. Am. Nucl. Soc., 23, 215 (1976)

/9/
W. Frisch, S. Langenbuch, P. Peternell
Untersuchungen von thermohydraulischen Rückwirkungseffekten in einem SWR
Atomkernenergie, September/Oktober 1976

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/10/
H. Finnemann, W. Frisch, J. Lockau, W. Werner
The Application of Advanced Physics. Models in LWR Safety Analysis
ANS Meeting, Washington, 1976

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S. Beliczey, W. Frisch, P. Peternell
Vergleichsrechnungen mit den SWR-Transientenmodellen ALMOS und COSTAX-LOOP. Interner Bericht
MRR-I-64, IRS-AB-271, April 1976

/12/
R. Meißner
Druckhaltermodell zur Simulation von Betriebsstörungen in Druckwasserreaktoranlagen
MRR 154, Januar 1976

8. Degree of Availability

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

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

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Classification 10.2

Title 1 Statisk Reaktorfysik	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 Development of calculation methods for static and quasistatic reactor physics in light water reactors	Project leader Hans Neltrup
Initiated 1968 Completed Status Last updating progressing	Scientists C.F. Højerup G.K. Kristiansen L. Mortensen H. Neltrup T. Petersen

1. General aim

Provision of static and quasistatic reactor physics information relevant for safety assesment 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 burn up,

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, flux synthesis and nodal theory for the reactor core completed.

d) 3D flux synthesis coupled to hydraulic channel calculation for BWR including burn up completed.

2. Essentiel Results

Reactor physics code system verified in calculation on Yankee Rowe, Connecticut Yankee and Dresden.

5. Next Steps

1. Complete 3D flux synthesis with multichannel hydraulics for PWR (in progress)
2. Test the system further against BWR measurements - to the extent they become available.

6. Relation with other projects

Provision of cross-sections and parameters for dynamics projects.

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.

7. Degree of availability

Partly available, partly available on exchange basis.

Classification: 10.3

Title 1 (Original Language):	COUNTRY: BRD
Mischungseffekte bei parallel durchströmten Kanälen (RS 81 - II.1.3 , Jahresbericht A 76)	SPONSOR: BMFT
	ORGANIZATION: Euratom, Ispra
Title 2 (English): Mixing Effects in Parallel Channels with Water Two-Phase Flow	**Project Leader:** H. Herkenrath W. Hufschmidt
Initiated (Date): 1.1.1975 — **Completed (Date):** 31.12.1977	
Status: Continuing — **Last Updating (Date):** December 1976	

1. General aim

The investigation of the interaction and mixing effects between adjacent subchannels of a fuel rod bundle cooled by boiling water is aimed at providing experimental data on the local distribution of mass-flow, enthalpy and void-fraction so as to achieve a better thermohydraulic description and more precise DNB margin for a BWR 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 steady-state and the transient conditions will be studied later.

2. Particular objectives

The mixing programme was started in the frame of collaboration contracts between CNEN (Rome) and BMFT (Bonn) on the one side and Euratom on the other. These 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 will be extended to a pressure of 160 barsusing 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 various preparations for the 16-rod bundle tests in BOWAL.

3. Research programme

Experimental investigation of subchannel interactions in two-phase flow conditions (mixing).

The original programme provided the following studies:

- Steady state measurements of the subchannel interaction in two-phase flow conditions with a two-channel test section of different geometries for fundamental studies and with a 16-rod bundle test-section in BWR-geometry
- Investigation of the mixing phenomena in transient conditions for partial depressurization with the above mentioned test-sections.

Due to personnel reasons in the JRC Ispra the fundamental studies with the two-channel test-section have to be postponed and a new programme has been set up concentrating on the investigations with 16-rod clusters. The fundamental studies will be resumed only in case of necessity for the interpretation of the test results with the bundle experiments.

The actual programme covers:

- steady state measurement of mixing effect with a 16-rod cluster in BWR-geometries (70 bar)
- mixing studies in transient conditions with 16-rod clusters in BWR- (70 bar) and PWR- geometries (160 bar).

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 [2] and are schematically shown in figures 1 and 2.

5. Progress to date and results

The activity in 1976 has been concentrated on the completion of the modifications for BOWAL and PRIL and the operational tests for the mixing experiments with the 16-rod cluster test-section in BWR-geometry (70 bars). These operational tests have shown that different modifications, especially for the measurement device, are necessary: The heat-exchanger for the secondary water flow of the condensers (see Fig. 1, supplementary heat-exchanger) showed a higher cooling performance than expected and hence a bypass with a regulation valve has been installed.

The preliminary tests with the isokinetic measurement technique [3] at the outlet of the test-section yielded a sufficient sensitiveness of the BARTON-cells to be used for the "zero-measurement" of the static pressures in the outlet cross-section (± 1 mm W.G. are reached by supplementary calibration in confront to the guaranteed values of the furnisher of ± 5 mm W.G.). The profile indicators

foreseen for the signals from the pressure transducers, however, have to be replaced by more precise digital instruments. Furthermore, the motor regulation valves in the sampling pipes from the subchannels of the test-section were less sensitive than expected and the regulation spindles had to be replaced after different calibration tests (Fig. 3).
The mean outlet temperatures at the primary and secondary side of the condensers for the heat-balance measurements (for outlet-enthalpy determination in the sampled subchannels) showed remarkable deviations (about ± 5%). By means of the installation of appropriately developed flow mixing devices (Fig. 4) the deviations could be decreased to a value of about ± 1% corresponding to the tolerances of the thermocouples used. These thermocouples are embedded in silver crosses as described in [2] and [3] to equilibrate temperature variations and fluctuations in the connection tubes of the condensers. As these silver crosses are manufactured by a cast-process small cavities (Fig. 5) caused tightness-troubles and had to be eliminated.
During 1976 the installation of the calibration device in PRIL loop (Fig. 2) has been completed and now an installation is available for the necessary continuous control and calibration of the X-ray void meters under real pressure and temperature conditions (up to 160 bars and 347°C). By means of two preheater tubes (300 kW power-input, each) and with an appropriate mixing chamber different qualities and flow-patterns of the steam-water mixture are adjustable. In a pressure housing after this mixing chamber the annular probe voidmeter [4] for the qualitative determination of the flow pattern will be tested. In this probe at the outlet a mixing device is incorporated providing an homogeneously distributed two-phase mixture over the tube cross-section (i.d. 16mm). After this device the X-ray voidmeters [5] are controlled and calibrated with the same arrangement used in the sampling pipes of BOWAL [3]. Due to administrative difficulties the originally foreseen X-ray tubes of TUBIX, Paris (30 kV 600 W) can not be used for the boiling mixing studies and new types (Siemens, Karlsruhe with an energy up to 60 kV and a power of 3000 W) have been ordered and have to be tested and calibrated.

7. Next steps

- Steady state measurements with a 16-rod cluster test section in BWR geometry (70 bar)
- Development and construction of the 16-rod bundle in PWR geometry (160 bars) together with CISE, Milan
- Calibration of the new X-ray void meters on the optical bank as well as under real operational conditions in a calibration loop (PRIL).

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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., Mörk-Mörkenstein, P.
The pressurized and boiling water loop of the Technology Division at Ispra
EUR 4683 e (1971)

[2] 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 JRC, Ispra
EUR (to be published in 1977)

[3] 1975 Safety Programme Progress Report of the Ispra Establishment, Sheet 2.4., pp 73-83, ACS - 66 e

[4] Herzberger, P., Hufschmidt, W.
Bestimmung des Dampfgehaltes und der Strömungsform eines Zweiphasengemisches in konzentrischen Ringspalten
EUR 5077 d (1974)

[5] Herzberger, P., Hufschmidt, W., Wind, F.
Die Verwendung von Gamma- oder Röntgenstrahlen zur Messung des Dampfgehaltes in Zweiphasenströmungen
EUR 5415 d (1975)

10. Degree of availability of the reports
Reports mentioned in 9, except ref. 3, are not classified and are on sale at the office for official publications of the European Communities, 37, rue Glesener, Luxembourg.

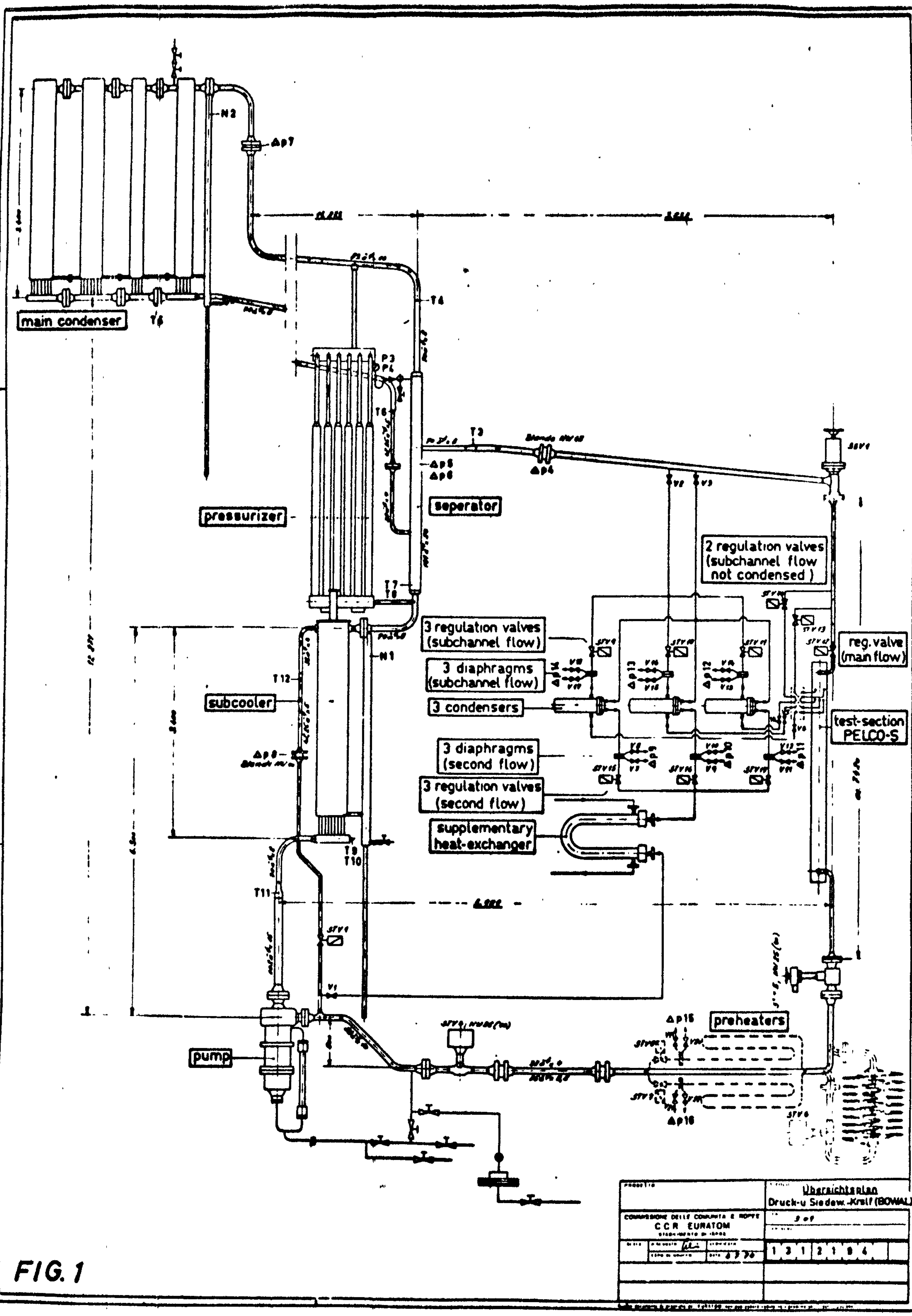
main condenser
N2
Δp7
T4
P3
P4
T5
T3
Δp4
Δp5
Δp6
pressurizer
seperator
T7
T8
N1
T12
subcooler
Δp8
T9
T10
T11
2 regulation valves (subchannel flow not condensed)
3 regulation valves (subchannel flow)
3 diaphragms (subchannel flow)
3 condensers
3 diaphragms (second flow)
3 regulation valves (second flow)
supplementary heat-exchanger
reg. valve (main flow)
test-section PELCO-S
preheaters
Δp15
Δp16
pump
V1
Übersichtsplan
Druck-u Siedew.-Kreisl (BOWAL)
C C R EURATOM
1 3 1 2 1 8 4

FIG. 1

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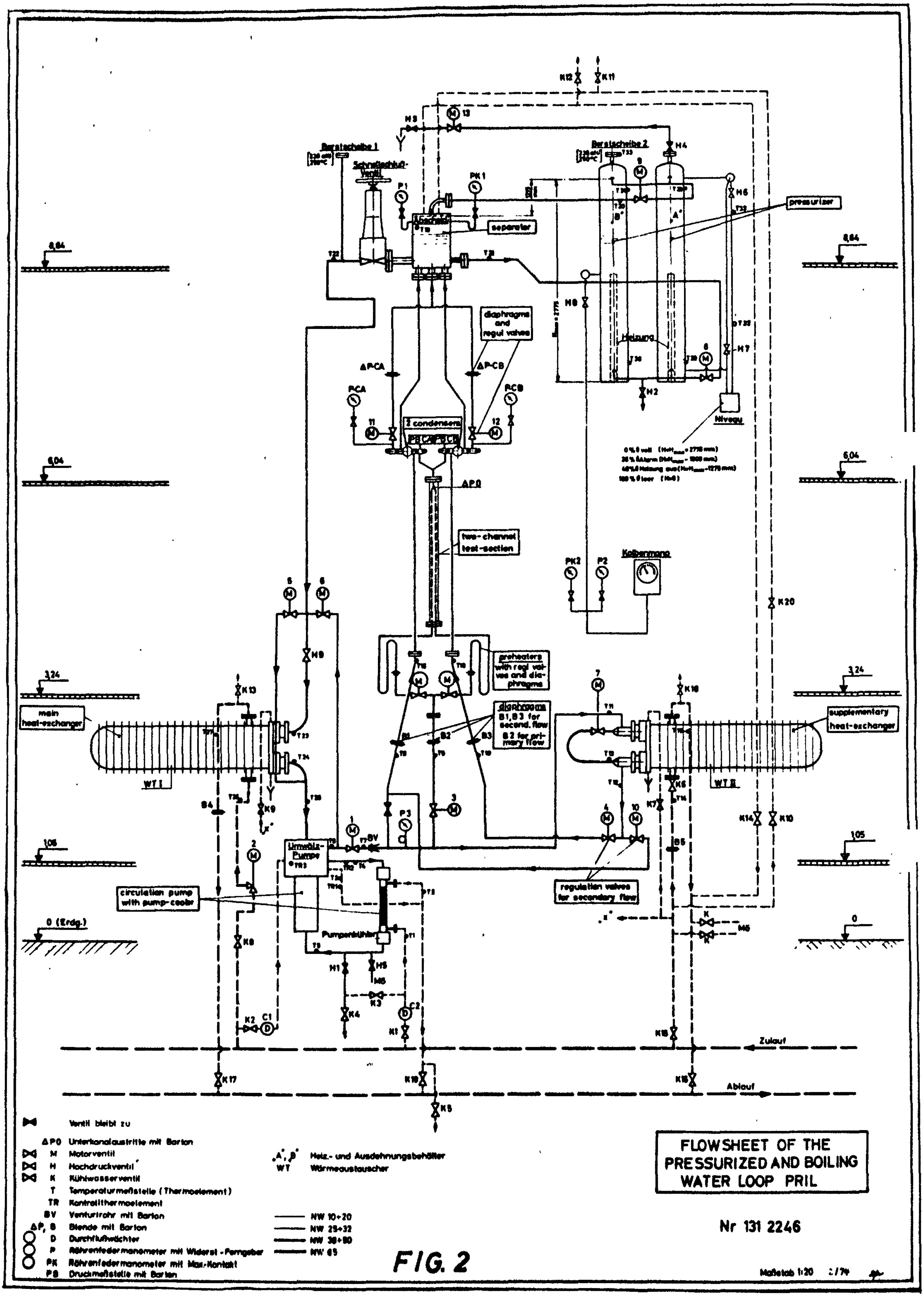
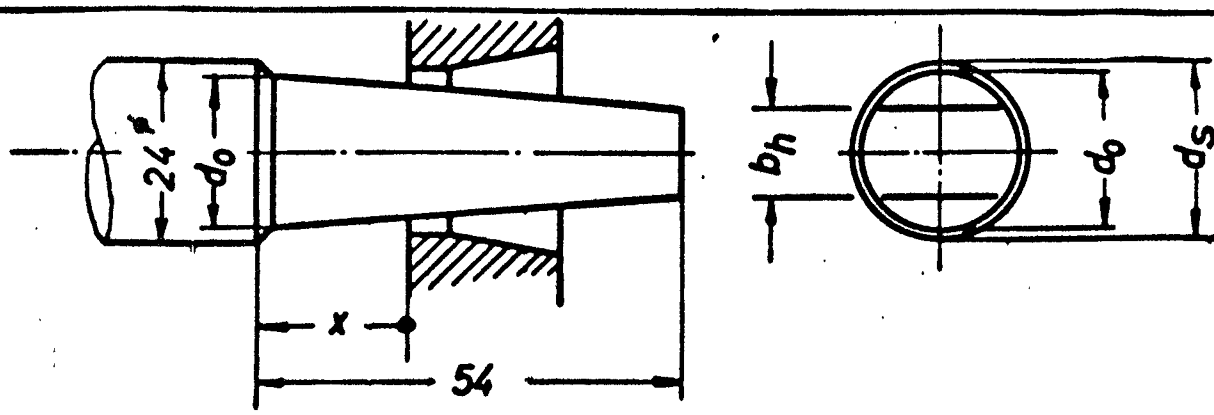
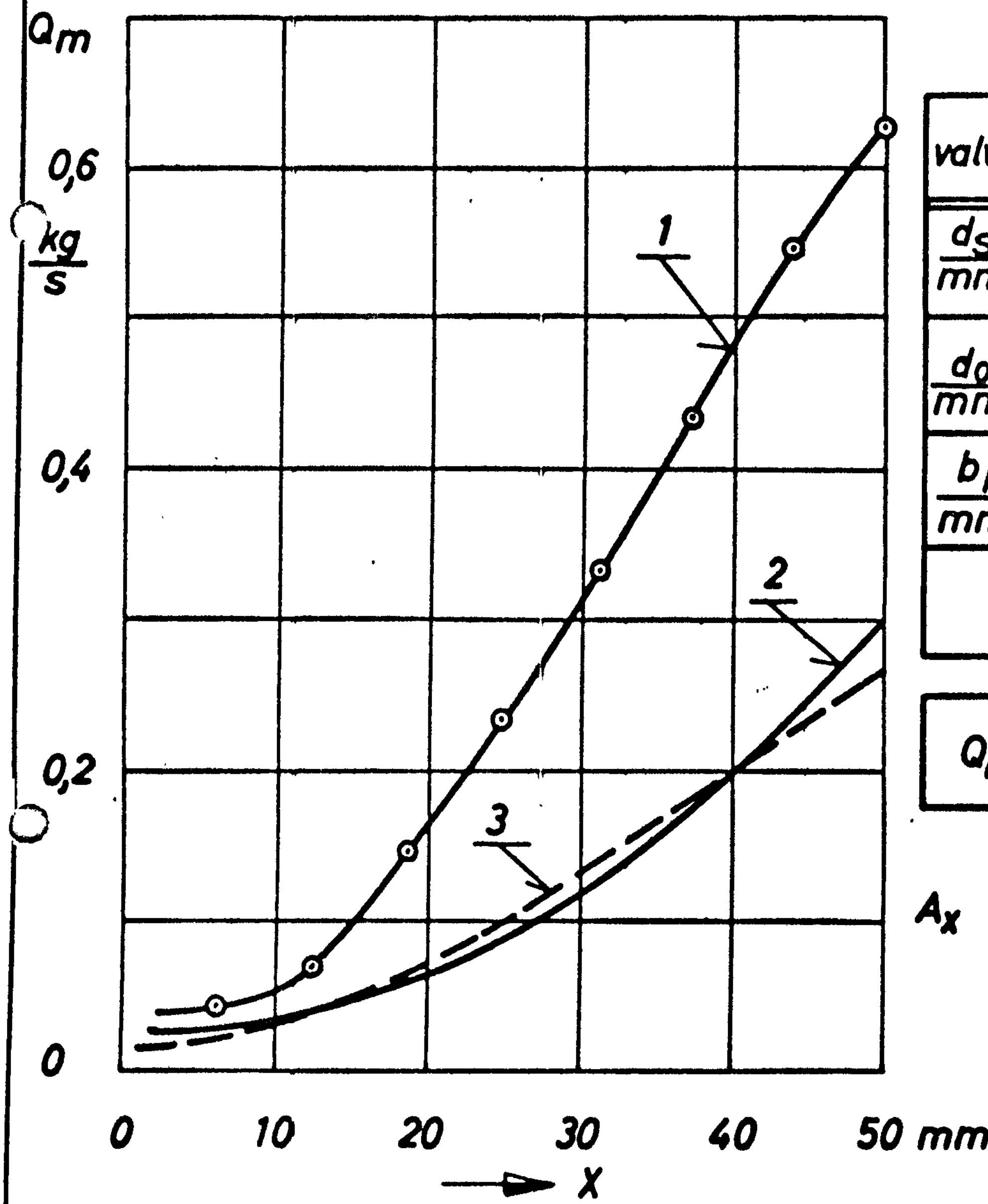


FLOWSHEET OF THE PRESSURIZED AND BOILING WATER LOOP PRIL
Nr 131 2246
Maßstab 1:20
pressurizer
separator
diaphragms and regul valves
2 condensers
two-channel test-section
preheaters with regul valves and diaphragms
diaphragms B1, B3 for second. flow B2 for primary flow
main heat-exchanger
supplementary heat-exchanger
WT I
WT II
circulation pump with pump-cooler
Umwälz-Pumpe
Pumpenkühler
regulation valves for secondary flow
Kolbenmano
Niveau
Heizung
Berstscheibe 1
Berstscheibe 2
Schnellschluß-Ventil
Zulauf
Ablauf
8,64
6,04
3,24
1,05
0 (Erdg.)
Ventil bleibt zu
ΔP0 Unterkanalaustritte mit Barton
M Motorventil
H Hochdruckventil
K Kühlwasserventil
T Temperaturmeßstelle (Thermoelement)
TR Kontrollthermoelement
BV Venturirohr mit Barton
ΔP, B Blende mit Barton
D Durchflußwächter
P Röhrenfedermanometer mit Widerst.-Ferngeber
PK Röhrenfedermanometer mit Max-Kontakt
PB Druckmeßstelle mit Barton
A°, B° Heiz- und Ausdehnungsbehälter
WT Wärmeaustauscher
NW 10÷20
NW 25÷32
NW 38÷80
NW 65

FIG. 2



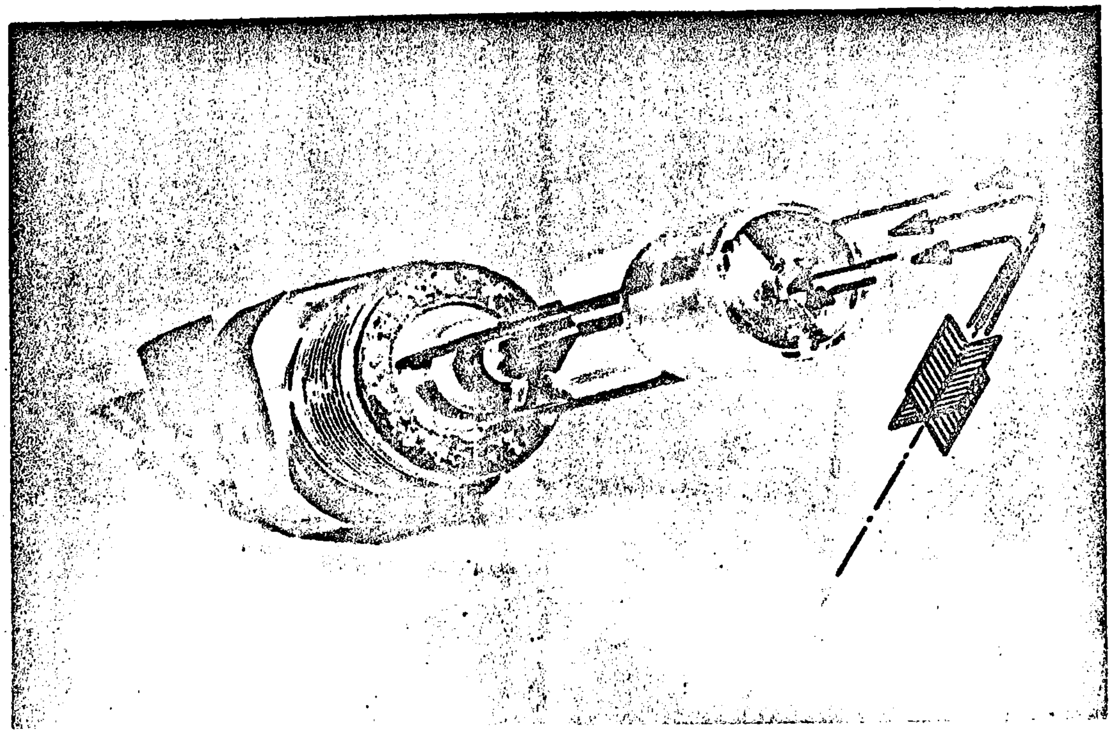


valve	1	2	3
$\frac{d_s}{mm}$	20,0	20,0	12,5
$\frac{d_o}{mm}$	19,8	19,8	12,3
$\frac{b_h}{mm}$	11,8	15,1	7,2
	measured	calculated	

$$Q_m = K_x \cdot A_x \sqrt{\varrho \cdot \Delta p}$$

A_x = flow-cross section

FIG. 3 Mass-flow vs. stroke for regulation valves in the sampling pipes of BOWAL (valve type METALGHISA NW 20 , ND 400) for a pressure drop of 0,5 at and a density of 1000 kg/m³



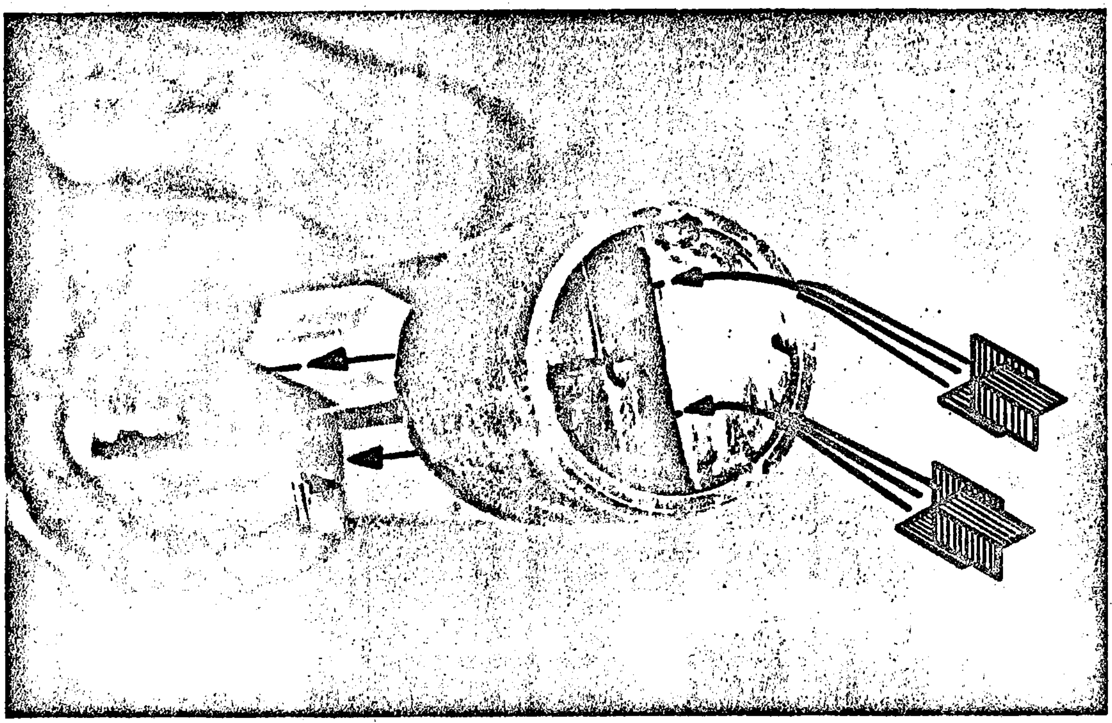


FIG. 4 Flow mixing chamber and silver cross for the mean outlet temperatures of the condensers for the heat-balance measurements (outer diameter about 30mm, 4 turn-round sheets.)

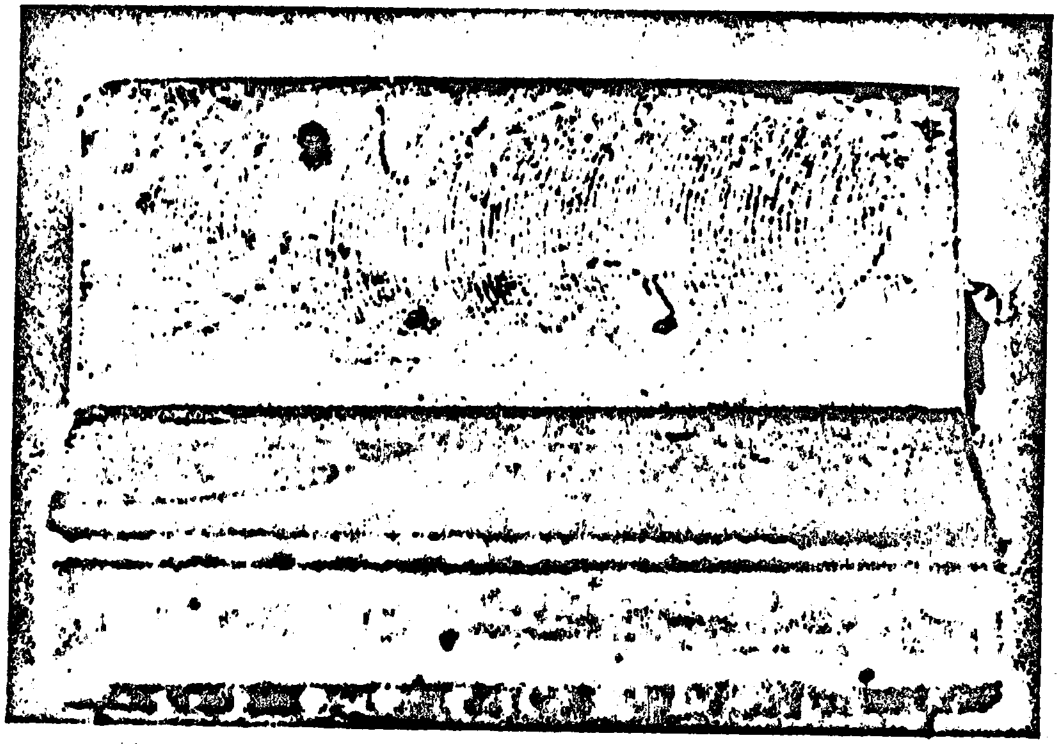


FIG. 5 Cavity effects in the silver crosses for the mean temperatures of the condensers (length 30 mm, diameter about 30 mm, thickness of wing 6 mm.)

Classification 10.3

Title 1	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 SDS, a thermohydraulic subchannel programme for steady-state analysis of water-reactors	Project leader A. Olsen
Initiated: May 1970 Completed: Autumn 1974 Status: Code made, verification continues. Last updating: Autumn 1974	Scientists: 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 phenomenas 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) annulus geometries.

4. Project status

1. Progress to data

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

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fuel-box or core-basis. (No limit on number of subchannels, so far up to 400 has been tested).

2. Essential Results

On the modelling side a substantial improvement have 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 present Danish project is an extension of the work.

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, October 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
Void fraction experiments in 8x8 BWR/6 test sections	Sponsor NUCLITAL Organisation NUCLITAL/CISE
Date initiated January 1977 Date completed June 1979 Last updating May 1977	Project Leader 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 Sponsor : CNEN Organisation : CISE
Thermohydraulic behaviour of a CIRENE power channel during reactor start-up	
Date initiated June 1975 Date completed December 1976 Last updating 1977	Project Leader 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 **Organisation** NUCLITAL/CISE
Date initiated January 1977 Date completed June 1979 Last updating May 1977	**Project Leader** 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 pro va 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

Classification
10.3

Title 1 Burnout experiments in BWR round tubes	**Country** Italy **Sponsor** CNEN **Organisation** CNEN-CISE
Title 2	
Initiated 1.1.75 **Completed** 30/9/75 **Status** planning **Last updating** Dec. 74	**Project leaders** 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 MLit. / Personnel: 40MM

Title 1 (Original language)	Classification
Esperienze di burnout su elementi di combustibile per BWR	10.3
Title 2 (English)	**Country** : ITALY
Burn-out experiments in BWR fuel elements	**Sponsor** : CNEN **Organisation** : CNEN-CISE
Date initiated 1972 **Date completed** August 1977 **Last updating** April 1977	**Project Leader** G. Basso (CNEN) 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.

1753

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 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 thermo-hydraulics 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

31.719

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- 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).

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ENERGIEONDERZOEK CENTRUM NEDERLAND		CLASSIFICATION: 10.3
TITLE: Ontwikkeling rekenprogramma VITESSE Stationaire temperatuur en snelheidsverdelingen in onsamendrukbare fluïda		COUNTRY: NETHERLANDS. SPONSOR: ECN ORGANIZATION: ECN
TITLE: (ENGLISH LANGUAGE): Development computer code VITESSE Sationary temperature and velocity distribution in incompressible fluids		PROJECTLEADER: Slagter, W.
INITIATED: February 1975	LAST UPDATING: April 1977	SCIENTISTS: Roodbergen, H. Vonka, V. Spreeuw, E.
STATUS: progressing	COMPLETED: 1977	

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.

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.

Degree of availability

Upon mutual agreement at ECN-Petten

Budget: -

Personnel: -

2137

RCN - project D 2	CLASSIFICATION: 10.3
TITLE: Experimenteel onderzoek naar mixing en de invloed van rooster op mixing in een 36-staafs bundel.	COUNTRY: NETHERLANDS SPONSOR: R.C.N. ORGANIZATION: R.C.N.
TITLE (ENGLISH LANGUAGE): Mixing experiments in 36-rods bundle and the influence of supporting grids on mixing.	PROJECTLEADER: van der Molen, S.B.
INITIATED: 1973 — COMPLETED: December 1977 STATUS: in progress — LAST UPDATING: July 1976	SCIENTISTS: Heil, J.A. Hoogland, H. Warmenhoven, A.

1. General aim

Experimental investigation of the mixing phenomena in a bundle geometry.

2. Particular objective

Experimental study of the heat and mass exchange between the subchannels in a bundle in which a nonhomogeneous heat distribution is generated.

3. Experimental facilities

Testloop for low-pressure experiments.

4. Project States

A 6x6 bundle is provided with the 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 non-homogeneous heatflux distribution due to the heatproduction by only 6 rods of the total of 36.

5. Next steps

The results of the temperature measurements correlation techniques will be used to study the heat exchange between the subchannels based on values of the cross-correlation functions between different thermocouple signals.

1139

RCN - project D 3	CLASSIFICATION: 10.3
TITLE: Burnout onderzoek aan 9-staafs bundels.	COUNTRY: NETHERLANDS SPONSOR: R.C.N. ORGANIZATION: R.C.N.
TITLE (ENGLISH LANGUAGE): Burnout experiments on 9-rod bundles.	PROJECTLEADER: van der Molen, S.B.
INITIATED: 1972 COMPLETED: Spring 1977 STATUS: nearly finished LAST UPDATING: July 1976	SCIENTISTS: Hoogland, H. Middleton, D.W. Werner, D.

1. General aim

Study of the burnout phenomena in bundles.

2. 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.

3. Experimental facilities

A testloop for pressures up to 100 bar is available and the power supply of 700 kw gives the opportunity of the high heat fluxes needed for the burnout investigations. As burnout detection method, the Wheatstone bridge priciple is used.

4. 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 the 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 suppording 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, where-as the radical heat flux distribution can be varied.

5. Next steps

Void distribution calculations with COBRA-IIIC

6. Reference documents

R.C.N. Technical memo 1063-020 (in Dutch)

1144.

RCN - project D 1	CLASSIFICATION: 10.3
TITLE: Belsnelheden en dampfracties in bundels.	COUNTRY: NETHERLANDS SPONSOR: R.C.N. ORGANIZATION: RCN
TITLE (ENGLISH LANGUAGE): Slip measurement and bubble size distribution in bundles.	PROJECTLEADER: Heil, J.A.
INITIATED: 1973 — COMPLETED: Spring 1977 STATUS: nearly finished — LAST UPDATING: July 1976.	SCIENTISTS: Winters, L.

1. General aim

Study of the influence of slip between the two-phases of the cooling medium in nuclear reactors.

2. Particular objective

Investigation of the velocity of bubbles as a function of size, the bubble size distribution and the bubble concentration over the cross-section of the bundle.

3. Experimental facilities

Testloop for low pressure experiments.

4. Project States

Experiments determining the velocity of the bubbles and void fraction have been carried out with a high speed film camera. From the experiments the bubble size distribution and the bubble velocity as a function of size were determined.

5. Next Steps

Comparing the results with existing slip correlations.

6. Reference documents

a) R.C.N. Technical memo 1063 - 013
b) R.C.N. Technical memo 1063 - 016
c) R.C.N. Technical memo 1063 - 017
} in Dutch

	Classification 10.3
Title 1 DRYOUT IN BOILING CLUSTERS	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION AEE WINFRITH
Title 2	Project Leader J OBERTELLI
Initiated 1966 Completed : 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).

Classification: 10.4	
Title 1 (Original Language): Körperschallmessungen an Reaktordruckbehältern und am Primärkreislauf von Kernkraftwerken (RS 97 A - II.3.5, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Allianz, Ismaning
Title 2 (English): Solid-Borne Sound Measuring on Reactor Pressure Vessels and on the Primary Circuit of the Cooling System	Project Leader: Raible
Initiated (Date): 15.4.1975 Completed (Date): 14.4.1977 Status: continuing Last Updating (Date): Dec. 1976	

1. General aim

The object of this project 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 the work are:

a) Study of parameters on an experimental loop for a Leak Detection and Monitoring System based on acoustic emission.

b) Development of an enlarged Loose Parts Monitoring System (LMPS), with which loose and loosened parts can be detected and located.

3. Research program

The research program for the study of parameters for the leak monitoring system embraces a total of eighty experiments:

Variation of the following parameters was carried out:

Four types of leaks	leaking flange, leaking valve, crack, nozzle troat
Two leakage rates:	24o l/h, 24 l/h
Three enthalpies:	steam, water/steam, water
Unterground noise:	yes/no

Investigations were carried out to determine in which manner different types of leaks are discernible in solid-borne sound (noise analysis, correlation, power spectra, delay-time measuring).

4.1 Experimental facilities

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 Computer codes

The Loose Parts Monitoring System was enlarged by a computer code, which computs the ranges of the sources of the burst-signals after input of the delay times between the different measuring points.

For analysing the nois 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

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 an 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.

6. Results

The analyses have not yet been completed.

7. Next steps

In the coming weeks, the still pending analyses will be carried out. Above all, the practicability of the triangulation of leakages using correlation analysis will be investigated.

8. Relation with other projects

Battelle-Institut Frankfurt:

"Pre-Experiments for Leak Monitoring by means of Acoustic Emission Analysis".

9. References

"Körperschallmessungen an Reaktordruckbehältern und am Primärkreislauf von Kernkraftwerken" in publication series.

Quarterly research reports:

IRS-F-27
IRS-F-28
IRS-F-3o
IRS F-31
IRS-F-33
Annual report 1975: IRS-F-29

1o. Degree of availability of the reports

All reports specified under 9 are available at the "Institut für Reaktorsicherheit", Glockengasse 2, 5000 Cologne 1.

Classification: 10.4	
Title 1 (Original Language): Modellmäßige Behandlung der Schwingungen von Kerneinbauten in Leichtwasserreaktoren (RS 0066A - II.4.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: University Erlangen
Title 2 (English): Model calculation of vibrating core components in light water reactors	Project Leader: Prof.Dr.-Ing. E. Klapp Lehrstuhl für Apparatetechnik und Anlagenbau D-8520 Erlangen Erwin-Rommel-Str.1
Initiated (Date): March 1972 Status: Continuing	Completed (Date): Dec. 1977 Last Updating (Date): December 1976

I. General aim.

Apparatus and plants using large quantities of fluids are often damaged by flow-induced 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.

II. 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 coused 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 ist always a selective amplification of the load energy about dominant frequency bands associated with an increase of spatial correlation. Without feedback-mechanism the load energy in a turbulent flow, which causes the mechanical structure to vibrate, has a statistical distribution being unable to excite the structure systematically.

III. Research Program

Figure 1 gives a diagrammatic representation of the investigation program: Using only the hydrodynamic analogy of the flow, the structure of the model should be rigid, because it does not correspond in its elastic properties to the actual installation. There are dimensionsless numbers enabling a transfer of measurement values taken at the model to the actual reactor-pressure-vessel. In this model it is possible to investigate fluid-dynamic and fluid-resonant feedback - whenever they exist in the hydrodynamic system.
If one takes into account not only the hydrodynamics but also the kinetics of the core components one obtains a hydroelastic similar model. This model allows the investigation of fluid-elastic feedback between structure and fluid. The difficulty of this model ist that six dimensionsless numbers and the boundary conditions of the flow and the structure must be fulfilled.
It is provided that the measurements taken at the model will be compared with those of the original reactor.

IV. Experimental facilities; computer codes

A.) Experiments conducted on the hydrodynamically similar model

A geometric scale-model of the pressure vessel with core components is constructed. The experiments are conducted in open circuit with air as flow medium. The points at which measurements are taken are where the flow medium enters and leaves the model, in the ring gap between the pressure vessel and the core barrel, on the lower core support and on the control rod guide assemblies. Dependent on various parameters the pressure fluctuations on the core components are measured by means of pressure transducers.

B.) Experiments conducted on the hydroelastically similar model

This model corresponds not just in its hydrodynamic similarity but even in its elastic features to the actual installation. The experiments conducted in this model are carried out in closed circulation with water as the flow medium. Three main groups of variables will be measured:

1) vibrations, i.e. displacements and accelerations, of core components
2) strains of the structure due to vibrations and
3) pulsations of the hydrodynamic pressure, which induce core component vibrations.

C.) Computer codes

1) Computerprogram for the analysis of stochastic signals (FORTRAN IV)
2) Computerprogram based on the "lumped-mass" technique for the simulation of the dynamic behaviour of reactor internals by means of beam structures. (FORTRAN IV)
3) Finite element program (SAP IV, FORTRAN IV)

V. Progress to date

a) Experimental setup of the hydro-elastic model to investigate the coupling between structure and fluid. The flow medium being water.
b) Completion of the core assembly and experiments with the hydrodynamic model. The flow medium being air.
c) Analysis of original data from a nuclear power station.
d) Theoretical work in the field of flow-induced structural vibrations.

VI. Results

Essential results of the experiments with the hydrodynamic model will be published comprehensively in 1977. Figure 2, for example, shows typical power spectra of dimensionless pressure pulsations (y), which were measured in different positions in the ring gap between core barrel and pressure vessel.

All spectra thus measured show a more or less pronounced concentration of the energy around the some dominant frequencies. Since the frequency content varies only slightly when changing the volume flow, the geometry and the flow medium obviously are the essential parameters as

shown in figure 3, which shows a spectrum where the geometry of the gap between core barrel and pressure vessel has been changed by a ram ledge (Stauleiste). It is obvious that this variation has a great influence on the frequency content of the pressure signals.
During 1976 the core barrel and the pressure vessel with which the hydro-elastic model is to be carried out has been completed. In manufacturing the core assembly of the model very detailed designung of the components was necessary. The instrumentation has been initiated.

VII. Next steps

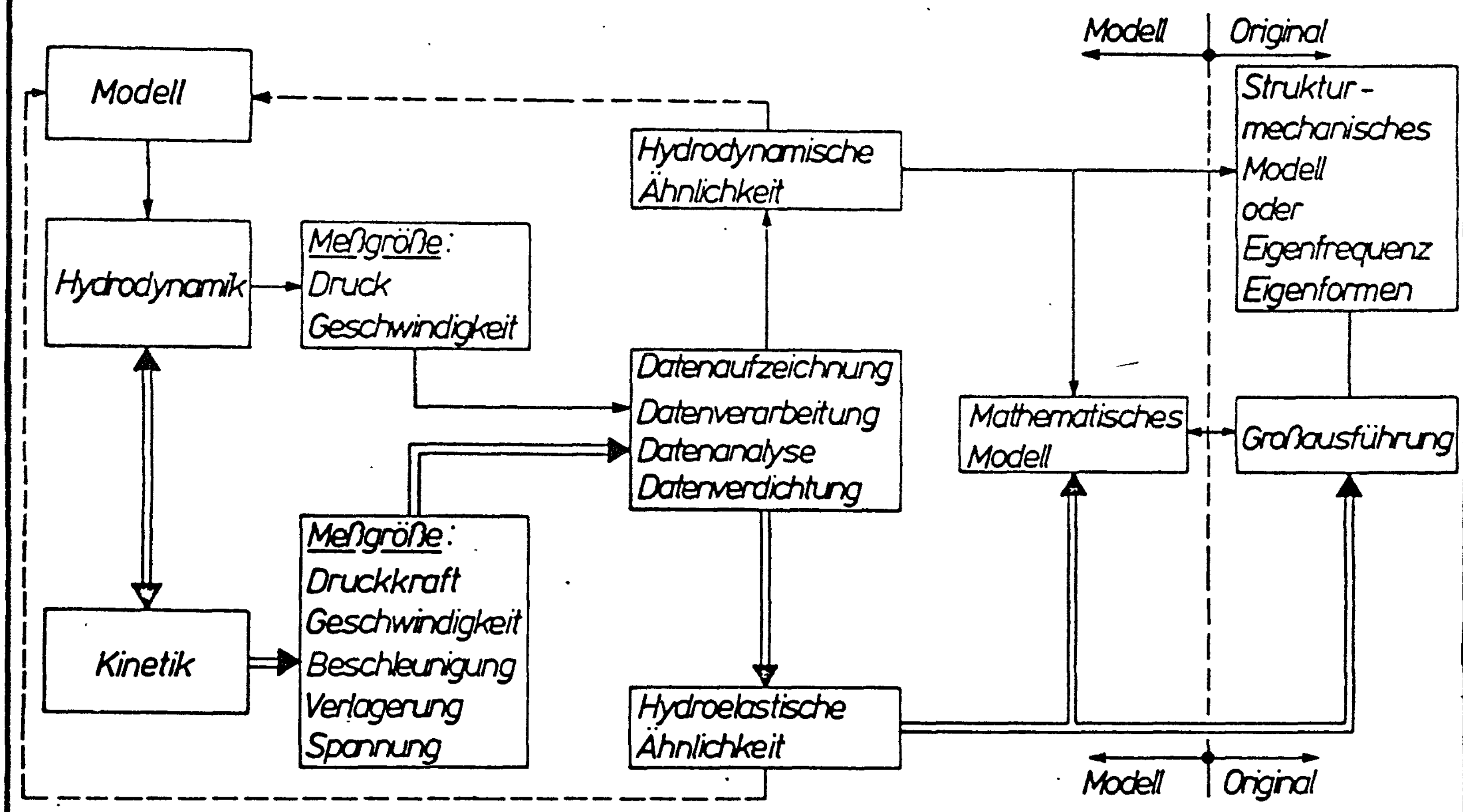
a) Preliminary conclusion of the experiments with air as the flow medium
b) Experimental determination of the mechanical properties of the hydro-elastic model, i.e. eigenfrequencies, damping etc.
c) Measurements with the hydro-elastic model.
d) Development of a semi-empirical model to calculate the flow induced vibrations of the core components of PWR.

VIII. Relation with other projects

See this report for 1974

IX. References

Annual reports for the IRS/BMFT (German)



UEN AA	Übersicht: Forschungsvorhaben RS 66 Modellmäßige Behandlung d. Schwingungen v. Kerneinbauten i. Druckwasserreaktoren	RS 66

Abb. 1

1153

1154

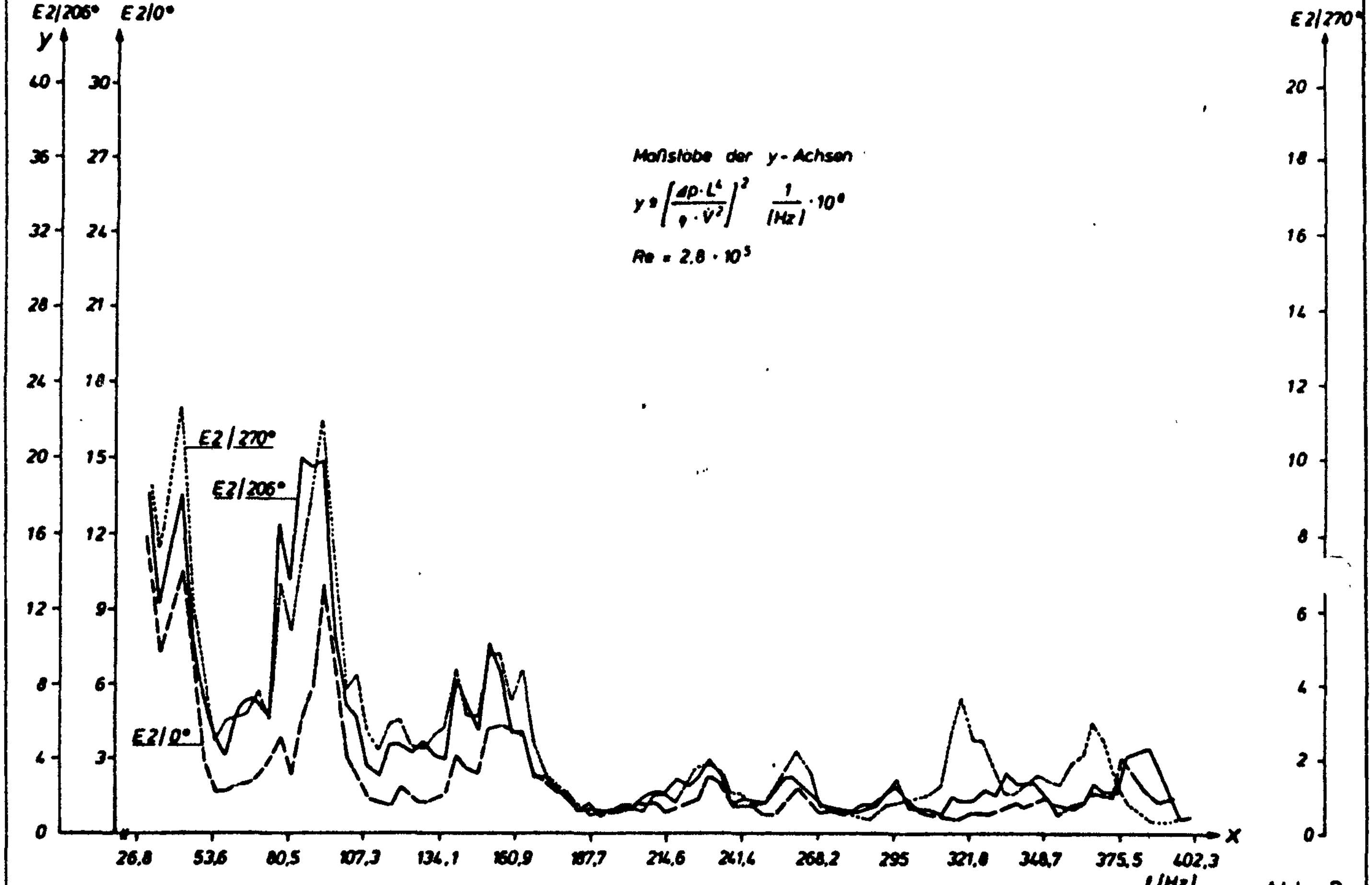


Abb. 2

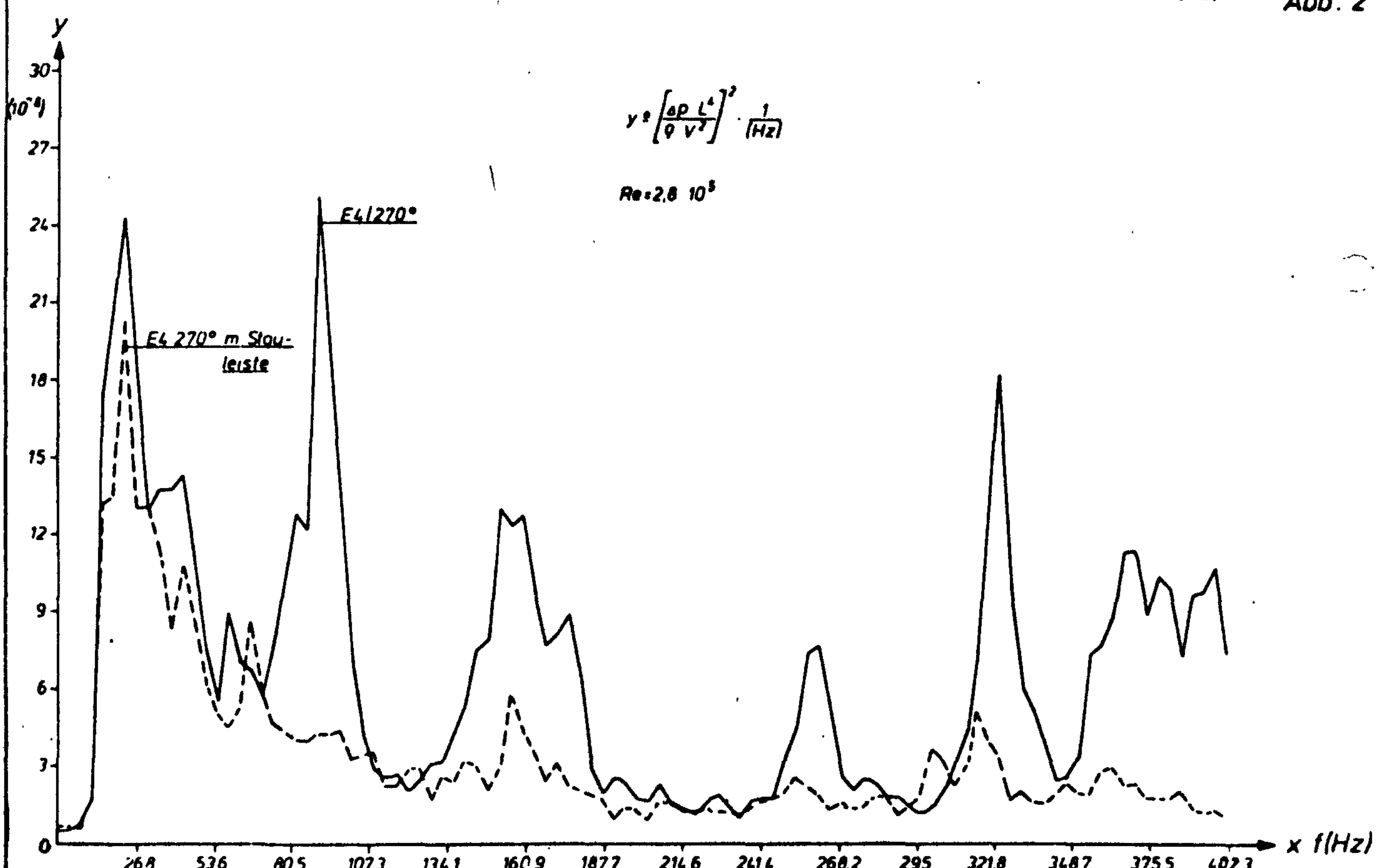


Abb. 3

1155

Classification: 10.4

Title 1 (Original Language):		COUNTRY: BRD
Experimentelle Untersuchungen zur Absicherung von Druckbehältereinbauten gegen strömungsinduzierte Schwingungen beim Siedewasserreaktor (RS 218 - II.4.1, Jahresbericht A 76)		SPONSOR: BMFT
		ORGANIZATION: KWU, Karlstein
Title 2 (English): Experimental Investigation of Protection of Pressure Vessel Internals Against Flow-Induced Vibrations in BWRs		Project Leader: Dr. Simon
Initiated (Date): 1. 9. 76	Completed (Date): 31. 5. 77	
Status: Continuing	Last Updating (Date): 31. 12. 76	

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 messurements at the Karstein test stand

3.5 Evaluation of tests

3.6 Measurements in a 1300 MW standard plant

1156

4. <u>Experimental Facilities</u>

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>

The BWR flow model test facility 1 : 5 scale was modified and furnished with a complete set of measurement points and the additionally prepared control rod guide tubes. Due to the delayed delivery of the pressure taps necessary for the fluctuation measurements, the model could not yet be placed in operation. Presently the pressure taps are being installed and calibrated in the control rod guide tubes. In addition, the outlet pressure-regulating valves of the control rod guide tubes are being adjusted to the distributions in the reactor.

6. <u>Results</u>

No results on velocity distribution and pressure fluctuation in the final model version have been obtained up to date due to the preparatory work. Measurements are scheduled for early Januar 1977.

7. <u>Next Steps</u>

The following measurements are planned according to the test program:

1. The velocity distributions at 22 measurement points will be determined on a 360 ° model with parallel operation of all pumps. The two additionally prepared control rod guide tubes which can be rotated allow for an overall pressure-fluctuation measurement around the control rod guide tubes. These measurements will be recorded on a magnetic tape, and the frequencies occurring will then be analysed.

2. After measurement on the 360 ° model has been completed, the flow model will then be modified by simulating a 36 ° pump section, using pump No. 7. The necessary bottom parts and side walls have been partially completed and/or will be installed during model modification. Then the same test will be performed as with the 360 ° model in order to find any possible deviations. In this case, one of the by-pass lines can be modified in order to adjust the flow distributions to those of the full-scale model.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

1159

	Classification
	$\boxed{7.1}$/10.4/11.2/11.4
Title 1 BEAMDYN - Program til lineær 3-dimensionel statisk og dynamisk analyse af bjælkestrukturer.	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2 BEAMDYN - Program for the linear 3-dimensional static and dynamic analysis of beam type structures.	Project leader Per Lundsager Scientists:
Initiated (date): November 1975 Completed: (date) 1976 Status: progressing Last updating (date) February 1976	Per Lundsager

134 - 21 - 21/4113.02

Classification : 10 - 4

1161

Titre Vibrations induites dans les circuits par les écoulements fluides	Pays : FRANCE Organisme directeur : CEA
Titre (anglais) Flow induced vibrations in pipes	Organisme exécuteur : CEA/DEMT Responsable : LIVOLANT
Date de démarrage : 1/1/73 Date prévue d'achèvement : 31/12/78 Etat actuel : en cours Dernière mise à jour : 5/11/76	Scientifiques : GIBERT

Objectif général :

Connaissance fondamentale des éléments de circuit soumis à un écoulement fluide turbulent, permettant d'évaluer, dès le stade de la conception, les phénomènes vibratoires apparaissant au niveau des singularités.

Objectifs particuliers :

1) Etude des élargissements brusques
2) Etude des diffuseurs
3) Etude des fonctions en "T" de deux éléments de circuit
4) Etude des systèmes type diaphragme (vannes à opercule)
5) Etude des obstacles transversaux
6) Etude d'assemblage simples de structures en écoulement fluide.

Installations expérimentales et programme :

Boucle à eau (Gascogne) DEMT

Etat de l'étude :

1) Avancement à ce jour :

ETUDES DE SINGULARITES D'ECOULEMENT -

L'étude d'un certain nombre de singularités courantes est maintenant achevée.

ETUDE DE TUBES EN ECOULEMENT TRANSVERSAL -

Les premières difficultés expérimentales ont été surmontées. De nombreuses et longues mises au point ont été nécessaires.

.../...

2) Résultats essentiels :

a) Singularités :
Les principaux résultats ont été regroupés et une synthèse a pu être réalisée.

b) Ecoulement transversal :
Les essais concernant l'effet d'une vibration forcée sur l'effort exercé par le fluide sur un tube isolé viennent d'être effectués et sont en cours d'interprétation.

Prochaines étapes :

Les expériences sur deux barreaux seront effectuées courant 77.

Documents de référence :

Rapport EMT/76/40 : "Etude des fluctuations de pression dans les circuits parcourus par des fluides - sources de fluctuations engendrées par les singularités d'écoulement (Théorie Générale - Formulaire et abaques)".
- "Etude expérimentale de deux singularités d'un circuit" (Note C.E.A. 1735)
- "Sources acoustiques associées à la réunion de deux écoulements dans un circuit" (Rapport EMT/75/13).

TITLE 1 (original language) Comportamento vibrazionale di componenti del nocciolo	Classification 10.4
TITLE 2 (english) Vibration Behaviour of Core Components	Country: ITALY Sponsor: CNR - CNEN Organisation: University of Pisa
Date initiated 1973 Date completed 1976 Last updating June 1976	Project Leader 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 Organisation CNEN-CISE
Date initiated 1974 Date completed In progress Last updating March 1977	Project Leader 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 showa 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.

1166.

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.

8 - Degree of availability

To a limited extent.

Classification 10.4	
Title 1 Ruisanalyse van de reactor en het primaire circuit	Country The Netherlands Organization KEMA
Tittle 2 Reactor and primary circuit noise analysis	Scientists J. Hoekstra J. v.d. Veer E. Türkçan

1. General aim

On load surveillance of vital systems.

2. Particular objectives

Dynamic behaviour of a BWR.

3. Experimental facilities

Dodewaard nuclear power plant.

4. Project status

Still in progress.

5. Next steps

Not applicable.

6. Relation with other projects

Same investigation at Borssele nuclear power plant by E. Türkçan.

7. Reference documents

None.

8. Degree of availability

Through the organizations KEMA and RCN.

11. MATERIALS AND MECHANICAL PROBLEMS IN NORMAL AND ACCIDENT CONDITIONS

1169

	Classification 11.1
Title 1 Mixed oxide fuel element behaviour under irradiation	**Country** : Belgium **Sponsor** : BELGONUCLEAIRE **Organisation** : BELGONUCLEAIRE-CEN/SCK
Title 2	
Initiated : 1972 **Completed** : - **Status** : in progress **Last updating** : June 1977	**Project leader** : Mr. H. Bairiot.

1. General Aim

Study of mixed oxide fuel element behaviour under irradiation.
Post-irradiation examination.

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 mixed oxide fuels. Neutron radiography.
- 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.

Essential results : Power peaking effects, burnable poison effects, fuel-clad interaction, fission products.

5. Next steps : -

6. Relation with other projects : -

7. Reference documents : -

8. Degree of availability : Proprietary.

Classification: 11.1	
Title 1 (Original Language): Untersuchungen zum Einfluß des Oxidbrennstoffes und von Spaltprodukten auf die mechanischen Eigenschaften von Zry-Hüllrohren bei Störfalltransienten (PNS 4235.3 - I.1.3, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** **ORGANIZATION:** GfK Karlsruhe Projekt Nukleare Sicherheit
Title 2 (English): Investigations of the Influence of Oxide Fuel and Fission Products on the Mechanical Properties of Zry-Cladding Tubes under Accident Conditions	**Project Leader:** P.Hofmann / IMF I
Initiated (Date): 1974 **Completed (Date):** 1978/79 **Status:** Continuing **Last Updating (Date):** 31.12.1976	

1. General Aim

Determination of the extent of inner corrosion and of the influence of chemical interactions between the oxide fuel and fission products, on the one hand, and the Zry-cladding material, on the other hand, on the mechanical properties of Zry-4 at elevated temperatures.

2. Particular Objectives

Influence exerted by the oxygen potential (O/U-ratio) of UO_2 on the strain and rupture behavior of short Zry-4-cladding tubes under LOCA-conditions.

UO_2/Zry-reaction experiments under PCM-(power cooling mismatch) conditions. Study of the chemical interactions between fission products and Zry.

3. Experimental Facilities

The burst tests with Zry-cladding tubes were performed in a tube furnace under cover gas.

The UO_2/Zry-reaction experiments were carried out in a high pressure autoclave system.

4. Project Status

4.1 Progress to Date

The tubular specimens were introduced in a furnace kept at a steady-state temperature of 1100 and 1200°C, respectively. The inner prepressure at room temperature varied between 1 and 60 bar. The temperature rise and pressure build up in the specimens was measured continuously. Partly, the strain rate of the Zry cladding tubes was also measured by an optical method.

The UO_2/Zry-reaction experiments are performed at temperatures up to 1400°C. The UO_2/Zry-contact pressures varied between 0 and 70 bar.

The studies have been carried on relating to the chemical interaction between the fission products and Zry.

4.2 Essential Results

Whereas at low internal pressures hyperstoichiometric UO_2 causes a Zry cladding strengthening due to the oxygen uptake, at high internal pressures (> 15 bar) the influence of the oxygen potential of the UO_2 (O/U-ratio) on the strain and rupture behavior of the Zry cladding tubes is clearly reduced. At large differential pressures the heating rate and the pressure plot have practically no influence on the burst temperature of the cladding tubes. In accordance with data from the literature the burst strains of the tubular specimens vary in argon atmosphere between 20 and 130% and show a minimum in the two-phase region of Zry at about 930°C.

Initial UO_2/Zry-reaction experiments performed under PCM-conditions show that the chemical interactions depend on the contact pressure at the UO_2/Zry-phase boundary. Under good contact conditions UO_2 is reduced above 1000°C by Zry with the simultaneous formation of α-Zr(O), uranium and a (U,Zr)-alloy.

Compatibility investigations of fission products and Zry up to 1200°C showed that only tellurium, iodine and cesium carbonate react with Zry.

5. Next Steps

Continuation of the experiments with short LWR fuel rod simulators under LOCA-typical temperature and pressure conditions. Continuous determination of the strain rate of the cladding tubes.

Influence of fission products and thin oxide layers on the chemical interactions of UO_2 and Zry under PCM-conditions.

Isothermal and transient studies relating to stress corrosion cracking of Zry-cladding tubes due to iodine and other volatile fission products.

Determination of reaction products occuring as a result of the chemical interactions of the fission products and Zry.

6. Relation with other Projects

PNS 4235.1 and 4235.2

7. Reference documents

/1/ 1st PNS Semi-Annual Report 1976, KFK 2375 (German with English abstract)

/2/ 2nd PNS Semi-Annual Report 1976; KFK (German with English abstract)

8. Degree of availability

Unrestricted distribution

Classification: 11.1	
Title 1 (Original Language): KFA/KWU-Leistungsrampen Teil A Testbrennstabbestrahlungen 1976 (RS 203 - I.1.3, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): KFA/KWU-Power Ramps Part A Fuel Rod Irradiation Tests 1976	Project Leader: H. Knaab
Initiated (Date): 1. 4. 76 Completed (Date): 31. 12. 77 Status: Continuing Last Updating (Date): 31. 12. 76	

1. General Aim

Power ramp tests in HFR Petten are being carried out with PWR and BWR standard fuel rods of different production parameters, in order to investigate the reason of fuel rod defects, especially the fuel-cladding tube interaction.

2. Particular Objectives

The mechanism causing fuel rod defects by alternating power loads and positive power ramps is not yet well known. In practice fuel rod defects have been discovered after local power variations in power reactors and in power ramp tests at 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 and UO_2/PuO_2 respectively) 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 5.000 - 24.000 MWd/t(U) burn-up.

1176

3. Program

3.1 Extraction and sectioning of segmented PWR and BWR fuel rods, intermediate inspection of the fuel rod segments/test fuel rods.

3.2 Neutronradiografic 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 to 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 cycles. In the pool-side-facility of the HFR Petten power ramps will be carried out up to linear heat rates $q' \leq 620$ W/cm with ramp rates $\dot{q}' \leq 100$ W/cm min, and heat rate increases $\Delta q' \leq 350$ W/cm. The test capsules will be instrumented with inductive elongation detectors in order to measure changes of the fuel rod length. The tests will be performed as a joint project of KFA Jülich and Kraftwerk Union.

The following parameters will be varied:

fuel:	UO_2, UO_2/PuO_2 density, pellet geometry
cladding tube:	production parameters (e.g. graphite on inner surfaces)
fuel rod:	pressurization, burn-up

5. Project Status/ Progress to Date

The segmented fuel rods No. 1902, 1201, 1202, 1203 and 1204 were sectioned and inspected in the hot cells at Karlstein. The segments were transported to Petten.

6 preirradiated segmented fuel rods with UO_2 and UO_2/PuO_2 pellets were unloaded from Obrigheim.

Several test fuel rods were ramped in in-situ and start-up ramp tests; the linear heat rates were increased from 300 W/cm to 460 - 608 W/cm with different ramp rates.

Finally the fuel rods were investigated by neutronradiography, γ-scanning and eddy current tests. The diameter was measured.

2 not irradiated fuel rods were transported to Petten, 49 segments preirradiated in KWO were transported from Obrigheim to the hot cells in Karlstein. The transport of 10 ramped fuel rods from Petten to Karlstein was prepared.

The evaluation of the first results was started.

6. Project Status/ Essential Results

No fuel rod defects were detected (no fission products were found in the coolant of the irradiation capsule) during the slow or fast start-up ramps. The eddy current inspection however analysed some influences of the ramps. The signals indicated circumferential ridges of the cladding tubes. No signals were found, which were due to cladding defects.

Only during the in-situ ramp IS 5 water has penetrated through the cladding and has filled half the bottom plenum.

The neutronradiography showed that at high heat rates the dishings of the UO_2-pellets have nearly been closed and that some cracks existed in the pellets.

The γ-scanning (La-140) indicated that during the ramp tests more than 60 % of the active rod length have been in the ramp of 90 % - 100 % maximum load.

The length of the fuel rods was about 0,2 - 0,3 o/oo higher after the ramp test than before ramping.

The visuell inspection showed no failures, but on some fuel rods deposits caused by surface boiling were found. The diameter of the test fuel rods was not essentially changed, but circumferential ridges up to 4 μm height were found, which were concentrated on the pellet-pellet interfaces. Some ridges were placed near pellet cracks.

The start-up ramps up to 530 W/cm caused larger ridges than the in-situ ramps up to 460 W/cm.

7. Next Steps

Continuation of the specified irradiation program and post-irradiation-examination of the ramped test fuel rods.

Retransfer of 10 irradiated test fuel rods from Petten to Karlstein for post-irradiation examination.

Intermediate inspection of 28 preirradiated test fuel rods in hot cells at Karlstein.

Transport of 28 preirradiated test fuel rods to Petten.

Continuation of the evaluation of results.

Preparation of a publication of the first results.

8. Relation with Other Projects

The preirradiation of 34 segmented UO_2- and UO_2/PuO_2-fuel rods was sponsored by the BMFT (AtT 7384 and 7704).

9. References

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10. Degree of Availability

1181

Classification: 11.1

Title 1 Multi - Rod Burst and Flow Blockage	Country BRD (U.S.A) Sponsor Babcock and Wilcox Proprietary
Title 2	Organisation BBR Mannheim
Initiated: June 1974 Completed: Status: Last updating: Dec.1975	Project leaders Dr. B. E. Bingham Dr. W. A. Fiveland

1. General aim

To provide data on rod swelling and burst characteristics of B & W production Zircaloy-4 cladding under loss-of-coolant accident (LOCA) conditions and to investigate possible channel flow blockage conditions.

2. Particular Objectives

3. Experimental Facility and Program

In 1975, rod burst and flow blockage tests were conducted with single-rod and five-rod test sections. These tests were conducted over a range of pressure differentials from 400 to 1400 psi at a heating rate of 50F per second.
The rods are internally heated by a resistance element. The test section containment is evacuated to minimize natural convection and the shroud is heated at the same rate as the cladding to simulate the proper boundary conditions of adjacent rods during the postulated LOCA. Test section power, rod pressure and temperature, and containment pressure and temperature are monitored. After the rods have been burst, blockage factors are determined over a wide range of Reynolds numbers. Finally, the test sections are encased in epoxy and sectioned at 1/2" intervals to determine the diametral expansion, blocked flow area, and thinned cladding thickness as a function of axial length at both ruptured and swelled locations.

1182

4. Project Status

4.1 Progress to Date
The experimental phase has been completed.
Evaluation of the data is in progress.

4.2 Essential results

5. Next Steps

Publication of topical report.

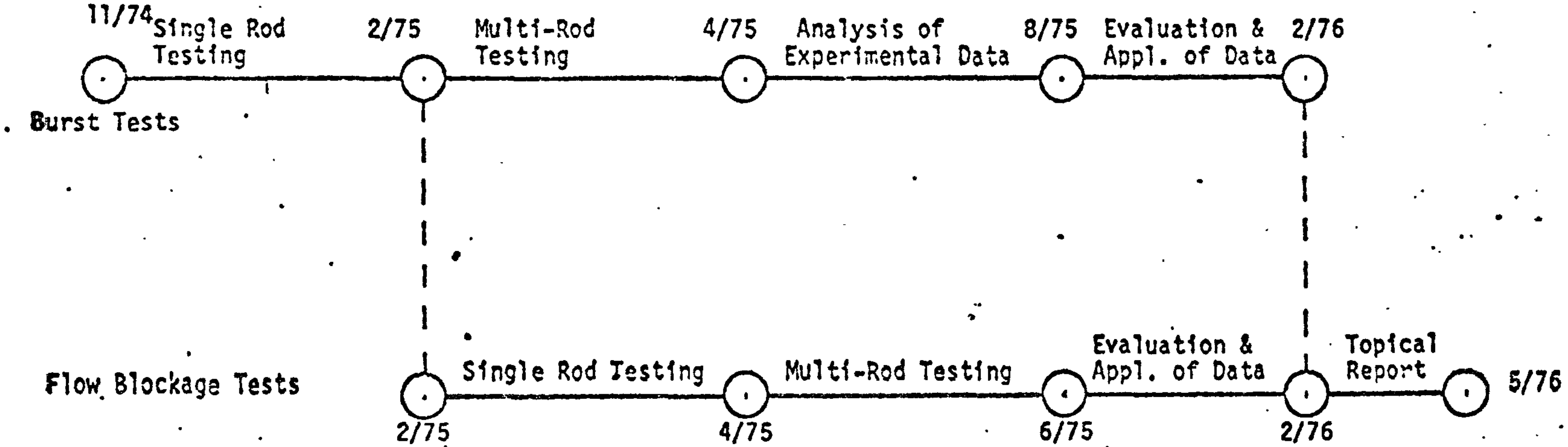
6. Relationship With Other Projects

7. Reference Documents

8. Degree of Available

Proprietary
(It is intended to publicly release the results during the fourth quarter of 1976)

MULTI-ROD BURST & FLOW BLOCKAGE



1/10/75

Classification:

Title 1 Fuel Post Irradiation Examination		Country BRD (U.S.A) Sponsor Babcock and Wilcox Proprietary
Title 2		Organisation BBR -Mannheim
Initiated: Mid 1974 Status: Progressing	Completed: Last updating: End 1975	Project leaders M. H. Montgomery L. J. Ferrell

1. General Aim

This program consists of two parts; Part I, surveillance of the Mark B (15 x 15) assembly, and Part II, surveillance of the Mark C (17 x 17) assembly, Part I provides verification of performance for the 15 x 15 design in the present generation of B&W reactors. Part II will provide verification of performance for the Mark C (17 x 17) design of fuel.

2. Particular Objectives

To obtain performance data on irradiated fuel.

3. Experimental Facilitica and Program

The following data will be accumulated with a non-destruction examination at the reactor site of Oconee I fuel at the end of each of three cycles.

a) fuel assembly length and grid location
b) fuel assembly grid spring and holddown spring constants
c) guide tube inside diameter
d) fuel rod length, diameter and bow
e) rod to rod spacing
f) surface deposit characterization
g) gamma scans
h) visual examination

The destructive examination of a fuel assembly at the end of each of three cycles of Oconee I will be performed at the B&W Lynchburg Research Center. This will include the following work.

a) visual examination
b) gross and spectral gamma scan
c) fuel rod length
d) fuel rod diameter and ovality
e) eddy current examination of clad integrity
f) ultrasonic clad thickness
g) internal gas volume pressure and composition
h) mechanical properties of cladding
i) pellet density determination
j) crud analysis
k) chemical burnup analysis
l) graphics of tube and clad

4. Project Status

A. Progress to Date

Precharacterization was performed on selected fuel assemblies in Oconee I core 1 .

The non-destructive examination of Oconee I cycle 1 has been completed and the second cycle examination initiated.

The destructive examination of a selected fuel assembly started September 1975 and will be completed by mid 1976.

B. Essential Results

Mark B (15 x 15)

The first refueling of a B&W 177-Fuel Assembly NSS took place in later 1974. During this refueling, six (6) fuel assemblies that were reinserted in the core, as well as thirty (30) discharged fuel assemblies, were examined. All fuel assemblies were visually and/or dimensionally examined.

All four sides of each fuel assembly were viewed along the entire fuel length. The visual examination showed the fuel assemblies to be structurally sound with no evidence of wear or gross rod bowing. One hundred seven (107) fuel rods were gamma scanned for fuel stack length. Measured fuel column shortening was conservatively predicted by the B&W densification kinetics model. In excess of 7000 water channel spacings were measured and results showed that 95 % of all rod-to rod spacings changes less than 14 mils. Fuel rod and fuel assembly growth measurements were well within the design envelope. Fuel assembly holddown spring measurements show little or no spring relaxation resulting from irradiation after one cycle of operation.

Mark C (17 x 17)

The two demonstration fuel assemblies have been fabricated for insertion into Oconee II cycle 2.

5. Next Steps

Program will be continued as indicated in attached time schedule diagrams.

6. Relation With Other Projects

7. Reference Documents

8. Degree of Availability

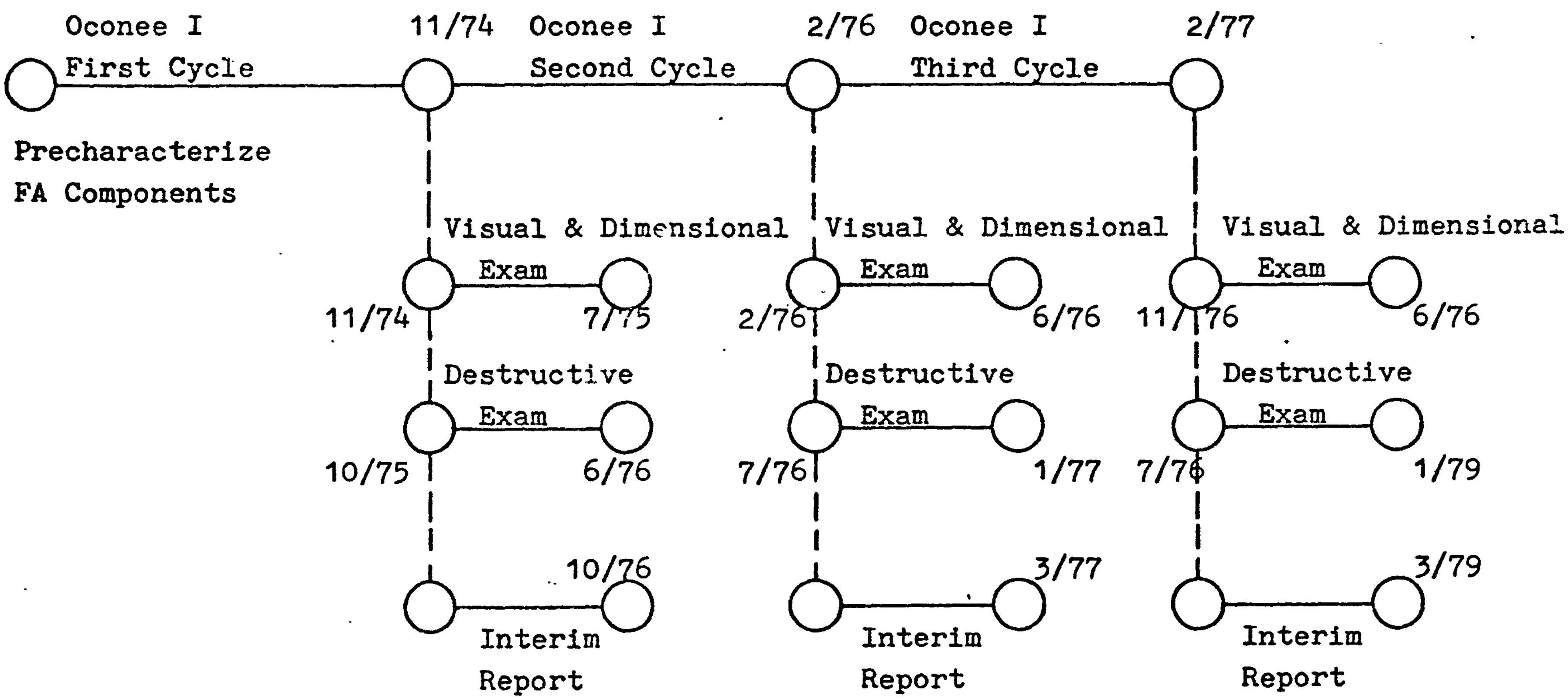
Proprietary

1. Budget

2. Personnel

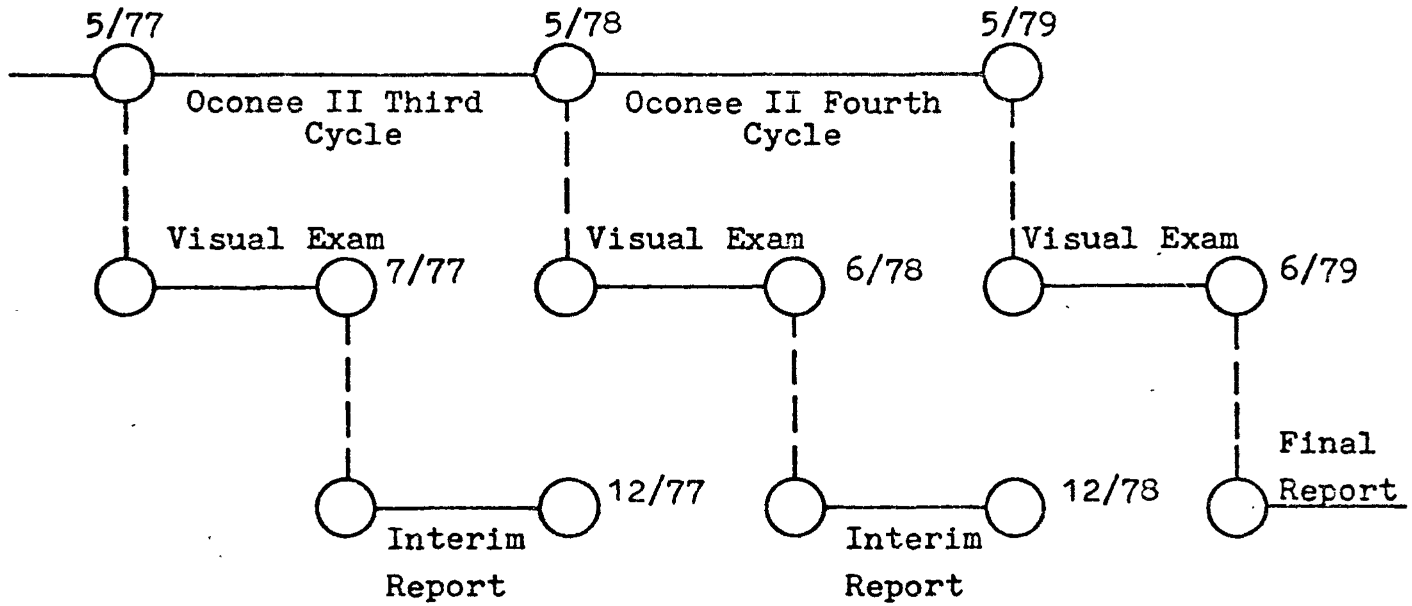
3. Additional informations
See attached time schedule diagrams.

FUEL POST IRRADIATION EXAMINATION PART I - MARK B (15 x 15)



Subject to Plant Operating Schedule

RADIATION EXAMINATION PART II - MARK C (17 x 17)



Subject to Plant Operating Schedule

1191

Classification: 11.1

Title 1 Ramp Testing of UO_2-Zr Fuel	COUNTRY: Denmark SPONSOR: DAEC, Risø ORGANIZATION: DAEC, Risø
Title 2 Ramp Testing of UO_2-Zr Fuel	Project Leader: P. Knudsen
Initiated: 1972 Completed: 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

Classification: 11.1

Title 1 Fuel Modelling	COUNTRY: Denmark SPONSOR: DAEC, Risø ORGANIZATION: DAEC, Risø
Title 2 Fuel Modelling	Project Leader: N. Kjær-Pedersen
Initiated: 1972 Completed: Status: Progressing Last Updating:	Scientists:

1. General Aim

To develop a computer model complex for fuel pin performance evaluation under nonaccident operating conditions.

2. Particular Objectives

A new version, WAFER-2, of the WAFER model is under development. Differences from WAFER-1 are: a) An improved finite difference treatment of the pellet with the aim of obtaining a more realistic fuel stress distribution and cracking pattern, and b) programmatic changes to minimize accumulation of error.

A simplified (disc) model, HOTCAKE, covering several fuel pin cross-sections is being built for faster analysis and improved fission gas release evaluation.

3. Experimental Facilities and Programme

Danish ramp testing programme.

4. Project Status

4.1. Progress to date: WAFER-1 in normal production status.

5. Next Steps

Completion of WAFER-2 and HOTCAKE.

1194

6. Relation with Other Projects

Will utilize data from Danish overpower test programme for verification.

7. Reference Documents

P. Knudsen and N. Kjær-Pedersen: Performance Analysis of PWR Power Ramp Tests. ASME Winter Annual Meeting 1975, Houston, Tx. (75-WA/HT-68).

N. Kjær-Pedersen: Mathematical Description of WAFER-1, A three-Dimensional Code for LWR Fuel Performance Analysis. Nucl. Eng. and Des. 35 (1975) 387-398.

8. Degree of Availability

Not generally available.

130-11-11/4112-07

Classification : 11-1

1195

Titre	Pays : FRANCE
Mise au point des codes de calcul de sûreté, pour les éléments combustibles des réacteurs à eau en régimes permanents et transitoires.	Organisme directeur : CEA/DSN
Titre (anglais) Development of PWR fuel elements safety computer codes under steady state an transient conditions.	Organisme exécuteur : CEA/DMECN-SECS (Saclay)
	Responsable : M. CHAGROT (DSN-SETSSR) G. LESTIBOUDOIS (SECS)
Date de démarrage : 01/01/77 Date prévue d'achèvement : 31/12/80 Etat actuel : en cours Dernière mise à jour : 01/03/77	Scientifiques : G. ORSIER D. HACKER

Objectif général :

Le but de cette action est de disposer, à terme, d'un code de calcul des éléments combustibles susceptibles de rendre compte de leur comportement mécanique et thermique, tant en régime permanent qu'en régime accidentel.

Objectifs particuliers :

Juger de la représentativité des modèles introduits dans les codes existants.
Comparer les codes entre eux.
Développer un code susceptible de décrire le comportement en en régime permanent (état de référence) et pendant les transitoires, y compris les transitoires accidentels.

Etat à l'étude :

1) Avancement à ce jour :

Etude commencée.

2) Résultats essentiels :

Néant.

Prochaines étapes :

Juger de la représentativité des modèles existants.

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1196

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

- Evaluation des limites de fonctionnement sur des combustibles.
- Dépouillement des résultats des essais PHEBUS. Evaluation du comportement du combustible au cours des accidents.

1197

Classification 11.1

Title 1 (original language)	COUNTRY France
SURETE ELEMENT COMBUSTIBLE	SPONSOR
	ORGANIZATION
	C.E.A. DSN
Title 2 (english)	Project leader C.RINGOT
	Scientists : H. VIDAL
Initiated (date) 1972 — Completed : (date)	
Status : progressing — Last updating (date)	

1. OBJECTIF PRINCIPAL

L'intégrité de la gaine des éléments combustibles qui constitue la première barrière aux produits de fission doit être assurée pendant toute la vie. Cela signifie qu'il convient d'éviter toute rupture à caractère systématique et que le comportement d'un élément rompu ne conduise pas à une évolution catastrophique Pendant un accident, tel que la dépressurisation du circuit primaire, l'élément combustible doit se comporter de telle sorte que le maintien de la fonction sûreté de la centrale soit assuré.

2. OBJECTIF PARTICULIER

Le programme des études est directement rattaché à l'objectif principal.

3. SITUATION

Les principales études sont :

a) comportement d'éléments rompus en pile-taux de relâchement des produits de fission d'éléments rompus.

Test d'irradiation BOUFFON dans le réacteur SILOE à GRENOBLE

1/12/74 - 12/75

Détermination de la durée de vie possible en pile d'un élément présentant une rupture primaire.

Les premiers résultats font apparaître un dégagement accru des produits de fission avec des éléments combustibles rompus.

b) comportement des gaines à haute température pendant le LOCA. Programme EDGAR

Cette étude constitue un support indispensable des essais en réacteur-appelés PHEBUS.

Le but est d'obtenir une courbe réelle des déformations diamétrales de la gaine en fonction du temps. Les résultats sont directement appliqués à l'élaboration d'un code.

- essais en laboratoire et en pile
- planning 1/12/75 - fin 76.

1198

c) critère de rupture des éléments combustibles des réacteurs bouillants pendant des transitoires accidentels :

		- boucle en pile	
planning	(	- étude de la boucle	1975
	(	réalisation de la boucle	1975
	(	expériences	78-79

d) comportement des éléments combustibles des réacteurs bouillants dans des conditions normales.

Détermination d'une courbe limite de puissance admissible avant rupture en fonction du taux de combustion : "STRIP" Test

programme en collaboration avec la Suède

- réacteur R2 à STUDVICK
- planning Juillet 75 - 80

Titre	Pays : FRANCE
Limitation du fonctionnement des éléments combustibles dues à l'existance de ruptures de gaine.	Organisme directeur : CEA
Titre (anglais) Operational limits of fuel elements dueto cladding defects.	Organisme exécuteur : CEA/DMECN-SECS (Saclay)
	Responsable : G.LESTIBOUDOIS
Date de démarrage : 01/01/72 Date prévue d'achèvement :31/12/77 Etat actuel : en cours Dernière mise à jour : 01/03/77	Scientifiques : G.DE CONTENSON B.HOUDAILLE

Objectif général :

Déterminer le comportement d'un crayon combustible des PWR présentant une rupture de gaine.
En déduire éventuellement des limitations à ses conditions de fonctionnement.

Objectifs particuliers :

Etudier les problèmes d'hydruration des zircaloy en présence de vapeur d'eau.
Rassembler des résultats sur le dégagement des produits de fission à partir de crayons défectueux.

Installations expérimentales et programme :

Les essais d'irradiation sont effectués à SILOE.
Les dispositifs expérimentaux sont appelés BOUFFON.
Les diverses irradiations sont effectuées à des niveaux de puissance variable et pendant des temps différents.

Etat de l'étude :

1) Avancement à ce jour :

- De nombreuses difficultés de mise au point du dispositif ont retardé les programmes. Des essais ont été réalisés dans de bonnes conditions, avec des flux thermiques superficiels, sur la gaine, étagés entre 140 et 180 W/cm2.

.../...

1200

2) Résultats essentiels :

Les essais n'ont pas montré, contrairement à ce que l'on pouvait penser, qu'il se produisait une détérioration rapide de la gaine par attaque hydrurante lorsque le crayon fonctionnait avec une rupture de gaine ouverte.

Prochaines étapes :

Vérification, par examen en cellule chaude, des conditions dans lesquelles se développent les couches d'oxyde à l'intérieur des gaines en zircaloy, et la répartition des hydrures aussi bien à l'intérieur qu'à l'extérieur.

Relation avec d'autres études :

Etude des limitations des conditions de fonctionnement des centrales de puissance.

Documents de référence :

"Examen irradiation de BOUFFON 08", DTECH-SECS, Compte-Rendu d'examen N° 45.
"Examen après irradiation de BOUFFON 09", Note Technique SECS/SEECRE-704.
"Emission de produits de fission par un crayon présentant une rupture en fabrication - BOUFFON 09J49", Compte-Rendu DMG 35/76.

130-31-11/4112-02

Classification : 11-1

1201

Titre	Pays :
Conséquence d'une rupture de gaine sur les combustibles CARAMEL.	FRANCE
	Organisme directeur : CEA
Titre (anglais) Effect of a cladding rupture ou CARAMEL fuel element behaviour.	Organisme exécuteur CEA/DMECN-SECS
	Responsable : G.LESTIBOUDOIS (SECS)
Date de démarrage : 01/01/77 Date prévue d'achèvement 31/12/80 Etat actuel : en cours Dernière mise à jour : 01/03/77	Scientifiques : G.DECONTENSON B.HOUDAILLE

Objectif général :

Etudier les problèmes de détection des ruptures de gaines d'un élément combustible CARAMEL.
Etudier l'évolution d'une rupture de gaine sur ce type d'élément.

Objectifs particuliers :

Vérifier comment réagissent les dispositifs courants de détection de rupture de gaine dans le cas des CARAMEL.
Vérifier s'il y a dégradation du combustible ou du gainage.

Installations expérimentales et programme :

Les essais sont envisagés à SILOE et à OSIRIS.
Les dispositifs à utiliser seront les suivants : BOUILLEUR, IRENE, BOUFFON.
Le programme 1977 sera limité à 2 essais dans le dispositif BOUFFON.

Etat de l'étude :

1) Avancement à ce jour :

Préparation des irradiations en cours.

2) Résultats essentiels :

Néant.

.../...

1202

Prochaines étapes :

Irradiation du premier crayon en février 1977.

Relation avec d'autres études :

Etude des limitations de fonctionnement des réacteurs à eau à combustibles CARAMEL.

Classification 11.1

Title 1 OVERPOWER TESTS	**Country** Italy
Title 2	**Sponsor** CNEN
Initiated 1.3.1974 **Completed** sept. 75 **Status** Progressing **Last updating** End 74	**Organisation** CNEN **Project Leaders** 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. Esperimental 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 subsequent over power tests aleready set-up

1204

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.

1205.

Title 1 (Original language) Prove di scoppio su guaina	Classification 11.1
Title 2 (English) Burst tests	Country ITALY Sponsor CNEN Organisation CNEN
Date initiated January 1973 Date completed 1976 Last updating December 1976	Project Leader 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: $6\div30\times10^{20}$ 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

Open

1207

TITLE 1 (original language) Sviluppo elementi di combustibile per HWR	Classification 11.1
TITLE 2 (english) HWR fuel development	Country: ITALY Sponsor: CNEN Organisation: CNEN
Date initiated 1975 Date completed In progress Last updating June 1976	Project Leader 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.

1209

PROJECT TITLE : Cladding in LOCA conditions	CLASSIFICATION 11.1
SPONSORING COUNTRY : Italy	ORGANISATION : CNEN
DATE INITIATED : January 1975 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'intercape-dine combustibile/guaina in barre di combustibile BWR	Classification 11.1
Title 2 (English) Gap conductance in BWR fuel rods	Country ITALY Sponsor CNEN Organisation CNEN
Date initiated 1968 Date completed in progress Last updating April 1977	Project Leader 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.

1212

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 **Organisation** CNEN
Date initiated 1977 Date completed 1977 Last updating April 1977	**Project Leader** 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_i (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 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.

1215

Title 1 (Original language) Comportamento meccanico e chimico di guaine di Zr ad alta temperatura	Classification 11.1
Title 2 (English) High temperature mechanical and chemical behaviour of cladding tubes.	Country ITALY Sponsor CNEN Organisation CNEN
Date initiated 1975 Date completed in progress Last updating April 1977	Project Leader 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.

12.17

PROJECT TITLE : RELIABILITY OF CLASSICAL NDT	$\boxed{12.3}$ 11.2.3 LWR 11.1
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : JRC Ispra
DATE INITIATED : 1977 DATE COMPLETED : 1980	PROJECT LEADER : S.J. CRUTZEN

Classification 11.1	
Title 1 Splijtstofverlenging tijdens bedrijf	Country The Netherlands Organization KEMA
Title 2 Fuel elongation during power rating	Scientists W.R. van Engen J. Hoekstra W. Slegers

1. General aim

Pellet-clad interactions.

2. 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

3. Experimental facilities

Dodewaard nuclear power plant.

4. Project status

Still in progress.

5. Next steps

To come to an additional method for incore measurements to have better knowledge over actual core conditions.

6. Relations with other projects

Measuring technique is based on the Halden incore instrumentation experience with differential transformers.

1220

7. Reference documents

None.

8. Degree of availability

Through the organization KEMA.

Classification 11.1

Title 1 Fuel Clad Distortion under Accident Conditions.	**Country** UK
Title 2	**Sponsor** UKAEA **Organisation** Reactor Fuel Laby. Springfields
Initiated 1971 **Completed** **Status** Continuing **Last updating**	**Project leaders** 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 UNITED KINGDOM
HIGH TEMPERATURE CREEP OF ZIRCALOY	SPONSOR UKAEA
	ORGANIZATION RFL SPRINGFIELDS
Title 2	Project Leader 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 FUEL CLAD DISTORTION UNDER ACCIDENT CONDITIONS	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION RFL SPRINGFIELDS
Title 2	Project Leader DR K M ROSE
Initiated 1971 Completed : 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.

Classification: 11.2	
Title 1 (Original Language): Konstruktive und rechnerische Entwicklungsarbeiten an Druckbehältern in Stahlbauweise (RS 35 - II.1.7., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fried. Krupp GmbH
Title 2 (english): Design and Calculation of Steel Pressure Vessels	Project Leader: Jorde
Initiated (Date): May 1, 1969 **Status:** finished	**Completed (Date):** December 31, 1975 **Last Updating (Date):** December 1975

1. General aim

Steel pressure vessels are currently finding an ever increasing field of application. Their use in modern technologies, such as reactor engineering, requires considerable design and calculation procedures as well as intensive theoretical and experimental work, because reliability of operation here is a prime consideration aside from the ever important aspects of economy.

This research programme, therefore, aims to analyze, by theoretical and experimental work and by development of appropriate designs the reliability and manufacturing economy of pressure vessels and to develop new approaches. Detail problems are to be solved and progress is to be made in the two, often opposing, aspects mentioned: reliability and economy.

2. Particular objectives

- Theoretical and experimental investigation of the strength behaviour, i.e. strain and stress characteristics, of such vessel components as closure shells with holes, flanges, seals, nozzles and nozzle arrays, supports.
- Design of a removable vessel seal
- Analysis of the plastic behaviour of vessel components for the assessment of their bursting properties.

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- Experimental fabrication of nozzle bodies while joining them to the shell of the vessel
- Extensive static and cyclic loading tests on vessel models made of 22 NiMoCr37 steel with arrays of nozzle.
- Accoustik emission tests

3. Experimental facilities and programme

Extensive static and cyclic loading tests on 6 vessel models (diameter 2200 mm, wallthickness 30 mm) made of 22NiMoCr37 steel with arrays of nozzle. Stress and strain measurements in the elastic and plastic range, accoustik emission analysis, bursting tests.

4. Project status

1. Progress to date
2. Essential results

- Methode to calculate flanges weakened by bolt holes and loaded by concentrated and distributed moments directed tangentially (upturning moments)
- Study of the available literature on stress distributions in cylindrical and spherical pressure vessels with nozzle connections
- Approximate strain and stress analysis of perforated shells of revolution
- Program to calculate plastic deformations of shells
- Removable vessel seal (patent application P 2118254.7)
- Extensive results of the stress and strain measurements and of the bursting tests
- Results of the accoustik emission analysis

5. Next steps

6. Relation to other projects

- Development of large pressure vessels of multi-layer steel construction
- Investigation relating to the manufacture of components with large cross-sectional areas by the deposition of Ni-(Cr)-Mo steels by machine welding
- Connection of multi-layer wall areas to solid components for pressure vessels.

7. Reference documents

(in German language with 1 exception)

- Study report 50/70
  Design and calculation of steel pressure vessels,
  Report No. 1
- Study Report 79/70
  Design and calculation of steel pressure vessels,
  Report No. 2
- Study Report 1001/71
  Design and calculation of steel pressure vessels,
  Report No. 3
- Patent application P 2118254.7,
  Removable vessel seal
- The behaviour of circular rings with bores loaded by equidistant upturning moments; F. Baumgart, R. Dietze, J. Jorde. Structural mechanics in reactor technology, Berlin September 20 - 24, 1971.
- Approximate determination of the strains and stresses of perforated shells of revolution; H. Coenen, H.P. Lehrke, J. Jorde. Structural mechanics in reactor technology, Berlin September 20 - 24, 1971.
- Quarterly and annual reports in the series IRS-Forschungsberichte, report period from January 1971 to December 1975.

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8. Degree of availability

Address for request:

BMFT, Bonn

9. Budget

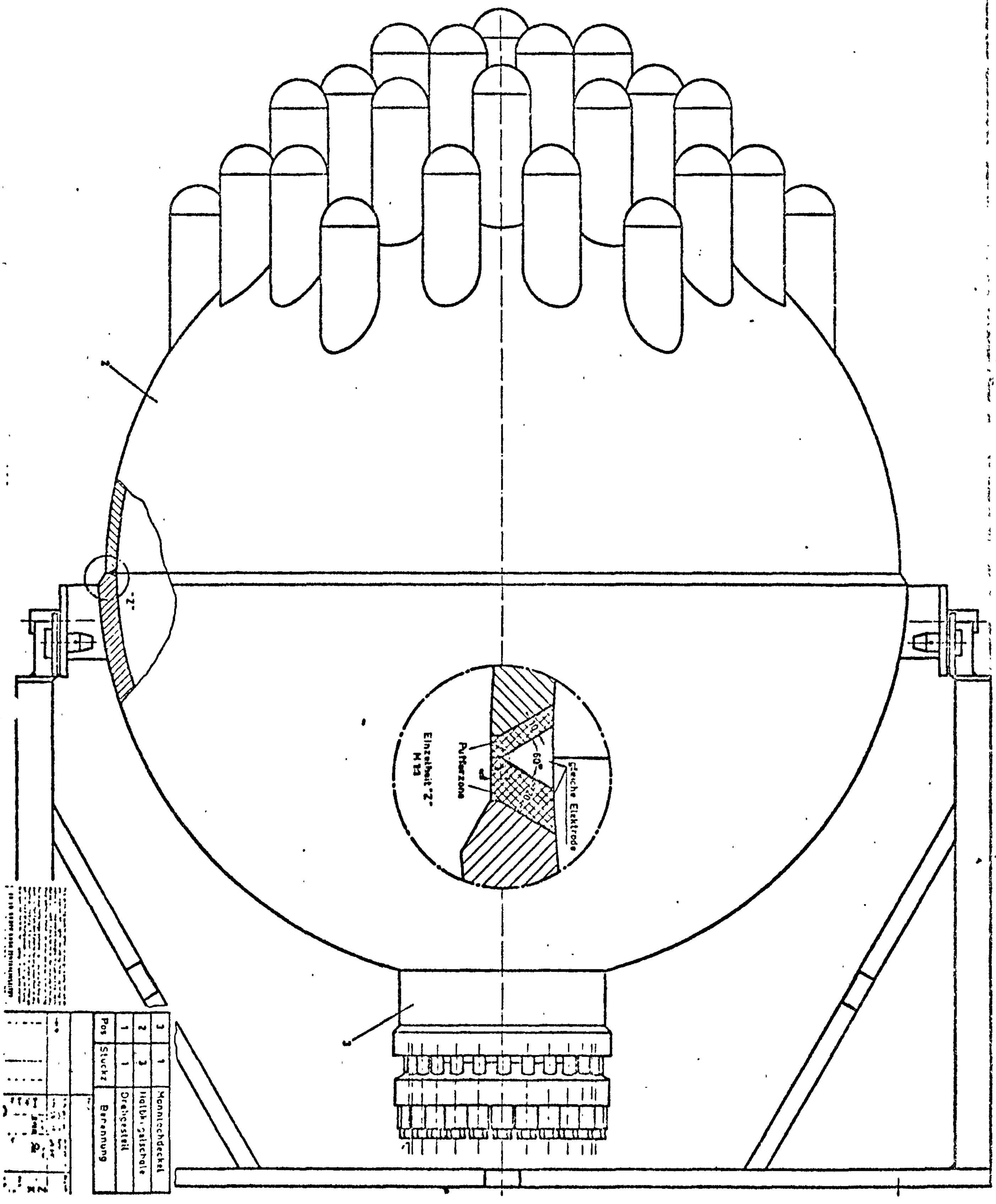
Cost of labor	DM 130.000,--
Cost of material	DM 90.000,--

10. Personnel

14 men/month

11. Additional informations

Side view of the vessel models



J233

Classification: 11.2	
Title 1 (Original Language): Statusbericht Reaktordruckbehälter (RA 63 - II.2., Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: IRS, Köln
Title 2 (English): Status Report Reactor Pressure Vessel	Project Leader: Röhrs Rittig
Initiated (Date): Dec. 28, 1971 Status: continuing Completed (Date): Last Updating (Date): December 1976	

1. General Aim

Within the Reactor Safety Research Program the studies relating to the reactor pressure vessel technique are one of the key points. This Status Report was made to determine really substantive priorities for the use of the available funds.

2. Particular Objectives

In detail the following targets are aimed:

1. To show the latest status of the technical development to the Federal Republic of Germany with respect to design, fabrication, test and examination of reactor pressure vessels (RPV) for light water reactors fabricated by means of rolling construction or forged steel construction by pointing out the principal aspects for the safety assessment.
2. To set up a resarch program for the extension of the basis of the RPV safety assessment and for an improvement of the techniques of design, manufacturing, and test.
3. Concentrated representation of targets, research programs, and essential results of the 12 topics of the Heavy-Section-Steel-Technology-(HSST) Program executed in the USA.

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3. Program

To provide a broad spectrum of experiences and guaranteed objectivity to a large degree it was decided to proceed as follows.
By order of the Federal Minister for Education and Science (BMBW) the Institute of Reactor Safety (IRS) took the lead.Experienced industrial firms, associations, and safety consultant institutions elaborated special contributions according to specifications and by order of the IRS. A group of independent experts analysed these reports with respect to principle essentials as to the safety assessment of RPV. An editing board consisting of leading experts with special experience in RPV-technique took the whole leading.

The Status Report will consist of three volumes.
Volume I will contain all results in concentric form. Starting from the state of RPV-technique at the end of 1973 positive as well as negative knowledge aspects will be shown. As the status report will essentially serve for setting up substantiated priorities for research activities the concise form of volume I will concentrate partly strongly on the gaps of knowledge.

Volume II will consist of text and figures of the data- and information collection. Here the status of RPV-technique will be layed down as found out by the authors of the special reports and as coordinated subsequently in detail.

Volume III will give a summary of the HSST Program presently running in the USA. For each of the 12 topics of this program information will be given as regards objective, research program, project status and essential results.

4. Experimental Facilities, Computer Codes

Not relevant.

5. Progress to Date

Drafts of Volume II (Collection of data and information) and Volume III (HSST-Program) were finished.

6. Results

Volume III has been printed. The editing board has reviewed Volume II.

7. Next Steps

Preparing printing of Volume II.

8. Relation with other Projects

RS 46 Study on Current and Planned Research Projects in the Pressure Vessel Field
STAATLICHE MATERIAL PRÜFUNGSANSTALT, Stuttgart, University
1969 - 1971

9. References

Reports in the series IRS-Forschungsberichte respectively GRS-Fortschrittsberichte:

1975	IRS-F-29	(in English)
Jan.-March 1976	IRS-F-30	(in German)
April - June 76	IRS-F-31	(")
July - Sept. 76	IRS-F-33	(")
Oct. - Dec. 76	GRS-F-35	(")

10. Degree of Availability of the Reports

Unrestricted distribution.

Classification: 11.2	
Title 1 (Original Language): Grob- und Detailspezifikation des Forschungsprogramms Komponentensicherheit (1. Phase) (RS 192 - II.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: MPA Stuttgart
Title 2 (English): Rough and detailed Specification Research Program "Structural Integrity of Components"	**Project Leader:** Dr.-Ing. D. Sturm Dr.-Ing. L. Issler
Initiated (Date): November 13, 1975 **Completed (Date):** April 30, 1976 **Status:** completed **Last Updating (Date):** December 31, 1976	

1. GENERAL AIM

The research program "Structural Integrity of Components" integrates all activities concerning reactor pressure vessel problems running or planned in the Federal Republic of Germany. Objectives of this program are to give basic information for the quantification of safety margin in order to reduce further the remaining risk through a specified further development of the reactor pressure vessel technique.

. PARTICULAR OBJECTIVES

In order to be able to work on this research program "Structural Integrity of Components" with largest possible efficiency it has been planned to divide the program into subsequent phases. For the first phase, which will last approx. 4 years, a rough specification had to be worked out and in addition to the first year a detailed specification.

3. RESEARCH PROGRAM

The first phase of the research program "Structural Integrity of Components" covers the following single projects:

1) Possibility of Defect Detection (Defect Classification)

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2) Homogeneity of the Mechanical Properties (Small Scale Specimens - not irradiated, irradiated)

3) Integral Test on Large Scale Specimens

4) Integral Test on Intermediate Size Vessels

5) Theory

They had to be classified in regard to basic proceedings including test setup, test performance, test time, temporal operating sequence and costs.

4. EXPERIMENTAL FACILITIES, COMPUTER CODES

not necessary for the specification.

5. PROGRESS TO DATE

Working-out of the specification which consists of a rough specification for the complete project of the first phase and of a detailed specification for the first test year. Preparation of the structure of targets, structures of projects, network plans and lists of costs.

6. RESULTS

Completion of the specification.

7. NEXT STEPS

not applicable, as project has been completed.

8. RELATION WITH OTHER PROJECTS

See Annual Report A 75.

9. REFERENCES

Quarterly Reports in IRS Research Reports, period reported on: January 76 until June 76 (German).

Annual Report A 75 (English)

10. DEGREE OF AVAILABILITY OF THE REPORTS

Upon request at BMFT.

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	Classification 7.1/10.4/11.2/11.4
Title 1 BEAMDYN - Program til lineær 3-dimensionel statisk og dynamisk analyse af bjælkestrukturer.	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2 BEAMDYN - Program for the linear 3-dimensional static and dynamic analysis of beam type structures.	Project leader Per Lundsager Scientists:
Initiated (date) November 1975 Completed: (date) 1976 Status: progressing Last updating (date) February 1976	Per Lundsager

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Classification 11.2

Title 1 (original language) SURETE DES CHAUDIERES - PROBLEME DES ACIERS.	COUNTRY France SPONSOR CEA ORGANIZATION CEA/DSN
Title 2 (english)	Project leader C.RINGOT Scientists : B.BARRACHIN A.PROT
Initiated (date) Completed : (date) Status : progressing Last updating (date)	

1. OBJECTIF PRINCIPAL

S'assurer que la conception et le comportement des circuits primaires des réacteurs à eau ordinaire construits en France répondent aux critères de Sûreté.

2. OBJECTIFS PARTICULIERS

- Tenue des aciers de cuve
- contrôle et inspection en service.

3. SITUATION

1/ - Comportement sous irradiation de l'acier A 508 d3 utilisé pour la construction des réacteurs à eau PWR du programme français.

- conditions d'irradiation - dose 10^{14} à 6×10^{19} n/cm2 (≫1 Mev)
  - température - 260°C à 290°C
- éprouvettes tirées des cuves en construction
- traitements thermiques représentatifs des diverses étapes de fabrication.
- propriétés mesurées : traction - résilience
- planning 72/75.

2/ - Comportement à la rupture brutale des aciers utilisés pour la construction des cuves PWR.

a/ Mesure K_IC sur grosses éprouvettes - CT et sur petites éprouvettes (mesure de J)

effet de l'irradiation examiné.

.../...

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premiers résultats publiés dans le cadre du programme coordonné de l'AIEA.

b/ Restauration des propriétésdes aciers pour cuves PWR dégradées par l'irradiation.

Planning 74/75

c/ Détermination des vitesses de propagation des fissures sur l'acier A508 d3 irradié.

essai sur éprouvette CT en cellule chaude

planning 74/76

d/ Préciser le comportement en fatigue plastique des matériaux utilisés dans les réacteurs à eau.

- influence de l'effet de l'environnement en fatigue olygocycle
- essai de fatigue en flexion - en température et en pression en présence d'eau.

planning 73/75

e/ Améliorer la valeur des analyses de tenue à la fatigue des enceintes sous pression de manière à mieux apprécier la sûreté sous l'action des chargements olygocycliques.

Les essais ont pour but de préciser les marges de sécurité obtenues dans le cas de l'application des recommandations du code ASME pour l'analyse à la fatigue.

efforts cycliques - de pression
- de flexion sur les piquages

en cours.

f/ Evaluer l'imprécision de la mécanique de la rupture en cas de grande déformation.

analyse de structures fissurées de géométrie simple

en cours.

3/ Contrôles non destructifs

Amélioration des méthodes de contrôle non destructif, dans le cadre des inspections périodiques des circuits primaires de réacteur, en particulier en vue d'une meilleure définition des paramètres caractéristiques des défauts décalés.

Moyens mis en oeuvre :

a / Ultrasons :

- mise au point de traducteurs à haute résolution permettant :
  - le contrôle au travers du revêtement inoxydable des cuves
  - le dimensionnement correct des défauts
  - le contrôle des soudures d'acier inoxydable austénitique
- évaluation de l'holographie acoustique : comparaison de ses possibilités avec les méthodes précédentes.

.../...

b/ <u>Courants de Foucault</u> :

Amélioration des procédés actuels en vue des inspections périodiques des tubes d'échangeurs (générateurs de vapeur).

c/ <u>Traitement des données</u> :

Mise au point de procédés permettant une évaluation rapide des résultats d'inspections périodiques.

<u>Etat d'avancement</u> :

- <u>Ultrasons</u> :
  - les traducteurs sont actuellement au point. Il reste à qualifier la méthode sur défauts réels et à comparer les dimensionnements obtenus par ultrasons et par micro ou macrographie.
  - en cours les développements pour le contrôle des soudures d'acier inoxydable austénitique, la détection des défauts mal orientés, l'holographie acoustique.

- <u>Courants de Foucault</u> :

  développement en cours de sondes spéciales et de l'électronique modifiée.

- <u>Traitement des données</u> : en cours

d/ Utilisation des techniques de contrôle par émission acoustique pour la surveillance en service des cuves.

- but : développement d'un système complet de surveillance en service avec localisation des défauts des cuves

  appareillage opérationnel

  1969/76

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131-11-10/4111-10

Classification : 11-2

1247

Titre	Pays :
Evaluation de la probabilité de rupture d'une cuve de réacteur à eau	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais)	**Organisme exécuteur :**
Failure probability calculation of a water reactor pressure vessel.	CEA+EURATOM+FRAMATOME
	Responsable : J. DUFRESNE (DSN-SETSSR)
Date de démarrage : 01/01/76 — Date prévue d'achèvement : 31/12/77 Etat actuel : En cours — Dernière mise à jour : 18/11/76	Scientifiques : Répartis dans différents laboratoires.

Objectif général :

Evaluation de la probabilité de rupture d'une cuve de réacteur à eau.
Comparer cette probabilité avec la probabilité de rupture des canalisations du circuit primaire.
Identifier le poids relatif des différents paramètres intervenant dans la rupture d'une cuve.
Rechercher l'emplacement des ruptures les plus probables.

Objectifs particuliers :

Relever des dimensions de défauts dans les cuves de réacteur. Rassembler les données sur le comportement des matériaux. Définir les conditions de fonctionnement programmées et accidentelles. Valider les lois de la mécanique des ruptures. Valider les lois de propagation des fissures. Valider les critères de rupture. Etablir un code général à partir d'histogrammes. Etudier la propagation des fissures dans des conditions particulières (défauts réels, mode II et III, propagation sous eau, à basse fréquence).

Installations expérimentales et programme :

Travaux effectués avec le concours d'EdF et de Framatome, ainsi que, par contrats, avec d'autres industries et universités. Les calculs et mise en oeuvre de codes sont effectués au CCR Ispra.

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Etat de l'étude :

1) Avancement à ce jour :

Recensement des formules de calcul des coefficients d'intensité de contrainte. Recherche sur les résultats publiés de ruptures fragiles sur éprouvettes. Recherche bibliographique sur les lois de propagation des fissures. Analyse des données statistiques sur les ruptures de récipients conventionnels. Analyse des différentes méthodes d'évaluation probabiliste de ruptures de cuves. Recueil de données sur la propagation de défauts dans des récipients conventionnels en cours d'exploitation. Définition des principales situations auxquelles est soumise la cuve. Modification du code COVAL (EURATOM Ispra) pour tenir compte du degré de corrélation de plusieurs variables.

Résultats essentiels :

La formule de calcul du coefficient d'intensité de contrainte établie par Shah et Kobayaski est pessimiste de 15 %. La dispersion des résultats est de ± 10 %.

Prochaines étapes :

Validation des critères de rupture. Propagation des défauts réels. Recueil de données sur les défauts de cuve. Mise en place du code de calcul.

Documents de référence :

"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, 15-3-76/15-9-76", J.DUFRESNE - Rapport DSN 116.

131-11-22/4111-02

Classification : 11-2

1249

Titre	Pays :
Développement des méthodes de calcul pour l'analyse mécanique des circuits primaires des réacteurs à eau.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Advances in computation methods for the mechanic analysis of PWR primary systems.	Organisme exécuteur : CEA/DEMT
	Responsable : B. BARRACHIN (DSN-SETSSR) R. ROCHE (DEMT)
Date de démarrage : 1/01/76 Date prévue d'achèvement : 31/12/81 Etat actuel : en cours Dernière mise à jour : 1/1/77	Scientifiques :

Objectif général :

Cette action vise la tenue mécanique du circuit primaire principal des réacteurs nucléaires à eau sous pression.

Objectifs particuliers :

- Amélioration de la définition et de la validation des critères utilisés dans les codes de construction.
- Calculs d'éléments de structure
- Analyse des règles du code ASME section III
- Développement des méthodes de calcul des structures mécaniques
- Application et développement de la mécanique de la rupture.

Etat de l'étude :

1) Avancement à ce jour :

- Mise au point de codes de calcul 2D et 3D (option plasticité)
- Application des structures de PWR et vérification des analyses effectuées par le constructeur
- Etude critique du domaine de validité de la LEFM

2) Résultats essentiels :

- Analyse détaillée du comportement des tubulures de sortie d'une cuve PWR 900.
- Validité de la LEFM. Comparaison avec les méthodes élasto-plastiques (Intégrale de RICE)

Prochaines étapes :

- Analyse détaillée de la fermeture d'une cuve PWR 900
- Instabilité plastique des composants : vérification des coefficients de sécurité.
- Validité de la LEFM pour des structures et changements complexes.

131-11-40/4111-10 Classification : II-2

1251

Titre	Pays :
Etudes de sûreté des composants du circuit primaire des réacteurs PWR.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais)	Organisme exécuteur : CEA/DSN - SETSSR
Safety analysis of PWR primary circuit components.	Responsable : B. BARRACHIN (DSN - SETSSR)
Date de démarrage : 01/01/77 Date prévue d'achèvement : 31/12/79 Etat actuel : A lancer Dernière mise à jour : 15/12/76	Scientifiques :

Objectif général :

Ces études ont pour but de préciser le comportement des composants du circuit primaire des réacteurs PWR en conditions accidentelles.

Objectifs particuliers :

Comportement de l'acier de cuve en milieu pressurisé influence du beurrage.
Interprétation des règles ASME en conditions accidentelles.
Comportement de composants particuliers en conditions accidentelles.

Titre	Pays : FRANCE
Recherche des procédures simplifiées pour l'évaluation des facteurs d'intensité de contrainte dans les enceintes sous pression.	Organisme directeur : CEA
Titre (anglais) Research of simplified procedures for the evaluation of stress intensity factor in the pressure vessel.	Organisme exécuteur : GAAA Responsable : B. BARRACHIN
Date de démarrage : 01/10/76 Date prévue d'achèvement : 01/09/77 Etat actuel : en cours Dernière mise à jour : 19/11/76	Scientifiques : S. BANDHARI

Objectif général :

Cette étude à pour but d'élaborer des procédures simplifiées pour l'évaluation des facteurs d'intensité de contraintes pour des fissures dans des ouvertures renforcées sous divers chargements.

Objectifs particuliers :

Application à la cuve d'un réacteur PWR.

Etat de l'étude :

Avancement à ce jour :

Démarrage. Etude bibliographique des cas bidimensionnels et tridimensionnels.

1255

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 **Organisation** University of Pisa
Date initiated 1968 Date completed 1980 Last updating May 1977	**Project Leader** 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.

1256

Title 1 (Original language)	Classification
Sull'inizio di propagazione di una rottu- ra in un recipiente in pressione costrui- to 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 propaga-zione 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 tempe-ratura 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 cri-tico del materiale, mediante prove di scoppio di modelli cilindrici fessurati.
Atti del IV Convegno Nazionale AIAS, Roma, Sept. 1976.

1257

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
Title 2 (English) Experimental research on incremental collapse and plastic fatigue of piping components.	Country ITALY Sponsor CNEN Organisation University of Pisa
Date initiated 1976 Date completed 1980 Last updating May 1977	Project Leader 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).

1258

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.

Title 1 (Original language)	Classification
Fatica ad alta temperatura di elementi struttu-rali	11.2
Title 2 (English)	**Country** ITALY
High Temperature Fatigue on Structural Elements	**Sponsor** CNEN **Organisation** University of PISA
Date initiated 1976 **Date completed** 1980 **Last updating** May 1977	**Project Leader** 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).

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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 St c tures.
4th International SMIRT Conference - San Francisco (1977)

12.61

Rotterdam Dockyard and others	CLASSIFICATION: 11.2
TITLE: Breukanalyse Onderzoek aan Stompen (BROS)	COUNTRY: NETHERLANDS. SPONSOR: Ministry of Economic Affairs ORGANIZATION: Rotterdam Dockyard and others
TITLE: (ENGLISH LANGUAGE): Fracture analysis research on nozzle intersections (BROS)	PROJECTLEADER: C.J. Drijver
INITIATED: March 1972 — LAST UPDATING: Febr. 1976 STATUS: in progress — COMPLETED: End 1977	SCIENTISTS: M.J.G. 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 cracks occurring 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.
- Acoustic Emission testing has been completed
- Fatigue and fracture (J_{IC}-) tests on standard specimens are in progress.
- The bi-axial loading bench is under construction.

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Next steps

The next steps will be the testing of flat plate test pieces using the bi-axial loading bench and the publishing of the final technical reports.

Relation with other projects

Results of this project will be used within the "EPOSS" project.

Reference documents

A series 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 Ministry of Economical Affairs, Laan van Nieuw-Oost Indië 123, The Hague. Requests for obtaining copies should be sent to this address.

Budget

ƒ 6.10^6

Personnel

46 manyears

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TH-Delft	CLASSIFICATION: 11.2
TITLE: Elastisch-plastisch breukmechanica onderzoek aan stompen van nucleaire hoge-drukvaten (EPOSS)	COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs ORGANIZATION: TH-Delft
TITLE: (ENGLISH LANGUAGE): Elastic-plastic fracture analysis research on cracks at nozzle intersections of nuclear high-pressure vessels (EPOSS)	PROJECTLEADER: Latzko
INITIATED: August 1974 — LAST UPDATING: May 1977	SCIENTISTS: V.d. Ruijtenbeek, Broekhoven, Peeters
STATUS: in progress — COMPLETED: 1977 (scheduled)	

General aim

Crack extension behaviour in heavy section nuclear steel pressure vessels in areas of complicated geometry.

Particular objectives

Evaluation of the applicability of the J-integral concept (an elastic-plastic fracture mechanics concept) for predicting elastic-plastic crack extension for complex crack configurations in nuclear pressure vessels, notably cracks in nozzle corner regions.

Experimental facilities and programme

Main activities are :

a. theoretical investigations

* Computation of J-integral values by the finite element method for 2-dimensional configurations, ranging from simple test specimens to uniaxially loaded plates with cracks emanating from a central hole;
* Computation of J-integral values by the finite element method for some 3-dimensional configurations, viz.
  - bars with a quarter-circular edge crack
  - uniaxially loaded plates with quarter-circular cracks emanating from a hole
  - flat plates with a central nozzle and a crack at the nozzle corner
* Evaluation of the applicability of simplified approximation procedures to determine J for said configurations.

b. experimental investigations

* J_{Ic}-tests on standard specimens for the model material, i.e. Al 2024-T3
* Model-tests on 2-dimensional configurations, i.e. uniaxially loaded plates with cracks emanating from a central hole
* Model-tests on the 3-dimensional configurations mentioned under a.

Project status

* Procedures for efficient computation of J-values by the finite element method have been established
* Computations of J-values for simple 2-dimensional configurations have been completed
* Computations of J-values for more complicated 2-dimensional configurations are in progress
* J_{Ic}-tests on standard specimens are in progress
* Experimental investigations on uniaxially loaded plates with cracks emanating from a central hole are in progress

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Next steps
Continuation of investigations of 2-dimensional configurations; initiation of theoretical and experimental investigations on 3-dimensional configuration

Relation with other projects
The study is an extension of the BROS-project into the elastic plastic regime

Reference documents
Report MMPP-110, Delft Un. of Techn., Lab. for Thermal Power and Nucl.Engineering

Degree of availability
Through Ministry of Social Affairs, C.R.V., Postbox 69, Voorburg, Netherlands

Budget
Approx. Dfl. 750.000

Personnel
100 Manmonths.

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Classification: 11.2.1	
Title 1 (Original Language): Bestimmung bruchmechanischer Sicherheitskriterien für elastisch-plastisches Werkstoffverhalten (RS 90 A-II.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen AEG HE/IMU Ffm.
Title 2 (English): Determination of Fracture Mechanical Safety Criteria for Elastic-Plastic Behaviour of Materials	**Project Leader:** Dr. Klausnitzer Prof. Detert
Initiated (Date): 1. 1. 76 **Status:** Completed **Completed (Date):** 31. 12. 76 **Last Updating (Date):** 31. 12. 76	

1. General Aim

For the collection of safety criteria with respect to the fracture due to elastic-plastic material behaviour the steel 22 NiMoCr 37 was investigated by the methods of fracture mechanics.

2. Particular Objectives

With the COD-concept, the J-Integral concept and the concept of resistance curves the methods of the linear elastic fracture mechanics were tried to be extended on the elastic-plastic material behaviour.

3. Research Program and 4. Test Facilities

In continuation of the research program RS 90 tests were carried out with big plates which were loaded until fracture. The experimental data were compared with analytical calculations in order to refine the calculation methods.

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5. Progress to Date

The plates with a hole in the center got a mechanical crack, which was prolonged by oszillation technique. When the plates were instrumented the tests began at room temperature. Strain, COD, accustic emission and forces were measured.

The results were evaluated.

6. Results

The evaluation of the results is still under work. The determination of the fracture force by the concept of resistance curver has shown good agreement with the results of the tests on small specimens.

7. Next Steps

Work has been completed.

8. Relation with Other Projects

RS 90 Determination of Fracture Mechanical Safety Criteria for Elastic-Plastic Behaviour of Materials

9. References

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10. Degree of Availability

1267

Classification: 11.2.1	
Title 1 (Original Language): Schweißversuche zum Plattieren von Reaktor-Druckgefäßen (RS 91 - II.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Welding Tests on Cladding of Reactor Pressure Vessels	Project Leader: Dr. Klausnitzer, Prof. Dr. Detert
Initiated (Date): 1. 12. 72 **Status:** Continuing **Completed (Date):** 31. 7. 77 **Last Updating (Date):** 31. 12. 76	

1. General Aim

It is the aim of the present work to examine the influence exerted by different parameters on residual stresses using a suitable measurement method.

2. Particular Objectives

In order to protect the inner surface of reactor pressure vessels against corrosion, band or strip welding procedure is used for cladding the ferritic vessel with austenitic stainless steel. During cladding high residual stresses occur in thick-walled components. In addition, the different thermal expansion coefficients of stainless steel and ferritic steel induce further residual stresses. These stresses cause under-clad-cracks during postweld heat treatment. Therefore, the knowledge of the magnitude and of the distribution of residual stresses after welding and after postweld heat treatment are of special interest.

3. Research Program and 4. Experimental Facilities

Cladding was made on A 508. The dimension of the plates was about 125 mm thick, 240 mm wide and 300 mm long. The thickness of the austenitic cladding was about 5 mm. The following measurement method was used: strain gauges were placed on the surface of the

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ferritic side of the plate and the cladded, opposite side was machined in steps. The strain response to the removal of the material was measured using strain gauges. These strains are used for compensating the stress distribution in cladding direction of the plate in dependance of the plate thickness.

5. Progress to Date

On a specimen with the bore hole method residual stresses were measured. The weld material, heat influence zone and basic material were studied. The instrumentation was specially arranged in order to measure the stress parallel and vertical to the connection weld.

The bore hole method was extended from 5 mm depth to 25 mm. Several methods were tested to get a smooth calibration function.

6. Results

First attempts were made to approximate the calibration curves of the extended bore hole method with analytic functions of second or third order. There were inconstancies however from one kernel to another. Therefore Fourier-approximations will be used.

7. Next Steps

The Fourier-method will be tested on austenitic cladding materials

8. Relation with Other Projects

RS 101 Reactor Pressure Vessel Urgent Program 22 NiMoCr 37

9. References

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10. Degree of Availability

1269

Classification: 11.2.1	
Title 1 (Original Language): "Forschungsprogramm Reaktordruckbehälter" - Dringlichkeitsprogramm 22 NiMoCr 37 - (RS 101, RS 101A - II.2, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** MPA Stuttgart
Title 2 (English): Reactor Pressure Vessel - High Priority Program 22 NiMoCr 3 7 - (RS 101, RS 101A, Annual Report A 76)	**Project Leader:** Prof. Dr.-Ing. E. Krägeloh Prof. Dr.-Ing. H. Uetz
Initiated (Date): December 1, 1972 **Status:** partly completed	**Completed (Date):** December 31, 1977 **Last Updating (Date):** December 31, 1976

1. GENERAL AIM

Determination of possible critical conditions in welding seams (strength seams) and their effect on strength or safety against fracture with the objective of optimizing the material choice and the parameters during welding and heat treatment (see also Annual Report IRS-F-18).

2. PARTICULAR OBJECTIVES

Thorough metallographic as well as electron microscopic investigations have confirmed that intercrystalline micro cracks can develop in the coarse grain zone of weldings (strength seams) in the material 22 NiMoCr 37 (A 508 Cl 2) during heat treatment after welding - especially during stress-relief heat treatment. While the reason for crack formation can be explained by the relaxation embrittlement in the range of the stress-relief heat treatment temperature, the question is, how much such coarse grain zones can influence the strength and toughness of the structural component. The Urgent Program 22NiMoCr 37 is supposed to compile the necessary documents for a safety analysis for reactor pressure vessels in accordance with the specific material crack conditions.

As a way towards solutions parameter studies have been considered, which can be divided in tests with small and large specimens as well as "component tests". In

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addition to the conventional simulated welding treatments, which can only be carried out on small specimens, simulated tests on large specimens have been made. Electron beam welding has been used for the creation of geometrically defined HAZs.

Parallel to the experimental tests it is tried to determine temperature fields and heat stresses theoretically.

3. RESEARCH PROGRAM

The research program has been classified in the following nine partial projects:

a) intermediate vessels (ZB 1)
b) welding of discs into reactor pressure vessel ring (S-UP-Ro)
c) simulated specimens - tests on the welding simulator (Si-SSi)
d) simulated specimens - annealing heat treatment and relaxation tests (Si-G/R)
e) simulation with electron beam (Si-E-beam)
f) submerged-arc-welding, circumferential seam (S-UP-RN)
g) submerged-arc-welding, longitudinal seam (S-UP-LN)
h) deformation and fracture mechanism (VBM)
i) device for the testing of large specimens (10 000 Mp Project)

4. EXPERIMENTAL FACILITIES, COMPUTER CODES

See Annual Report IRS-F-24.

5. PROGRESS TO DATE

The projects ZB 1, S-UP-Ro, Si-SSi, Si-G/R, Si-E beam, S-UP-RN, S-UP-LN, and VBM have been completed by composing the third Technical Report.

10 000 Mp Project

Completion of further small parts, construction and production of auxiliary devices and facilities for the machine, planning of the installation. Construction works for the testing hall (RS 101 A) have been started.

6. RESULTS

See Annual Report A 75.

7. NEXT STEPS

10 000 Mp Project:

Completion of the foundation. Mounting of the clamping device.

8. RELATION WITH OTHER PROJECTS

See Annual Reports IRS-F-18

9. REFERENCES

Quarterly Reports in IRS Research Reports, period reported on:

January 76 - December 76 (German)

Annual Report A 76 (English).

10. DEGREE OF AVAILABILITY OF THE REPORTS

Upon request at BMFT.

1273

Classification: 11.2.1	
Title 1 (Original Language): "Bestrahlung von Druckgefäßstählen im Rahmen des Koordinierten Programmes der IAEA" (RS 85 - II.2, Jahresbericht A76)	COUNTRY: BRD SPONSOR: IAEA ORGANIZATION: GKSS, Geesthacht
Title 2 (English): Coordinated Programme on Irradiation Embrittlement of Pressure Vesse Steel	**Project Leader:** Dr.-Ing. F. J. Schmitt
Initiated (Date): April 1973 **Status:** finished **Completed (Date):** 1976 **Last Updating (Date):** December 1976	

1. General Aim

The purpose of the programme is to promote coordinated research in the field of irradiation embrittlement of reactor pressure vessel steels, at an international level. The results obtained will give a possibility for a more precise interpretation of individual surveillance results.

2. Particular Objectives

Eight institutes in seven nations are participating in this programme whose basic goal is to establish that the bases for describing embrittlement, measurement of the neutron spectrum and fluence on mechanical properties (on a typical-standard reference-plate of steel) were sufficiently standardized to permit direct intercomparison among international programmes and to compare embrittlement sensitivity of national steels with the standard steel.

3. Research programme

Samples of the pressure vessel steel ASTM A533 B Class 1 (Plate 03 of the HSST-Programme) were irradiated and different materials tests were carried out, to study the influence of neutron dosis and irradiation temperature on the mechanical properties.

4. Experimental Facilities

Further Charpy-V-notch and tensile specimens were neutron irradiated in several cap-

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sules in the Material Test Reactor FRG-2 in Geesthacht:

a) Irradiation of Cv- and tensile specimens at a temperature of 290 °C to a dose of $4{,}0 \cdot 10^{18}$ nvt, E> 1MeV.

b) Irradiation of Cv- and tensile specimens at temperatures between 40 °C and 75 °C to doses of $2{,}9 \cdot 10^{18}$; $1{,}6 \cdot 10^{19}$ and $1{,}0 \cdot 10^{20}$ nvt.

c) Irradiation of fracture mechanics specimens WOL-1X at 290 °C and to doses of $2{,}2 \cdot 10^{19}$ and $8{,}5 \cdot 10^{19}$ nvt.

The mechanical tests were conducted in the Hot Cells of the GKSS.

5. Progress to Date

All specimens have been irradiated, some were postirradiation annealed and the following mechanical tests were carried out in the Hot Laboratory:

- Impact test at the whole transition region and determination of the transition-temperature, shear fracture, lateral expansion.
- Tensile test (tensile strength, yield strength, elongation, reduction of area.)
- Fracture toughness test.

6. Results

6.1 Irradiations at 290 °C and $4{,}0 \cdot 10^{18}$ nvt

Impact tests

The shift of the NDT-Temperature for transverse specimens is
Δ NDTT = 22 °C (41 J - criterium)
" = 20 °C (50 % filsons fracture)

The lateral expansion of 0,9 mm was reached at 50 °C.

The shift of the NDTT for longitudinal specimens is Δ NDTT = 20 °C (41 J)
" = 8 °C (50 % fibrous fracture).

The lateral expansion of 0,9 mm was reached at 34 °C.

Tensile tests

Increase of the tensile strength at room temperature from 617 N/mm^2 to 632 N/mm^2.
Increase of the yield strength from 458 N/mm^2 to 489 N/mm^2.

6.2 Irradiations between 40 °C and 75 °C

a) Dose of $2{,}9 \cdot 10^{18}$ nvt

Impact tests

Δ NDTT = 60 °C (transverse specimen)
" = 68 °C (longitudinal ") 41 J.

Δ NDTT = 40 °C (transverse specimen)
" = 31 °C (longitudinal ") 50 % fibrous fracture

Tensile tests

Increase of σ_{UT} from 617 N/mm^2 to 636 N/mm^2
Increase of σ_y from 458 " to 598 "

b) Dose of 1,6 . 10^{19}nvt

Impact tests

Δ NDTT = 113 °C (transverse specimen)
" = 140 °C (longitudinal ") 41 J.

Δ NDTT = 82 °C (transverse specimen)
" = 106 °C (longitudinal ") 50 % fibrous fracture

Tensile tests

Increase of σ_{UT} from 617 N/mm^2 to 766 N/mm^2
Increase of σ_y from 458 N/mm^2 to 766 N/mm^2

c) Dose of 1,0 . 10^{20}nvt

Impact tests

Δ NDTT = 182 °C (transverse specimen)
" = 172 °C (longitudinal ") 41 J.

Δ NDTT = 155 °C (transverse specimen)
" = 132 °C (longitudinal ") 50 % fibrous fracture

Tensile tests

Increase of σ_{UT} from 617 N/mm^2 to 912 N/mm^2
Increase of σ_y from 458 " to 912 "

6.3 Irradiation of fracture mechanics specimens

Type: WOL-1X
Irradiation temperature: 290 °C
" doses : 2,2 . 10^{19} and 8,5 .10^{19}nvt

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The temperature shift ΔFMTT reaches values of 54 °C and 70 °C for the two neutron doses
The experimental work has been finished and the results will be evaluated.

7. Next steps

A final report will be prepared.

8. Relation with other projects

See annual reports 1974 and 1975

9. References

a) Quarterly reports in the series IRS-Forschungsberichte RS85, (in German),

b) F.J. Schmitt, Zeitschrift für Werkstofftechnik, 7, (1976), 241-247 (in German)

10. Availability of the Reports

a) is available from the GRS, Cologne, Germany
b) is available from the Verlag Chemie, GmbH.
D-6940 Weinheim, Germany.

1244

Classification: 11.2.1	
Title 1 (Original Language): Untersuchungen zum Verschweißen sowie an Schweißverbindungen dickwandiger Reaktorbauteile (RS 102-01 - II.2, Jahresbericht A 76)	**COUNTRY:** BRD Germany **SPONSOR:** BMFT **ORGANIZATION:** FhG - IFAM
Title 2 (English): Narrow-Gap welding	**Project Leader:** I.D.Henderson
Initiated (Date): 1.5.73 **Status:** Finished **Completed (Date):** 31.12.76 **Last Updating (Date):** December 1976	

1. General Aim

The research program is aimed at evaluating the gas-shielded arc-welding process of Narrow-Gap welding for joining of heavy sections of the fine-grained low-alloyed pressure vessel steel 22 NiMoCr 3 7. The investigations are primarily designed to obtain information regarding the applicability of this newly developed welding process for one-sided welding of thick sections up to 300 mm, and specifically to determine the mechanical and metallurgical properties of welded joints produced in forged sections of the nominated reactor construction steel

2. Particular Objectives

Until the conception of this research project, a welding machine of this type and a welding process of this nature had not been investigated in Europe. To begin the research program it was therefore necessary to set up a laboratory facility and to install the equipment necessary to conduct such an investigation. Following this preparatory work, the welding tests could begin.

Using a range of welding consumables and various welding conditions, preliminary investigations were carried out on 100 mm rolled sections to establish the optimum parameters and electrode compositions for the main tests. Welds were then made in 200 and 300 mm forgings, whereby emphasis was not only placed on the weldability and functionability of the machine but also on the comparison of the mechanical and metallurgical properties of the welded joints in various heat-treated conditions with those properties established for the base material. The welding tests were conducted mainly in the flat position. As a result of increasing interest in the site-welding of heavy sections in the horizontal-vertical position, a preliminary investigation was also done in the three-o'clock position on 50 mm rolled plate. Other special investigations include the measurement of the temperature distribution and thermal cycles in the heat affected zone and welding with non-metallic backing strips to produce a suitable root geometry for supplementary cladding processes.

3. Research Program

The research program was initially drawn up to include welding tests on the steel 20 MnMoNi 5 5 as well as the steel 22 NiMoCr 3 7. As a result of difficulties in procuring the steel 20 MnMoNi 5 5 in small quantities and in sizes required for these welding tests, the investigations on this base material could not be carried out as part of this research program. It was also planned to produce welded joints using other welding processes such as electro-slag, submerged-arc and MMA-processes for the purpose of comparing the results.

However, after considering the state-of-the-art and with the wealth of knowledge which already existed with respect to the conventionally applied welding processes, it was decided to restrict this research program to welding tests using only the Narrow-Gap process and to concentrate on the determination of the influence of the welding conditions on the mechanical and metallurgical properties of the joints. Of particular interest was the effect of preheating and post-weld heat treatment on the weld metal and HAZ properties.

4. Experimental Facilities

The Narrow-Gap welding machine was originally conceived to permit one-sided welding of section thicknesses up to 230 mm. Following the successful completion of the welding tests on 200 mm thick forged sections, modifications were made to the welding torch assemblies and to the work holding system to permit the tests on heavier sections to be done. Fig. 1 shows the welding head arranged for welding of 300 mm thick forgings.

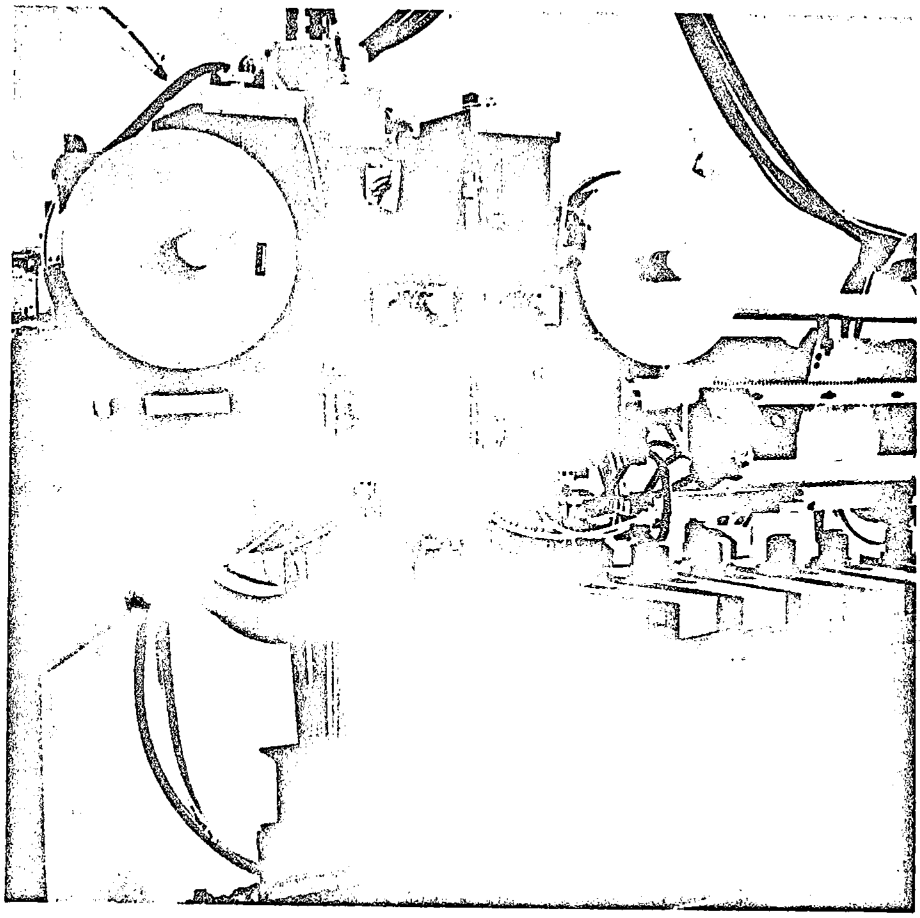

Fig. 1: General view of the Narrow-Gap welding head and 300 mm thick test pieces prepared for welding.

These tests were done using a preheating temperature of 200 to 250 °C. The welded joints were then non-destructively and mechanically tested in both the as-welded and in the stress-relief heat-treated conditions.

Further modification of the machine and the holding system enabled orientation welding tests on 50 mm thick plate in the horizontal-vertical position to be carried out. Fig. 2 shows the arrangement of the test facility for these welding investigations.

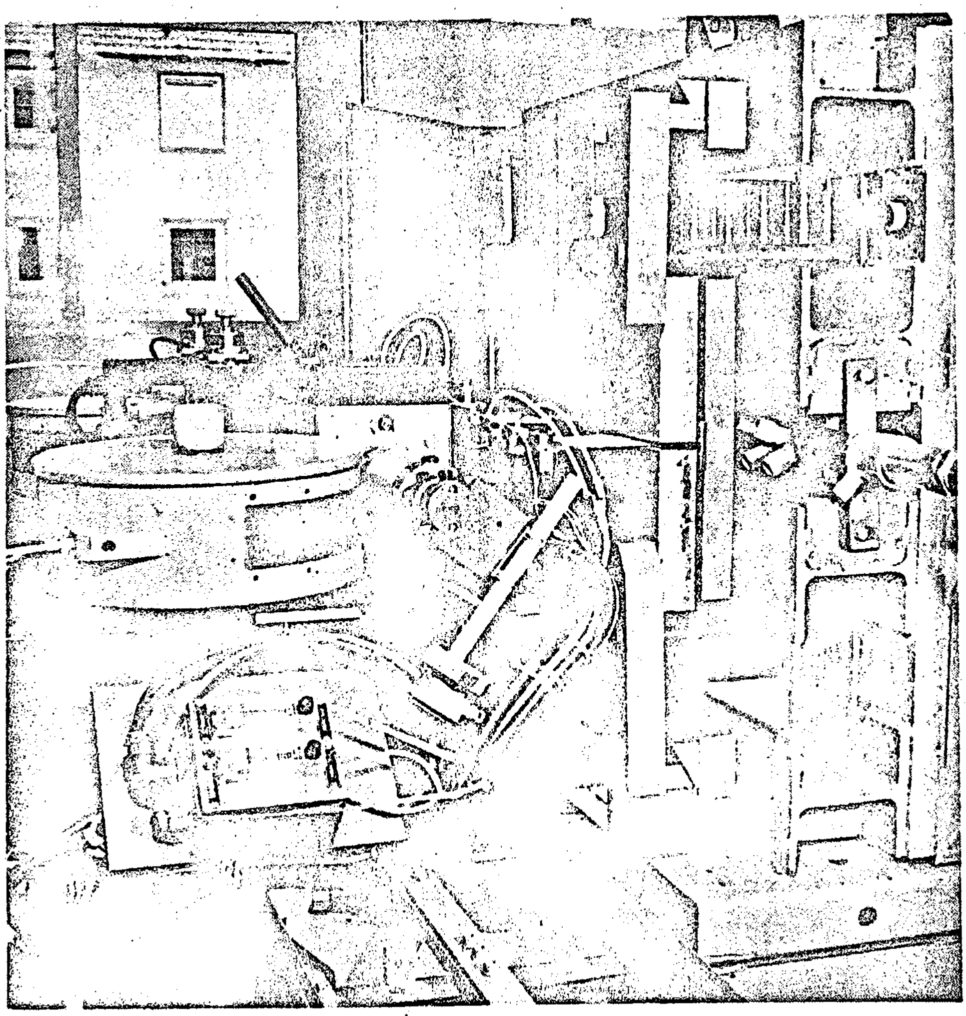

Fig. 2: Narrow-Gap welding of 50 mm test pieces of the steel 22 NiMoCr 3 7 in the three-o'clock position

5. Progress to Date

The extension of the research program period by six months and an increase in the funds by approximately 200 TDM allowed the planned research work to be successfully completed on 31.12.76. The extensive testing of welded joints produced in 200 mm forged sections of the steel showed that in addition to the preheating of the weldments to 200 to 250 °C, a stress-relief heat-treatment at 600 °C is necessary to achieve optimum fracture toughness in the weld metal. The fact that in the as-welded condition the Narrow-Gap weld metal exhibits comparable mechanical porperties to the base material in the forged condition meant that this test condition could not be excluded for the welding tests on 300 mm sections.

6. Results

The non-destructive testing of these joints could not be conducted successfully on full-thickness specimens, so that the x-ray examinations were carried out on smaller segments of the joints. These tests revealed the usual low level of porosity associated with gas shielded arc welding. The mechanical testing and metallurgical examination of the joints simply duplicated the test results established on 200 mm test pieces. Fig. 3 shows a cross-section of a weld seam produced in

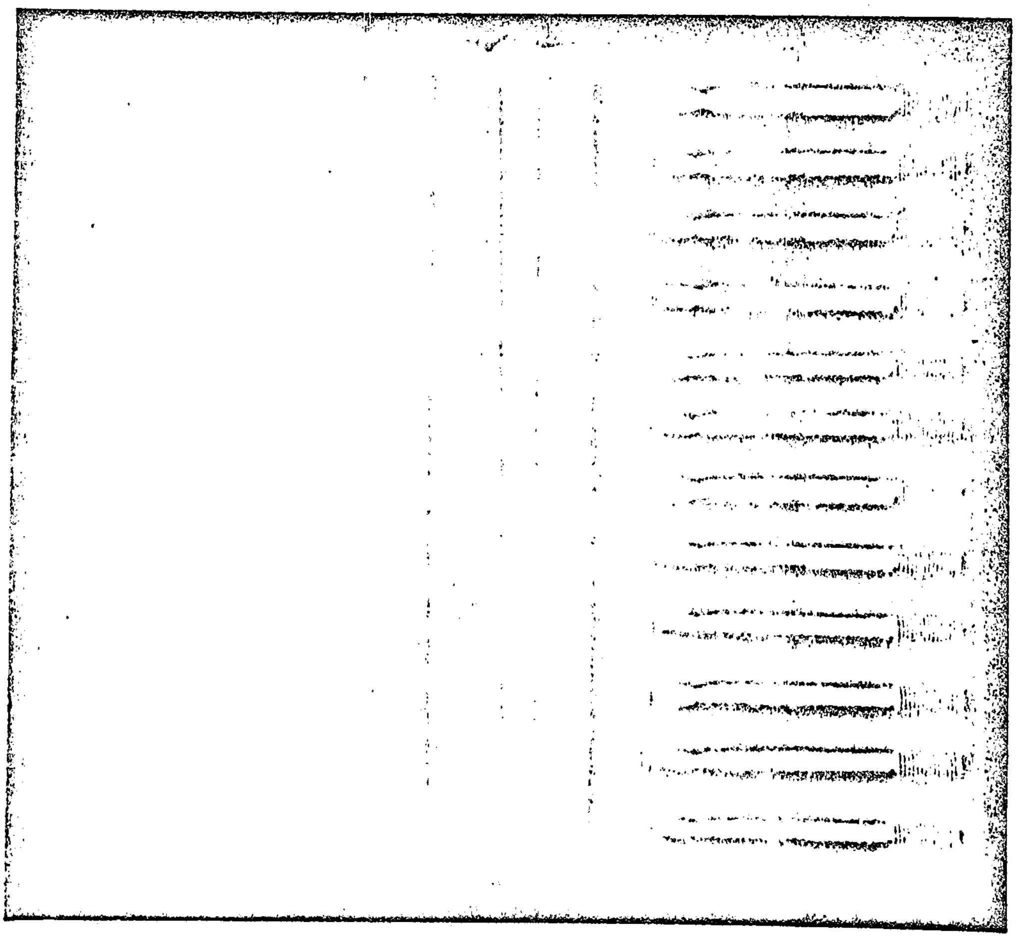

Fig. 3: Section of a Narrow-Gap weld seam in 300 mm thick forgings of the steel 22 NiMoCr 3 7 with tested round tensile specimens in the as-welded (LHS) and stress-relief heat-treated conditions (RHS)

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300 mm thick sections. As a result of an excessive allowance for shrinkage, the weld seam is not uniformly parallel but despite the gap width of 12 mm near the top of the weld, a satisfactory seam geometry with adequate side-wall and centre-line fusion resulted. The tested tensile specimens from the section thickness reveal that in both the as-welded and the stress-relief heat-treated condition, the welded zone is not the critical region with respect to tensile strength. Tensile testing of mini-tensile specimens extracted from the small heat affected zone revealed tensile strengths (0,2-proof stresses) in the HAZ of approximately 660 N/mm^2 in the as-welded condition and between 474 and 620 N/mm^2 in the heat-treated condition. The base material 22 NiMoCr 3 7 has in comparison a yield strength of 478 to 550 N/mm^2 at room temperature.

In Fig. 4 is a cross-section of a Narrow-Gap weld seam produced in the horizontal-vertical position.

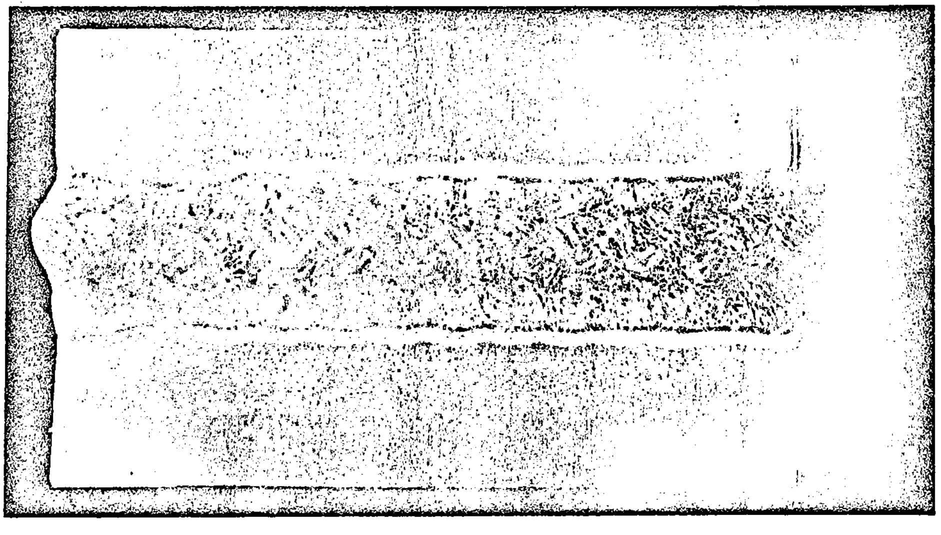

Fig. 4: Section of a Narrow-Gap weld in 50 mm thick plate produced in the three-o'clock position

7. Next Steps

Considering the good results obtained during the Narrow-Gap welding tests on section thicknesses from 50 to 300 mm using a one-sided welding technique it is to be expected that this work will be continued. However, the financing of a continuation project must be met jointly by the industry as well as governmental authorities such as the BMFT.

Following discussions with several representatives of companies involved in pressure vessel construction, the main interest seems to be directed to the evaluation of the Narrow-Gap welding process for welding of girth seams in heavy sections of the pressure vessel steel 20 MnMoNi 5 5 in the three-o'clock position. These investigations would simulate the present welding problem arising as a result of on-site construction and assembly of large pressure vessels for both pressurized water reactors and boiling water reactors.

8. Relation with Other Projects

Associated with the Reactor Safety Program there have been no additional projects approved which relate to Narrow-Gap welding. However, several developments related to this subject are worth reporting:-

a. During a visit to Japan in September 1976 it was shown that considerable emphasis is being placed on the development and application of narrow gap welding techniques for thick section welding.

b. In Italy, a major construction firm is developing a narrow gap welding process for joining of heavy section austenitic and ferritic steels for nuclear applications.

c. Following a FhG-Seminar in IFAM on 14.6.76 several firms in Germany have shown interest in supporting a continuation project to further evaluate the Narrow-Gap welding technique.

d. The committee of the DVS-Working Group UG 30-1 is presently considering ways and means of promoting this research work.

9. References

Henderson, I.D.
K. Seifert

Untersuchung verschiedener Badsicherungen beim Engspaltschweißen eines Reaktorbaustahls.
Schweißen u.Schneiden, Jg.28, Aug.76, H.8

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Henderson, I.D. H.-D. Steffens	Fracture toughness of Narrow-Gap welded joints in the nuclear pressure vessel steel 22 NiMoCr 3 7 Nuclear Engineering and Design Vol. 36, No. 2, Febr. 1976, S. 273-285
Henderson, I.D. A. Ducrot	Narrow-Gap welding of heavy section pressure vessel steel Paper 16-6-1, IIW-1976-MTC, 23.-27.Aug. 1976, Sydney
Henderson, I.D. S. Klingauf H.-D. Steffens	Temperaturfeldmessungen beim Engspaltschweißen des Reaktorbaustahls 22 NiMoCr 3 7 Schweißen u. Schneiden, Jahrgang 28, Nov. 1976, Heft 11.

10. Degree of Availability of the Reports

All reports about this research project are unclassified.

1285

Classification: 11.2.1	
Title 1 (Original Language): Bruchverhalten von teilweise durchgehenden Rissen (RS 102-11 -II.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: IFKM Freiburg
Title 2 (English): Fracture Properties of Part-Through Cracks	**Project Leader:** Dr. E. Sommer Dipl-Ing.H.Kordisch
Initiated (Date): May 1973 **Status:** Finished **Completed (Date):** Oct. 30, 1976 **Last Updating (Date):** December 1976	

1.- General Aim

The general aim is to study the fracture properties of part-through cracks in nuclear reactor components with experimental methods, based on the concepts of fracture mechanics.

2.- Particular Objectives

The extension of part-through cracks in thick-walled plates and tubes under fatigue loading has been investigated. Growth characteristics have been determined by analysing the fracture surfaces. The crack extension in the direction of the semiaxes as well as the crack growth along the whole front has been evaluated and interpreted in terms of the parameters: initial crack geometry,boundary conditions and type of loading.

3.- Research Program

To show the propagation characteristics, artificial markings were induced on the fracture surfaces during a fatigue test by an overload technique. In order to establish the test conditions, a calibration curve da/dN versus ΔK measured on compact specimens is used.

4.- Experimental Facilities, Computer Codes

The fatigue tests were performed on a 1.6 MN servohydraulic machine with an

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internal pressure device giving up to 150 N/mm^2 and the calibration tests on a 0.25 MN servohydraulic machine.

5.- Progress to Date

Growth characteristics have been determined for the extension of part-through cracks under fatigue loading. The part-through cracks were located in plates and tubes of the reactor steel 22 Ni Mo Cr 3 7 (Fig.1). Markings on the fracture surfaces introduced by an overload technique were used to locate the crack front at specified time intervals and hence to evaluate the crack growth. The loading was confined to mode-I and was applied by either simple tension or internal pressure.

6.- Results

A careful evaluation of these fracture surfaces shows the following results (Fig. 2):

1. For small crack depths (a), i.e. ($a/d < 0.5$) crack growth in thickness (d) direction predominates over crack growth in the width direction - as expected from analytical calculations.
2. For large crack depths ($0.6 < a/d < 1$) the converse behaviour occurs - in contradiction to the analysis of the investigated case.

When the relative growth $\Delta L/\Delta a$ - ΔL being the crack growth at the polar angle φ of a crack front; Δa the crack growth at $\varphi = 90^0$ - is plotted versus the polar angle φ, it can be demonstrated that the maximum crack growth is shifted from the centre of the crack front at $\varphi = 90^0$ to smaller angles as the crack depth increases (Fig. 3,4).

In order to explain these results a simple model will be discussed. This takes into account the variable crack resistance along the crack front which arises due to changes in the stress state. The interaction between stress state and material response is again affected by geometry as well as by the degree and nature of loading, the loading rate and temperature.

7.- Next Steps

The project is completed . Final report is forthcoming.

8.- Relation with other projects

This project is closely related to RS 102-12, RS 102-13 and RS 90.

9.- References

- Annual Report A 75 (Series IRS Forschungsberichte)
- E. Sommer, L. Hodulak, H. Kordisch: Growth Characteristics of Part-Through Cracks in Thick-Walled Plates and Tubes. ASME Paper No. 76-PVP-39

10.- Degree of Availability of the Reports

The reports can be ordered from

Gesellschaft für Reaktorsicherheit GRS
5000 Köln - 1, Glockengasse 2

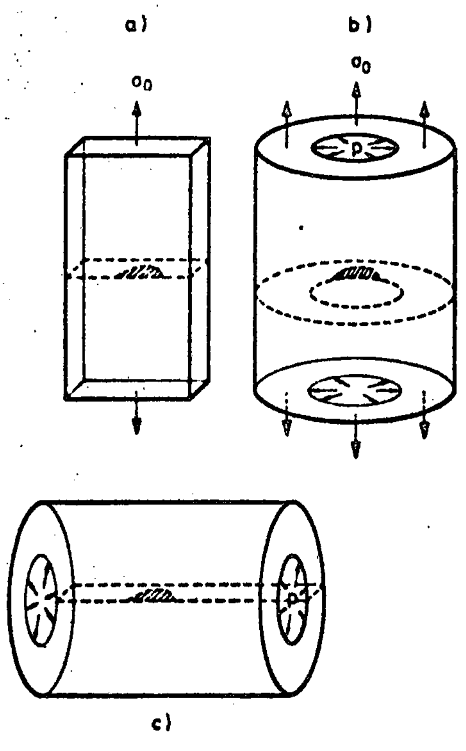

Fig. 1 :

Specimen configurations
a) Part-through crack in a plate;
b) Circumferential crack in a tube
c) Axial crack in a tube

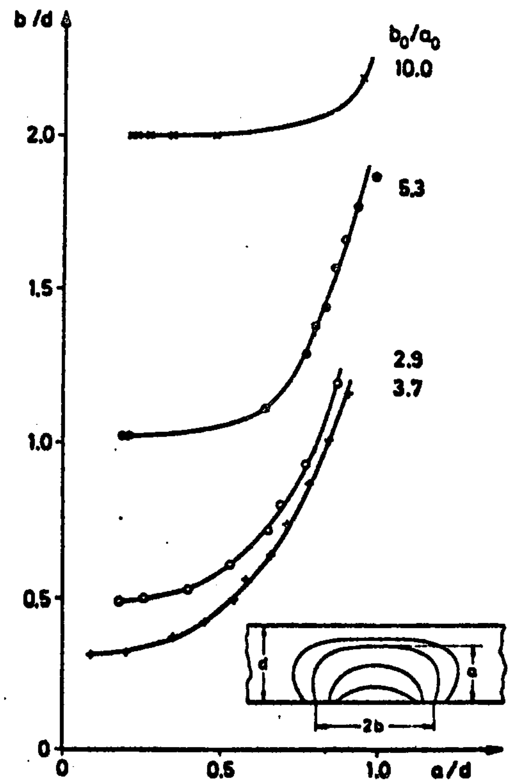

Fig. 2:

Normalized crack extensions in the direction of width b and depth a for different starter notches

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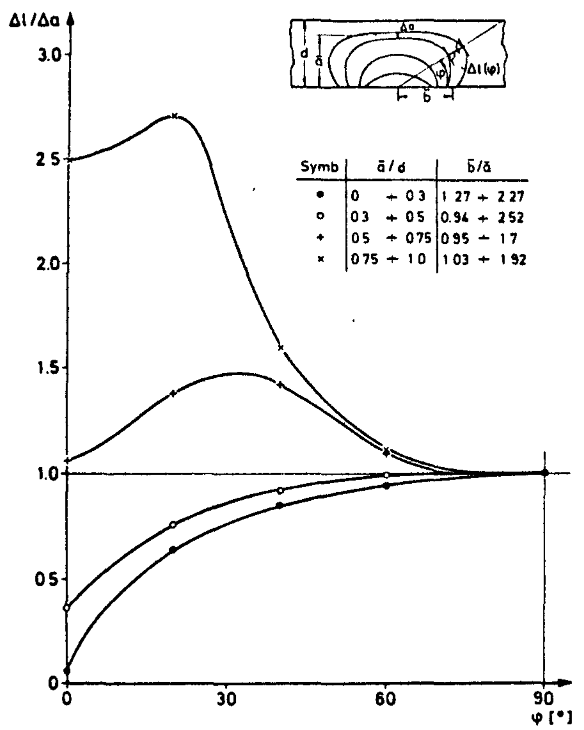

Fig. 3:

Relative crack growth $\Delta l/\Delta a$ versus polar angle φ

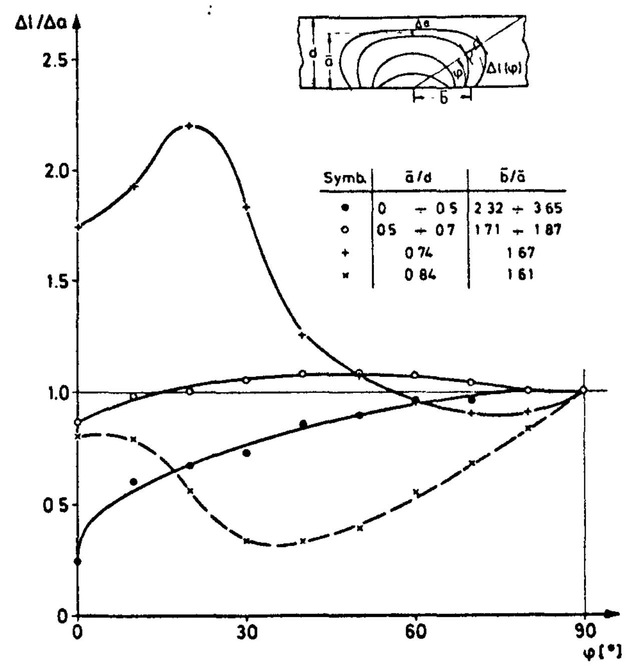

Fig. 4:

Relative crack growth $\Delta l/\Delta a$ versus the polar angle φ for an axial crack in a tube

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Classification: 11.2.1

Title 1 (Original Language): Untersuchungen zum Rißeinsatz und Arrest (RS 102-12 - II.2, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** IFKM Freiburg
Title 2 (English): Investigation of Crack Initiation and Arrest	**Project Leader:** Dr. J.F. Kalthoff
Initiated (Date): 1.5.1973 **Status:** Completed **Completed (Date):** 30.9.1976 **Last Updating (Date):** December 1976	

1. General Aim

The general aim in connection with the reactor safety program is to improve the knowledge of the fracture behaviour of reactor components in which cracks or cracklike defects have to be assumed.

2. Particular Objective

The particular objective is the investigation of crack initiation and crack arrest phenomena. Stress analytical and materialspecific aspects are considered. The tools of analysis are from linear elastic fracture mechnics.

3. Research Program

Cracks are initiated and subsequently arrested in wedge loaded double-cantileverbeam (DCB) specimens. The research program is carried out in two stages:

3.1 In model experiments the principal influence of dynamic effects on the crack arrest process is investigated.

3.2 Experiments with steel (22 NiMoCr 3 7) specimens are performed to measure crack initiation and especially crack arrest toughness values.

In both cases the crack velocities prior to arrest - varying as a consequence of length and sharpness of the starter notches - are measured.

4. Experimental Facilities, Computer Codes

A crack arrest testing device (load capacity 700 KN) has been built and is used together with a Cranz-Schardin high speed camera (24 photographs at a minimum interval time of 1 μs) to measure velocities and stress intensity factors for propagating cracks. Computer codes to evaluate the experimental data, especially to determine static as well as dynamic stress intensity factors and crack arrest toughness values have been developed.

5. Progress to Date

5.1 Applying the shadow spot method dynamic stress intensity factors for propagating and subsequently arresting cracks in DCB-specimens (made from an epoxy resin - Araldit B) have been measured and compared to corresponding static values. From these data fracture toughness values were determined. Several test series for two specimen sizes (321x127x10 mm^3 and 134x53x10 mm^3) in which the crack velocity prior to arrest was varied from 15 up to 300 m/s were carried out.

5.2 Following the guide line of the model experiments several series of steel (22 NiMoCr 3 7) experiments were carried out. DCB specimens of the ordinary or Duplex type - with or without side grooves - (each 25 mm thick) were tested. The material, differently heat treated, was investigated under different crack initiation conditions. The crack surfaces have been investigated metallographically.

6. Results

6.1 The model experiments showed that dynamic effects influence the crack arrest process and have to be taken into account in determining crack arrest toughness values and in applying a crack arrest safety analysis. In particular it was found:

- At the beginning of crack propagation the dynamic stress intensity factor K_I^{dyn} is smaller than the corresponding static value K_I^{stat}, at the end, especially at arrest, K_I^{dyn} is larger than K_I^{stat}.
- Dynamically determined crack arrest toughness values K_{Ia}^{dyn} are larger than the corresponding statically determined values K_{Ia}^{stat}. These K_{Ia}^{dyn}-values do not depend on the crack velocity prior to arrest and were found to be identical for both specimen sizes.

According to these results corrections have to be made to crack arrest safety analyses (ASME Pressure Vessel and Boiler Code Section XI - A 5300) as shown in Fig. 1. Consequences of such dynamic corrections are illustrated in Figs. 2 a and b for two special cases.

6.2 The wedge loading system has been optimized by a low friction insert and allows for an undisturbed symmetric (controlled by means of stress coating techniques) loading of the steel specimens and a faultless measurement of the crack opening to calculate stress intensity factors. Cracks could be initiated and arrested in plane specimens only by hardening of the whole specimen or by using so called Duplex specimens which have an electron beam welded starter region made from an ultra high strength steel. Better results were obtained in side grooved specimens. In all cases the crack velocity could be influenced (although with poor reproducibility) by variation of length and sharpness of the starter notch. Crack arrest was observed after crack jump distances of a few mm up to some cm. Difficulties were encountered in measuring the crack velocities by high speed photography or ladder gauges, velocities now are obtained according to the Hahn-Shmuely crack jump - crack velocity relation. This relation could be verified experimentally. Crack arrest toughness values K_{Ia}^{stat} then were determined for different crack velocities prior to arrest. Metallographic investigations showed, that the different stages of crack propagation in an arrest experiment (i.e. initiation at an overload, acceleration and subsequent deceleration) could be related to characteristic changes in the crack surface appearance (ductile dimple fracture, followed predominantly by cleavage fracture and then distinct ductile fracture behaviour again).

7. Next Steps

The model experiments are completed; some supplementary steel experiments are carried out. The final report is in progress.

8. Relation with Other Projects

This project is closely related to RS 102-11, -13 and -14.

9. Kalthoff, J.F.; Winkler, S.; Beinert, J. "Dynamic Stress Intensity Factors for Arresting Cracks in DCB Specimens", Int. Journ. of Fracture, 12 (1976) 317 ff.

Kalthoff, J.F.; Beinert, J.; Winkler, S. "Auswirkung dynamischer Effekte auf Rißarrest-Sicherheitsanalysen", Reaktortagung 1976 des Deutschen Atomforms e.V., Düsseldorf, 30.3.-2.4.1976

Kalthoff, J.F.; Beinert, J.; Winkler, S. "Einfluß dynamischer Effekte auf die Bestimmung von Rißarrestzähigkeiten und auf die Anwendung von Rißarrestsicherheitsanalysen", 8. Sitzung des Arbeitskreises Bruchvorgänge im DVM, 6./7.10.1976, Köln.

10. Degree of Availability

The reports can be ordered from Gesellschaft für Reaktorsicherheit, Glockengasse 2, 5 Köln 1.

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Fig. 1: Initiation and arrest considerations for a crack in the wall of a pressure vessel

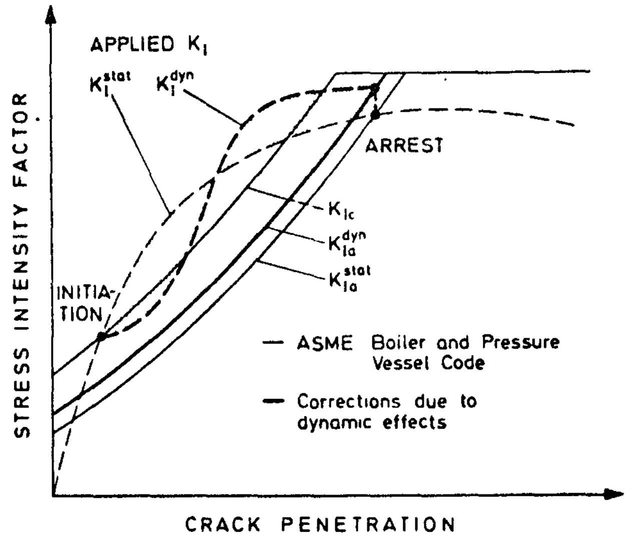
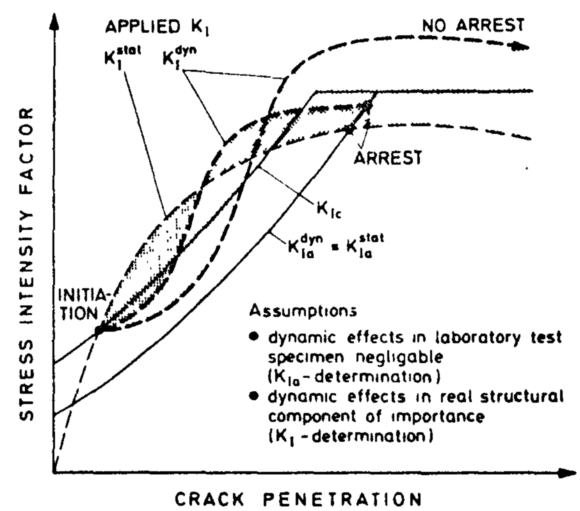


Fig. 2: Consequences of dynamic corrections for crack arrest safety analyses



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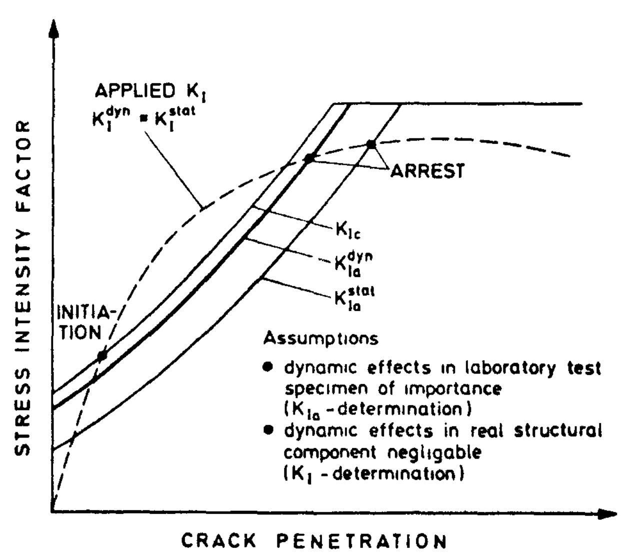

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Classification: 11.2.1	
Title 1 (Original Language): Rißaufweitung (COD) zur Beurteilung der Bruchgefährdung von Bauteilen (RS 102-13 -II.2, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** IFKM Freiburg
Title 2 (English): The Use of COD/COS Fracture Criteria in the Assessment of Component Strength	**Project Leader:** Dr. J.G. Blauel, T. Hollstein
Initiated (Date): May 1973 **Status:** Completed **Completed (Date)** June 30, 1976 **Last Updating (Date):** December 1976	

1. General aim

The general aim is to improve the knowledge on the assessment of safety for components of a reactor plant, in which cracks or cracklike defects have to be assumed.

2. Particular objectives

The particular objective is the assessment of the fracture behavior of components by measuring the crack opening displacement as a function of geometry and load parameters and comparing it to the material specific values COD_c/COS_c.

3. Research program

Measurement of the crack opening displacement COD along the full crack length for Compact (C) and Single -Edge -Notched (SEN) specimens with different thicknesses and crack lengths up to fracture. The load displacement curves are also evaluated in terms of the energy integral J. Two modifications of the steel 22 Ni Mo Cr 3 7 are investigated.

4. Experimental Facilities

Servo-hydraulic testing system, mechano-electrical, photoelectrical and photographical equipment for the measurement of COD, scanning electron microscope for the investigation of fracture mechanismes.

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5. Progress to Date

The crack tip opening displacement COS and the J-integral have been determined experimentally for increasing loads for series of C-specimens and SEN-specimens with different crack langths and thicknesses at room temperature. Critical values of COS, G and J of the material at fracture have been deduced and their correlations have been examined.

6. Results

6.1 For the material, geometry and loading conditions investigated values of COS were determined in an exact and reproducible way by direct optical measurement at the specimen surface. At the same time these values could be inferred from a single clip gage displacement measurement at the open end of the specimen on the basis of a plastic hinge mechanism with a load dependent center of rotation. As compared to other existing models a description is achieved which is the same and conservative over the wohle load range for all thicknesses and crack lengths of one type of specimen. Critical values $COS_{Ic}=0.26 \pm 0.05$ mm and 0.39 ± 0.05 mm, respectively, were determined for the two materials at the onset of stable crack extension (see next paragraph).

6.2 J-values were determined from C-specimen tests using the compliance method of Begley and Landes (1972) and the single specimen procedure of Rice, Paris and Merkle (1973). For these two methods a close correspondence was found only if a variable factor of proportionality $2.3 \geq N \geq 1.9$ for relative cracklengths $0.47 \leq a/W \leq 0.70$ could be assumed (see Sumpter and Turner (1975) and Kuna (1976)). The method of interrupted loading with heat tinting and the electrical potential method were used to generate crack resistance ($J - \Delta a$) curves allow· the determination of the onset of stable crack extension as a safe limit prior to attainment of maximum load and complete fracture. From extrapolations $\Delta a \rightarrow o$ critical values of the J-integral $J_{Ic} = 0.20 \pm 0.04$ MN/m and 0.28 ± 0.06 MN/m, respectively, were determined. A possible dependence on the crack length is indicated only for the 75 mm specimens but together with a change in fracture mechanism.

6.3 A detailed scanning electron microscope study of the fracture surfaces showed that first stable crack growth was preceded - at still lower J-values - by a blunting of the crack tip region through formation of a stretch zone. This localized tearing process is strongly influenced by microstructural inhomogenities (e.g. the flat rolled Mn-sulphides in the less tough of the two materials). Taking that into account by a blunting line intersecting the resistance curve changed the critical values only within the existing scatter band.

6.4 A linear relationship is shown to exist between consistantly determined values of COS and J for situations even beyond onset of stable crack growth. The term of proportionality is material dependent and includes the yield stress and a plastic constraint factor which amounts to 1.5 ± 0.2 and 1.1 ± 0.2 for the two steel modifications.

7. Next steps

The project is completed.

8. Relation with other projects

This project is closely related to RS 102-11, RS 102-12 and RS 90.

9. References

Final report (in German) IFKM 4/76

Hollstein, T.; Blauel, J.G.; Urich, B.: Rißaufweitung COD/COS zur Beurteilung der Bruchgefährdung von Bauteilen aus 22 Ni Mo Cr 3 7 Reaktortagung 1976, 813-816

10. Degree of availability

The reports can be ordered from

Gesellschaft für Reaktorsicherheit GRS
5000 Köln - 1, Glockengasse 2

Classification: 11.2.1	
Title 1 (Original Language): Stoßwellen als mögliche Schadensursachen in Kernkraftwerkskomponenten - Experimentelle Spannungsanalyse - (RS 102-14 - II.2, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** IFKM Freiburg
Title 2 (English): Damage of Nuclear Reactor Components by Shock Wave Loading	**Project Leader:** S. Winkler
Initiated (Date): 1.5.1973 **Status:** Completed **Completed (Date):** 30.4.1976 **Last Updating (Date):** December 1976	

1.- General Aim

In addition to static or quasistatic and thermal loading of reactor pressure vessels and components - which are very well investigated - extremely dynamic loading conditions (shock waves) may occur, for example by impinging fragments or projectiles, by internal explosions or - after certain cases of partial failure - by impact of the accelerated reactor pressure vessel against the containment. The investigation of damage in these dynamic cases is general aim of this work.

2.- Particular Objectives

The particular objective is the investigation of (1) the distribution of the produced dynamic stress fields (shock waves), (2) the interaction of these shock waves with the material, and (3) the initiation of damage in the material.

3.- Research Program

In this program plane shock waves are produced by impinging a target with a flat impactor in a gas gun. Reflected shock waves cause tension fields in the target by which microcracks are nucleated and subsequently activated to grow further, thus leading to internal damage in the material. Crack initiation thresholds and crack growing should be a function of material structure, history and temperature. The research program therefore consists of:

3.1 Investigation of materials with different crack nuclei and of the influence of material structure (texture), heat treatment and temperature during loading on crack initiation.

3.2 Measurement of the change in conventional material strength properties for such materials, containing damaged regions which are produced by impact.

3.3 Measurement and/or calculation of stress pulse profiles in order to understand their interaction with the material.

4.- Experimental Facilities, Computer Codes

Stress pulses are produced by the IFKM gas gun and target temperature (in-situ - heating) by a RF heat generator. The loaded specimens are investigated by common metallographic methods including scanning electron microscopy. Shock wave shapes are measured by a new method using the pressure dependent voltage generation in bimetallic junctions. A one dimensional wave propagation code (PUFF +) is available for the calculation of stress distributions in an impacted specimen.

5.- Progress to Date

Several series of impact experiments and subsequent metallographic studies have been carried out to investigate

5.1 the nature and the behaviour of crack nuclei, the influences of texture, annealing, and temperature on the initiation threshold which indicates the sensitivity against shock loading, especially for the commonly used steel 22 Ni Mo Cr 3 7 .

5.2 In tensile experiments the strength of shock loaded (i.e. damaged) material has been measured in various directionswith respect to the damaged zones.

5.3 A new method has been applied for measuring the shape of the shock pulses at the rear surface in order to get a better knowledge of the stress distribution at the instant of damage. The wave propagation code has been adjusted to the local UNIVAC computer. Difficulties were encountered because of the lower accuracy of this machine.

6.- Results

6.1 For steel, in particular the reactor pressure vessel steel 22 Ni Mo Cr 3 7,

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+ developed by Stanford Research Institute, Menlo Park, California , U.S.A.

the following results on the nucleation and growth of microcracks were found: Damage occured by exceeding a shock wave pressure threshold of approximately 25-26 kbar ($\approx$ 120 m/s impact velocity at 25^{o}C and about 20-22 kbar at 300^{o}). The exact value depends slightly on texture and annealing.
Two main types of crack nuclei were found: impurities (sulphides and oxides) and ferrite precipi tations, latter are less important for the reactor vessel steel. Microcracks were nucleated either by brittle cleavage of grains or brittle cracking of grain boundaries. Ductile extension and coalescence of microcracks lead to macroscopic fracture, by which the integrity of the vessel or component may be endangered severely.

6.2 Tensile specimens were machined from the damaged zone of the targets impacted with different velocities. Especially because of cracks nucleated at flat inclusions parallel to the rolling plane the tensile strength perpendicular to the rolling plane decreases linearly by about 50 % by increasing impact load from 30 to 60 kbar. Controversely the tensile strength parallel to the rolling or the transverse direction decreases only by less than 10 % of the strength of the unloaded material.

6.3 Shock profiles are measured successfully by using the method of voltage generation at a bimetallic junction during pressure loading. From these measurements the stress distribution at the damaged plane inside the specimen can be extrapolated with a higher accuracy than before.

The conclusion which can be made at this status of the investigations is that the effective tensile stress at the damage plane in our experiments does not reach the value of the impact pressure. The reason is the finite risetime of the shockwave (e.g. 0.5 microseconds for an impact pressure of 45 kbar in a 22 Ni Mo Cr 3 7 sample of 6 mm thickness. That means that the material especially at lower impact velocities, where the risetime increases, will be damaged at even lower stresses than reported above.

7.- Next Steps

The work has been finished. The final report is available.

8.- Relation with Other Projects

This work is related to the investigations 11, 12 and 13 within the project RS 102.

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9.- References

Results are published in the

IFKM Report 8/74 (German)
and
IFKM Report 3/76 (German), Final Report,
both entitled "Shock Waves as Possible Causes of Damage to Nuclear Reactor Components"

10.- Degree of Availability

This reports can be ordered from
Gesellschaft für Reaktorsicherheit
Glockengasse 2, 5 Köln - 1

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<table>
<tr><td colspan="3">Classification: 11.2.1</td></tr>
<tr><td colspan="2">Title 1 (Original Language):
Experimentelle Spannungsanalyse
(RS 102-14 - II.2., Jahresbericht A 75)</td><td>COUNTRY:
BRD
SPONSOR:
BMFT
ORGANIZATION:
Fraunhofer-Gesellschaft, IFKM</td></tr>
<tr><td colspan="2">Title 2 (english):
Damage of Nuclear Reactor Components by Shock Wave Loading</td><td rowspan="2">Project Leader:
S. Winkler</td></tr>
<tr><td>Initiated (Date):
1.5.1973
Status:
continuing</td><td>Completed (Date):
30.4.1976
Last Updating (Date):
December 1975</td></tr>
</table>

1. General Aim

In addition to static or quasistatic and thermal loading of reactor pressure vessels and components - which are very well investigated - extremely dynamic loading conditions (shock waves) may occur, for example by impinging fragments or projectiles, by internal explosions or - after certain cases of partial failure - by impact of the accelerated reactor pressure vessel against the containment. The investigation of damage in these dynamic cases is general aim of this work.

2. Particular Objectives

The particular objective is the investigation of (1) the distribution of the produced dynamic stress fields (shock waves) (2) the interaction of these shock waves with the material and (3) the initiation of damage in the material.

3. Experimental Facilities and Programme

In this programme plane shock waves are produced by impinging a target with a flat impactor in a gas gun. Reflected shock waves cause tension fields in the target by which microcracks are nucleated and subsequently activated to grow further, thus leading to internal damage in the material. Metallographic methods are used to study the crack initiation process. The influence of material structure, temperature, and material history, e.g. heat treatment on the crack initiation process are investigated. The decay of the dynamic stress fields with time is of special importance, therefore stress pulse profiles are measured.

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4. Project Status

4.1 Progress todate

Damage thresholds - especially for the commonly used steel 22 NiMoCr 3 7 - have been measured in relation to material structure, annealing and specimen temperature during impact. The nature of crack nuclei has been investigated.

The strength of the shock loaded (i.e. damaged) material has been investigated in various directions in tensile experiments.

Effort has been made in applying a new method for measuring the shock profile at the rear surface in order to get a better knowledge of the stress distribution at the instant of damage.

4.2 Essential Results

a) For steel, in particular the reactor pressure vessel steel 22 NiMoCr 3 7 the following results on the nucleation and growth of microcracks were found: Damage occured by exceeding a shock wave pressure theshold of approximately 25 - 26 kbar ($\approx$ 120 m/s impact velocity at 25^0C and about 20 - 22 kbar at 300^0). The exact value depends slightly on texture and annealing.
Two main types of crack nuclei were found: impurities (sulphides and oxides) and ferrite precipitatations, latter are less important for the reactor pressure vessel steel. Microcracks were nucleated either by brittle cleavage of grains or brittle cracking of grain boundaries. Ductile extension and coalescence of microcracks lead to macroscopic fracture, by which the integrity of the vessel or component may be endangered severely.

b) Tensile specimens were machined from the damaged zone of the targets impacted with different velocities. Especially because of cracks nucleated at flat inclusions parallel to the rolling plane the tensile strength perpendicular to the rolling plane decreases linearly by about 50 % by increasing impact load from 30 to 60 kbar. Controversely the tensile strength parallel to the rolling or the transverse direction decreases only by less than 10 % of the strength of the unloaded material.

c) Shock profiles are measured successfully by using the method of voltage generation at a bimetallic junction during pressure loading. From these measurements the stress distribution at the damaged plane inside the specimen can be extrapolated with a higher accuracy than before.

The conclusion which can be made at this status of the investigations is that the effective tensile stress at the damage plane in our experiments does not reach the value of the impact pressure. The reason is the finite risetime of the shockwave (e.g. 0.5 microseconds for an impact pressure of 45 kbar in a 22 Ni Mo Cr 3 7 sample of 6 mm thickness. That means that the material especially at lower impact velocities, where the risetime increases, will be damaged at even lower stresses than reported above.

5. Next Steps

The work in progress will be continued. This is in particular: Damage threshold values for the austenitic steel X 6 Cr Ni Mo 18.11 will be measured; shock wave behaviour will be investigated in more detail, to correlate the damage within the target with the actual load history in the respective plane.

6. Relation with other Projects

This work is related to the investigations 11, 12 and 13 within the project RS 102.

7. Reference Documents

Results are published in the IFKM-Report 8/74 "Shock Waves as Possible Causes of Damage to Nuclear Reactor Components", Dec. 1974

Annual Reports A 73: IRS-F-18

A/74: IRS-F-24 (Englisch)

8. Degree of Availability

The Reports can be ordered from

Institut für Reaktorsicherheit der TÜV

Glockengasse 2, 5 Köln - 1.

The IFKM Report can be ordered from Institut für Festkörpermechanik Rosastr. 9, 78 Freiburg

Classification: 11.2.1	
Title 1 (Original Language): Bestrahlungseinfluß auf Festigkeit und Relaxation von hochfesten austenitischen Stählen und Nickellegierungen für Verbindungselemente der Kernstruktur (RS 133 - II.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Irradiation Influence on Strength and Relaxation of Austenitic Steels and Ni-alloys for Core Structure Connections	Project Leader: H. Debray
Initiated (Date): 1. 2. 75 Status: Completed	Completed (Date): 30. 9. 76 Last Updating (Date): 31. 12. 76

1. General Aim and 2. Particular Objectives

The influence of neutron irradiation on the strength and relaxation of materials for core structure connections of light water reactors had to be investigated.

3. Research Program

For three characteristic materials and two heat-treatment conditions of each quantitative experimental results were expected:

1. Steel Nr. 1.4550, representative for austenitic steel as used in LWR's.

2. Inconel-X-750, representative for connection material (frequently used in USA).

3. Steel Nr. 1.4980 (USA: A 286) with lower tendency towards selective corrosion and lower thermal expansion coefficient compared with Inconel-X-750.

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4. Experimental Facilities

Irradiation in special irradiation channels in the Obrigheim Power Reactor (KWO). Post irradiation testing in the hot cell facility of KWU, Erlangen. Testing of unirradiated samples in the materials testing laboratory of KWU, Erlangen. Machines used for this program: Tensile testing machine and high-frequency pulsator (Fatique tester).

5. Progress to Date

The evaluation of the irradiated and unirradiated samples was continued.
The possibility of determining the modulus of elasticity from irradiated tension samples was investigated. Theoretically the influence of the irradiation on the modulus of elasticity was studied in order to find out, whether for the conversion of tension-data to fatique curves (Larger-Manson-Coffin-model) the E-module after irradiation was necessary.

6. Results

The measured values of the modulus of elasticity (unirradiated) were within $\pm$ 6 % in the range of the producers data.

The monitor evaluations showed that a mean neutron fluence of $3{,}9 \cdot 10^{20}$ cm^{-2} ($E > 1$ MeV) had been in the middle of the channel and 5,1 or $3{,}0 \cdot 10^{20} cm^{-2}$ for the samples, directed to the core or in the opposite direction.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

1309

Classification: 11.2.1	
Title 1 (Original Language): Literaturstudie: Selektives Korrosionsverhalten von in Leichtwasser-Reaktoren eingesetzten Werkstoffen (RS 159 - II.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Literature Study on Selective Corrosion of LWR-Materials	Project Leader: Stieding
Initiated (Date): 1. 4. 75 Status: Completed Completed (Date): 30. 6. 76 Last Updating (Date): 31. 12. 76	

1. General Aim

Some materials (austenitic, ferritic and martensitic steels and Ni-alloys) which are used in the nuclear steam generating system were to be investigated, whether they could be attacked by selective corrosion under certain conditions. As a first step a literature study has been conducted on the behaviour of some selected materials.

2. Particular Objectives

In order to raise the safety of operating reactors and to get more information about the risks of longtime effects the knowledge of the behaviour of primary materials had to be extended over the limits of the usual operation lifetime data.

3. Research Program

In the first step the following details will be studied from the literature:

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- corrosion under RPV-cladding, caused by material failures
- influence of the corrosion on the failure behaviour and the crack growth
- hydrogen induced corrosion
- determination of the corrosion limits of materials containing Ni
- influence of gaps and different material combinations
- electrochemical investigations on the corrosion mechanism.

4. Experimental Facilities

No experimental facilities necessary.

5. Progress to Date

The literature study was continued, regarding Ni-Cr-Fe alloys in the PWR primary circuit and in the BWR.

The intercristalline corrosion was studied
- under imperfection of the RPV cladding
- for different material combinations in pure hot water
- on the influence of special mechanisms, investigated by electrochemical methods.

6. Results

The published data on the behaviour of austenitic Ni-Cr-Fe alloys with respect to intercristalline corrosion, caused by Cr-pauperisation of the grain boundaries, allow a theoretical calculation of the attack on Ni-base-alloys after different heat conditions.

The contact of Inconel 600 with C- and low alloyed steel or stainless steel does not increase the intercristalline corrosion in pure hot water. Austenitic Cr-Ni-steel however increases the perforation-corrosion of RPV-steel in BWR-water.

RPV steel will be perforated in pure hot water. The velocity however was not defined quantitatively. Some publications show that the deepest holes have an asymtotic statistic and with the aid of the Gumbel-statistic the perforation prohabilities and perforation times can be determined.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

1313

Classification: 11.2.1	
Title 1 (Original Language): Reduktion der Aktivierung und Kontamination von Reaktorkreisläufen (PNS 4123 - II.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: ORGANIZATION: PNS/GfK
Title 2 (English): Reduction of Activation and Contamination of Reactor Loops	**Project Leader:** I. Michael, IRB
Initiated (Date): Jan. 1974 **Completed (Date):** Dec. 1977 **Status:** Continuing **Last Updating (Date):** Dec. 1976	

1. General Aim

The aim of this investigation is reduction of radiation dose to maintenance and repair personnel in nuclear power reactors.

2. Particular Objectives

Laboratory scale investigation of the metal los rate to the water and deposition of corrosions products under conditions of the modern pressurized water reactors,i.e.:INCOLOY 800 and other stainless steel materials, temperature 342 °C, pressure 150 bar, water chemistry (boron acid, LiOH).

3. Research Program

The influence of the water chemistry was examined in several steps started with pure water. The program will be finalized in 1977 with all additional components in the water.

4. Experimental Facilities

The protective gold plating applied on the inner side of the autoclave system I had to be renewed after investigation of the metal loss rate

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of INCOLOY 800 specimens by several test series.

The autoclave system II (same dimensions as the autoclave system I) also provided with an inner gold plating was nearly completed.

A small sized autoclave system III was completed which allows to study the metal loss rate to pressurized water of standard chrome-nickel steel, the material making up 15 % of the inner surface of primary circuits in PWRs.

5. Progress of Date

Investigations were performed of the metal loss rate of the INCOLOY 800 steam generator material at 342 °C and 150 bar with oxygen contents in the pressurized waters of 0,05 mg/kg, extending over twice the running period of 1000 hours and, also, with additional alcalization using lithium hydroxide (pH-value 9,7).

The older version of the atomic absorption spectrometer used for the determination of the metal content in water was automated for laboratory purposes.

6. Results

Reduction of the oxygen content in deionate through noble gas sweeping which is much more effective than chemical reduction by hydrazine was improved in combination with the autoclave system I. It appeared that the minimum oxygen content of 0,05 mg/kg required for primary loop water of pressurized water reactors can still be underrun by the factor of 20.

In the two metal loss rate runs an abrupt, discontinuously enhanced metal concentration was found in the pressurized water after about 400 hours. It suggests the growth of the surface layer on the INCOLOY specimens until an instable thickness is reached at which the metal oxide begins to detach.

Automation on the laboratory scale of the atomic absorption spectroscopy essentially resulted in making shorter the handling and evaluation period and in a higher accuracy of analytics.

7. Next Steps

Three autoclave systems will be available to continue the investigations. The discontinuous metal loss rate observed calls for further investigations. They will relate to the chemico-physical partial processes of transport of the metal and metal oxide content released in the pressurized water and of the determination of the solid content, classified by size.

8. Relation with Other Projects

The physico-chemical analysis of the corrosion layers is performed by Dornier System, Friedrichshafen.

9. References

I. Michael, C. Plog,
Dünnschichtanalyse von Druckwasserkorrosions-Schichten auf INCOLOY 800-Proben (to be published in "METALL")

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Classification: 11.2.1		
Title 1 (Original Language): Fremdstoffe in Leichtwasserreaktorkühlmitteln (PNS 4123 - II.2., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Corrosion Substances in Light Water Reactor Coolants		Project Leader: J. Michael
Initiated (Date): 1974	Completed (Date): 1977	
Status: continuing	Last Updating (Date): December 1975	

1. General Aim

The metal loss of primary loop materials to pressurized water is examined with a view to reducing the primary loop contamination of pressurized water reactors. This examination mainly relates to the INCOLOY 800 steam generator material.

2. Particular Objectives

A lower primary loop contamination is expected from the reduction of metal loss mainly by means of water chemistry measures.

3.1 Experimental Facilities

A high pressure autoclave system with a volume of 4 liters is available. The inner surface of that autoclave is coated with a thin layer of gold.

Atomic absorption spectrometry is used for the water chemical analysis. The surfaces of test samples are examined according to the solid-state analytic methods of the secondary ion and Augèr electron spectroscopy.

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3.2 Research Program

The main components of the alloy - iron, chromium and nickel - released from INCOLOY 800 to pressurized water - are determined by 500 h tests at 342°C and 150 bar. The water chemistry conditions are varied during this process.

4. Project Status

In several test runs the oxygen content of the pressurized water was lowered by means of thermal degasing, flushing with noble gas, and addition of hydrazine.

4.1 Progress to Date

The measures described above brought about a reduction by one power of ten of the deionate oxygen content by one power of ten.

4.2 Essential Results

In the process of lowering the oxygen content the metal content in the pressurized water was reduced by 33% of iron after 500 hours at 342°C and 150 bar. The metal content of nickel decreased by 40% while the chromium content could not be well defined due to the very low intitial values which were within the range of the detection limit of the atomic absorption spectroscopy.

The surface layers formed on the INCOLOY samples after 500 h under operating conditions exhibited oxide, hydroxide and hydride fractions of the main components of the alloy. As compared to the fractions of the other main alloy components, there is a noticeable enrichment in the chromium content in the surface layer so that such surface can be assumed to behave chemically as a protective layer.

5. Next Steps

The investigations of the metal loss to pressurized water are supplemented by variation of the pH-value of the pressurized water. Also, the solid content in the pressurized water will be especially analyzed.

To speed up the program, a second autoclave system is presently integrated.

6. Relations with Other Projects

Work is performed in contact with reactor operators and reactor manufacturers.

7. Reference Documents

G. Bechtold, I. Michael, R. Prümmer
Zur Gold-Innenbeschichtung von Autoklaven durch Explosivplattieren (in German)
Metall 29 (1975) H. 7, pp. 685 - 687

8. Degree of Availability

Preprint available at GfK, Karlsruhe

Classification: 11.2.1	
Title 1 (Original Language): Erweiterung des Einsatzbereiches kobaltfreier Werkstoffe für Reib- und Verschleißbeanspruchung im NDES von Leichtwasserreaktoren (RS 207 - II.2, Jahresbericht A76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Extension of the Use of Cobalt-Free Materials for Resistance of Friction, Wear and Corrosion in the Nuclear Steam Supply System (NSSS) of LWRs	Project Leader: H. Stieding
Initiated (Date): 1. 5. 76 Status: Continuing Completed (Date): 30. 9. 79 Last Updating (Date): 31. 12. 76	

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 und 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.

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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 intergranular 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

A literature search has been made to select suitable, commercial FeCrC alloys. Well-known manufacturers have also been contacted in order to discuss the possibilities of fabricating suitable plating materials and their willingness to do so.

6. Results

Literature searches gave the following results:

a) There are very few suitable FeCrC alloys available on the market.

b) "Pantanax 25/0 Mo", a TEW-made FeCrC alloy, presently not available for commercial use, is expected to be a suitable material for the planned tests.

c) In addition to the FeCrC alloys (high-crome cast-iron), overlay welding materials ("Nitronic"; Ni-based alloys) should also be tested.

Discussion with the manufacturers led to the following results:

a) FeCrC alloys

Co-free plating materials showing the required combination of characteristics have not yet been studied. The major problem here is the temperature requirement (approx. 320 - 350°C) during long term operation (40 years), which requires extensive work in order to investigate the suitability of just one FeCrC alloy. Therefore ist was agreed to proceed step by step. A TTT diagram will be made of Pantanax 25/0 Mo-G to provide a basis for specific development.

b) Nitronic steels

Comparative tests, performed by the manufacturers, showed that the friction and wear behaviour of these steels at RT is fairly good. Additional tests are required on: corrosion behaviour, friction and wear behaviour under defined conditions at RT and 300°C and the temperature stability of platings of these materials.

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c) Tribaloy

According to the manufacturers, the discussed alloy T 700 can be applied by shielded arc welding. Unfortunately, the material shows low ductility. While temperature stability seems to be guaranteed, friction wear and corrosion behaviour have still to be evaluated. As a first step, samples of cast material will be carefully pre-tested.

7. Next Steps

According to the current task matrix, the next steps will be as follows:

- establishment of a material list
- acquisition of test material

At the same time a literature survey will be made on "Plasma sprayed layers".

8. Relation to Other Projects

Preparatory work has been done under the scope of R & D task RB 72 I/S 11.

9. References

P. Hofmann
Wear resistance of materials and protective layers with respect to their behaviour in water-cooled nuclear power plants final report R & D task RB 72 I/S 11, KWU, Aug. 1973.

10. Degree of Availability

The report is classified Company Confidential

1325

Classification: 11.2.1.

Title: Dynamisk brudmekanik	Country: DENMARK
	Sponsor: Res. Establ. Risø
Title: Dynamic Fracture Mechanics Crack propagation in inhomogenous steel	Organization: Res. Establ. Risø
Initiated date: 1973 Completed date: Status: Some basic investigations finished	Scientists: A. Nielsen 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.

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131-11-23/4111.03 Classification : 11-2-1

Titre	Pays : FRANCE
Dommages consécutifs aux défauts des structures et/ou évolutions de chargement.	Organisme directeur : CEA/DSN
Titre (anglais) Damages subsequent to structure defects and/or load changes	Organisme exécuteur : CEA/DEMT
	Responsable : B. BARRACHIN (DSN - SETSSR) R. ROCHE (DEMT)
Date de démarrage : 1/01/76 Date prévue d'achèvement : 31/12/80 Etat actuel : en cours Dernière mise à jour : 01/01/77	Scientifiques :

Objectif général :

Cette étude a pour but de vérifier les lois de comportement en fatigue et fissuration des matériaux utilisés dans les circuits primaires des réacteurs à eau PWR.

Objectifs particuliers :

- Influence du milieu
- Influence de la biaxialité
- Comportement sous chocs thermiques

Etat de l'étude :

1) Avancement à ce jour :

Les essais effectués ont permis de préciser le comportement en fatigue des aciers pour cuve (acier Creusot-Loire type A 508 classe 3) et des alliages austénitiques en milieu PWR.

2) Résultats essentiels :

- Pour l'acier type A 508 classe 3, la courbe de fatigue du Code ASME section III est acceptable.
- Pour les aciers austénitiques, la courbe de fatigue du Code ASME section III est trop optimiste et devrait être réajustée.

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Prochaines étapes :

- Fatigue sous sollicitation biaxiale (fin 77)
- Essais sur récipients :corrélation avec les essais sur éprouvettes (1980)

Documents de référence :

"Low Cycle Fatigue of Austenitic Alloys in Hot Water"
C.GARNIER, R.ROCHE, B.BARRACHIN.
Smirt L 6/7 Londres 1975

"Low Cycle Fatigue of Steels for Nuclear Pressure Vessels in Hot Water"
C.GARNIER, G.KOWALCZUK, R.ROCHE, B.BARRACHIN.
Smirt L 8/h San Francisco 15 - 19 août 1977.

Titre	Pays : FRANCE
Comportement des organes d'isolement du circuit primaire des réacteurs à eau.	Organisme directeur : CEA/DSN
Titre (anglais) Behaviour of isolation valves of PWR primary system	Organisme exécuteur : CEA.DEMT Responsable : B.BARRACHIN (DSN-SETSSR) R.ROCHE (DEMT)
Date de démarrage : 1/01/77 Date prévue d'achèvement : 31/12/81 Etat actuel : Démarrage Dernière mise à jour : 01/01/77	Scientifiques :

Objectif général :

Cette étude à pour but de préciser le comportement des organes d'isolement du circuit primaire des réacteurs à eau.

Objectifs particuliers :

L'étude devra essentiellement permettre d'apprécier la validité des arguments présentés quant au bon fonctionnement de ces organes en condition normale et en condition accidentelle.

Prochaines étapes :

Essais préliminaires d'un couple siège-corps (fin 77)
Calcul dynamique-plastique du comportement (fin 78)
Essais complémentaires (1980)
Etablissement de régles générales (1981)

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Titre	Pays :
Comportement des aciers pour cuves - tenue à l'irradiation	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Research programme on irradiation embrittlement of pressure vessel steels.	Organisme exécuteur : CEA/DTECH
	Responsable : B. BARRACHIN (DSN-SETSSR)
Date de démarrage : 01/01/76 Date prévue d'achèvement :31/12/79 Etat actuel : en cours Dernière mise à jour :30/11/76	Scientifiques : P.PETREQUIN (DTECH - SRMA)

Objectif général :

Le programme à pour but de déterminer le comportement sous irradiation neutronique de l'acier Creusot-Loire (type A 508 classe 3) utilisé pour la fabrication des cuves PWR du programme nucléaire français.

Objectifs particuliers :

Le programme comporte essentiellement :
1) Les mesures de KCV et NDT,
2) Les mesures de KIC sur grosses éprouvettes,
3) L'étude de la restauration,
4) Les mesures de KID et JIC sur petites éprouvettes,
5) La participation au programme AIEA.

Installations expérimentales et programme :

Réacteur expérimental TRITON (CEN-Fontenay-aux-Roses).

Etat de l'étude :

1) Avancement à ce jour :

Le point 1 est terminé. Les points 2,3,4 sont en cours. Le point 5 est prévu pour 1977.

2) Résultats essentiels :

1) Evolution des caractéristiques de fragilité sous irradiation. Les essais ont porté sur des échantillons prélevés dans les viroles de coeur des 5 premières cuves du programme français. Les irradiations ont été menées à 290 degrés C et flux intégré de 4 X 10^{19} NVT. Le déplacement de la température de transition est faible pour le métal de base et plus important pour les soudures.

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2) Les mesures de KIC ont été effectuées sur matériaux non irradiés. Un dispositif pour l'irradiation de grosses éprouvettes est en cours de réalisation.

3) Les premiers essais ont permis de préciser la cinétique de restauration et d'examiner les équivalences temps-température.

<u>Prochaines étapes</u> :

1) Mesure de KIC sur grosses éprouvettes irradiées.

2) Vérification des équivalences temps - température.

3) Détermination des courbes de transition de KID. Définition de la méthode et mesures de JIC.

4) Définition du programme AIEA.

131-11-31/4111-66

Classification : 11-2-1

1333

Titre	Pays :
Propagation de fissures de fatigue sur les matériaux de structure irradiés.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais)	Organisme exécuteur : CEA/DTECH
Fatigue crack propagation on irradiated materials.	Responsables : B. BARRACHIN (DSN-SETSSR) P.PETREQUIN (DTECH-SEMA)
Date de démarrage : 1/01/76 Date prévue d'achèvement : 31/12/79 Etat actuel : en cours Dernière mise à jour : 1/1/77	Scientifiques :

Objectif général :

Le programme a pour but de déterminer les lois de propagation des fissures de fatigue sur les matériaux de structure irradiés.

Objectifs particuliers :

Les essais sont plus spécialement orientés sur les aciers pour cuves de réacteurs à eau sous pression PWR.

Etat de l'étude :

1) Avancement à ce jour :

La machine de fatigue a été montée en cellule chaude et est actuellement opérationnelle. Par ailleurs les essais d'irradiation sont en cours.

Prochaines étapes :

Essais sur matériaux de cuve :
- à l'ambiante (fin 77)
- en température élevée (fin 78)

140-21- /4111-104

Classification : 11.2.1
1335

Titre	Pays : FRANCE
Comportement des lignes de tuyauteries lors d'un accident .	Organisme directeur : CEA/DSN
Titre (anglais) Behaviour of pipings in the event of an accident	Organisme exécuteur : CEA/DEMT
	Responsable : B.BARRACHIN (DSN-SETSSR) R.ROCHE (DEMT)
Date de démarrage : 1/03/77 Date prévue d'achèvement :31/12/81 Etat actuel : Démarrage Dernière mise à jour :1/4/77	Scientifiques : B.VRILLON

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 : RCV, RIS, ANG, ASG. Les essais sont réalisés dans des conditions proches de l'échelle réelle.

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.

Installations expérimentales et programme :

Boucle HUREPOIX (CEN-SACLAY).

Prochaines étapes :

Mise au point de la boucle et premiers essais (fin 77)
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 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.

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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.

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TNO - Metaalinstituut	CLASSIFICATION: 11.2.1.
TITLE: Literatuurstudie: (1) Invloed van omgevingscondities op vermoeiings-scheurgroei. (2) Invloed van veroudering op de mechanische materiaaleigenschappen.	COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs ORGANIZATION: TNO-Metaalinstituut
TITLE: (ENGLISH LANGUAGE): Literature study: (1) Influence of Environment on Fatigue crackgrowth. (2) Influence of aging on fracture related material properties.	PROJECTLEADER: Van Rongen
INITIATED: June 1974 — LAST UPDATING: April 1977 STATUS: near completion — COMPLETED: mid 1977	SCIENTISTS: —

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.

Particular 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 theshold 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

1. Progress to date: literature survey completed
2. Essential results: to be reported mid 1977

Next steps: ready for the press

Relation with other projects: -

Reference documents: -

Degree of availability:

Trhough the Ministry of Social Affairs, C.R.V., Postbus 69, Voorburg, The Netherlands

Budget: Hfl. 50.000,--

Personnel: 2 scientists

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: NETHERLANDS. SPONSOR: Ministry of Social Affairs and others ORGANIZATION: NIL, MI - TNO
TITLE: (ENGLISH LANGUAGE): Literature survey of material parameters which are necessary for the diagnosis of defects localized with AE techniques.	PROJECTLEADER: Kloots
INITIATED: September 1974 / LAST UPDATING: April 1977	SCIENTISTS: Boerstoel, V.d. Brink
STATUS: in progress / COMPLETED: May 1977	

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: -

Project status: to be completed soon

Next steps: final report

Relation with other projects: see under 1

Reference documents: NIL-lastechniek, 40e jaargang, no. 5, mei 1974

Degree of availability: Through Nederlands Instituut voor Lastechniek

Budget: Approx. Hfl. 30.000,--

Additional information:

Survey will be executed by Metaal Instituut (TNO) and will be completed May 1977, approximately.

Classification 11.2.1	
Title 1 METAL FRACTURE (CRACK GROWTH)	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION RFL SPRINGFIELDS
Title 2	Project Leader DR B TOMKINS
Initiated 1971 Completed : 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

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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.

1345

Classification	
	11.2.1

Title 1 METAL FRACTURE (STABILITY)	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION RFL SPRINGFIELDS
Title 2	Project Leader DR B TOMKINS
Initiated 1971 Completed : 1976 Status : Last updating	Scientists:

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.

1347

Classification

11.2.1

Title 1 MECHANICS OF PLANE STRAIN FRACTURE	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION RFL SPRINGFIELDS
Title 2	Project Leader DR B TOMKINS
Initiated 1972 Completed : Status : Last updating	Scientists:

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.

Classification

11.2.1

Title 1 VESSEL FAILURE TESTS	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION RBNL RISLEY
Title 2	Project Leader DR A COWAN
Initiated 1969 Completed : 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 FRACTURE BEHAVIOUR OF STRUCTURAL STEELS	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION REML RISLEY
Title 2	Project Leader DR A COWAN
Initiated 1969 Completed : Status : Last updating	Scientists:

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.

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	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.

Classification 11.2.1

Title 1 Mechanics of Plane Strain Fracture	**Country** UK
Title 2	**Sponsor** UKAEA **Organisation** Fuel Element Labs. Springfields
Initiated 1972 **Completed** **Status** Continuing **Last updating**	**Project leaders** 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 applie 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.

1355

Classification
11.2.2/11.2.3

Title 1	
Spændingsanalyse af primære trykbærende stålkomponenter: Sammenligning mellem beregnede og målte tøjninger og spændinger i en BWR pumpestuts.	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC,Risø
Title 2 Stress analysis of primary steelcomponents: A comparison between calculated and measured stress and strain in a BWR main circulation pump nozzle.	Project leader S.I. Andersen Scientists:
Initiated (date) 74.04.01 — Completed: 1977 Status: concluded — Last updating(date) April 1977	S.I. Andersen

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 calcula-

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tions (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/d to be published at the 4th Int. Conf.
on Structural Mech. in Reactor Tech., San Francisco 1977.

8. Degree of availability

A limited amount of the results may be freely available.

1357

TITLE 1 (original language) Vibrazioni di sistemi di tubazioni	Classification 11.2.2
TITLE 2 (english) Structural and acoustic vibrations of piping systems	Country: ITALY Sponsor: CNEN-CNR-Italimpianti Organisation: University of Pisa
Date initiated 1973 Date completed 1976 Last updating June 1976	Project Leader 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.

Classification: 11.2.3	
Title 1 (Original Language): Zerstörungsfreie Wiederholungsprüfung an Reaktordruckbehältern mittels Wirbelstromverfahren (RS 89 - II.3.2; Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU Erlangen
Title 2 (English): Non-Destructive In-Service Inspection for Reactor Pressure Vessels with Eddy Current Methods	Project Leader: Gräbener
Initiated (Date): 1. 12. 72 Status: Completed Completed (Date): 30. 6. 76 Last Updating (Date): 31. 12. 76	

1. General Aim and 2. Particular Objectives

The eddy current method was to be qualified for the examination of cracks on the surface of cladded reactor pressure vessels. Appropriate conditions, detectors and procedures of registration for this method were developed. The limits for detecting flaws on sections under water have been investigated.

3. Research Program and 4. Experimental Facilities

The program was devided into two main parts:
In a pre-program first tests were carried out with existing detectors and evaluation equipment concerning the detection of flaws depth, considering thereby possible interference effects (detectorsurface distance, material inhomogenities). The pre-program showed which detectors and equipment were the most appropriate. In the main program appropriate detectors and evaluation equipment were developed according to the know-how obtained in the pre-program. With this prototype experiments were carried out under defined conditions. Data registration

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and processing were equally developed, as well as a system for carrying the detectors.

5. Progress to Date and 6. Results

The documentation of the results has been completed. The measuring technique was improved, visual inspection device and simple recording were not sufficient for rational inspections. Automation will be necessary.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

RS 27 "Development of Non-Destructive Inspection Techniques for In-Service Inspection Tests of Reactor Pressure Vessels"

9. References

Devrient, et.al.
"Zerstörungsfreie Wiederholungsprüfungen am RDB mittels Wirbelstromverfahren (Teilvorhaben 1 - 4)".
Abschlußbericht Förderungsvorhaben RS 89
Kraftwerk Union Aktiengesellschaft, Erlangen (Okt. 1976)

10. Degree of Availability

The report is company confidential.

1361

Classification: 11.2.3	
Title 1 (Original Language): Akustische Holographie - Labor- und anwendungstechnische Versuche mit dem Holscan 200 (RS 102-20/2 - II.3.2, Jahresbericht A 76)	COUNTRY: BRD
	SPONSOR: BMFT
	ORGANIZATION: Fraunhofer-Ges. IzfP, Saarbrücken
Title 2 (English): Acoustical holography - laboratory and application tests with the Holscan 200	**Project Leader:** Dr. V. Schmitz
Initiated (Date): 1.8.1975 **Status:** Continuing	**Completed (Date):** 28.2.1977 **Last Updating (Date):** December 1976

1. General Aim

In the non-destructive testing they have not yet succeeded in reaching an identical, three-dimensional picture of flaws, respectively of greater areas with defects by conventional ultrasonic methods. An exact knowledge of these flaws is necessary to determine the influence of such areas on the stability or safety of reactor pressure vessels. The acoustical holography links the conventional method of pulse-echo with the attitudes of optical holography and allows a non-destructive testing and a geometrically identical projection of flaws concerning either the basical testing or the repetitional testing. In the near future, testing procedures with the aid of acoustical holography should belong to the technical standard.

2. Particular Objectives

The acoustical holography can help to determine the exact size and locus of flaws in thick-walled vessels. We are working at two kinds of application:

- testing of the basic material, the cladding and the interface between austenitic and ferritic junctions of reactor pressure vessels;
- testing of thick-walled components both in laboratory tests and in situ.

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3. Research Program

The experiments can be devided into the following points:

3.1. Installation of an acoustical holographic equipment,

3.2. Experiments on specimens with artificial 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 instead of using focal probes as it works with the far-field length of a focal probe. The focal point in this case does not lie at the level of the flaw but can be set at the surface independently of whether the depth of the flaws is known in advance or not. All flaws reflected in the area of the ultrasonic beam will be picked up regardlessly of their distance from the surface and shown in the reconstruction phase.
Fig. 1 shows the principe. A point source in the surface is emitting a spherical wave and detects the reflected ultrasound, too. To conservate the information of the phase, there is a second wave, the reference wave with an inclination angle. This method has been realized in the holographic equipment holscan 200 of Holosonics, Inc., shown in fig. 2. On the left the tank filled with water containing the material to be examined. Above the mechanical scanner which moves a focal probe in a grid pattern. The ultrasonic signals converted into electronic signals are processed in the electronic processor. To enable the retention not only of the amplitude but also of phase a coherent phase-shifted reference signal is superimposed on the flaw signal. Thus a diffraction, the hologram, is formed which can be made visible on a viewing screen. In the optical processor on the right of the picture the negative produced by the screen is illuminated by a laser beam. The image of the flaws appears in the monitor where it can be measured.

5. Progress to Date

The following examples may serve to give some ideas of the potential of ultrasonic holography.
The test specimen shown on the right in fig. 3 is made of steel and has at four different levels in each case three flat bottom holes with a diameter of 3 mm and at distances from each other of 2, 3 and 4 mm. A 4 MHz transducer was used with a nearfield length in water of 18 inches.

The reconstructed flaws are shown on the left. Even the 3 mm flaws at a depth of 180 mm from the surface can be clearly distinguished and reconstructed. One advantage of holography is that it can employ the single probe technique as well as that the flaw lies within the widely spread beam. So shown in fig. 4, the holes with 30° and 45° slope could be recognized with the correct diameter.

In the next example the dependence of the flaw inclination with transverse wave beams was investigated. The flaws diverge from the acoustic axis (45° to the normal) by up to ± 15°. Nevertheless, all the flaws were unambiguously shown up (fig. 5).

In the following example we used a double clad specimen. The claddings were 2.4 inch wide and 1.2 inch offset against each other. The testing frequency was 1.25 MHz. As anticipated a markedly inferior reproduction of the reflector was the result. There remained however, a noteworthy and promising resolving power and sensitivity (fig. 6).

We have investigated systematically a test piece with a weld. The weld was insonified with longitudinal waves from both sides, that means in 5 inch depth, resp. in 10 inch depth. The length of the weld is about 20 inches, the height about 1 inch and the width more than 10 inches. In addition to the manual testing we have investigated the test piece with a focal probe 4 MHz, longitudinal wave and a diameter of 60 mm, Fig. 7 represents a C-scan image of the whole weld, the ultrasonic hologram and the reconstruction of the flaws. The testing frequency was 2.4 MHz. The holographic reconstruction gives a more detailed image of the flaw and allows to measure immediately the size of the flaws.

6. Results

Ultrasonic holography has shown itself to be the most promising area of testing thick walls. In all cases of experimental results the resolution capability of holography has been demonstrated.

7. Next Steps

A potential development lies in the replacement of the optical processing by a numerical processing scheme (fig. 8). With a numerical reconstruction the inspection data is not presented as a negative but fed into the core store of a computer via an analog-digital converter. The reconstruction then takes place numerically using a fast Fourier transformation and presented on a monitor or a print out. First

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experiments on a specimen shown in fig. 9 demonstrate that we can get reasonable good results - fig. 10 - in performing the numerical reconstruction. The advantages of numerical reconstruction are that it permits interferences and distortions to be removed and a free choice in the manner of representations.

8. Relations with Other Projects: none.

9. References

V. Schmitz:

Studie zur Anwendung der akustischen Holographie bei der zerstörungsfreien Prüfung von Reaktorkomponenten und Reaktoranlagen.
Teil I: Literaturrecherche
IzfP-Bericht 740106-TW, Saarbrücken 1974

C. Schmitz:

Eignung des Holscan 200 der Firma Holosonics zur Fertigungs- und Wiederholungsprüfung an Reaktorkomponenten und Reaktoranlagen.
Teil II: Prüfsystemanalyse.
IzfP-Bericht 750207-TW, Saarbrücken 1975

10. Degree of Availability of the Reports

On request by GRS, Köln.

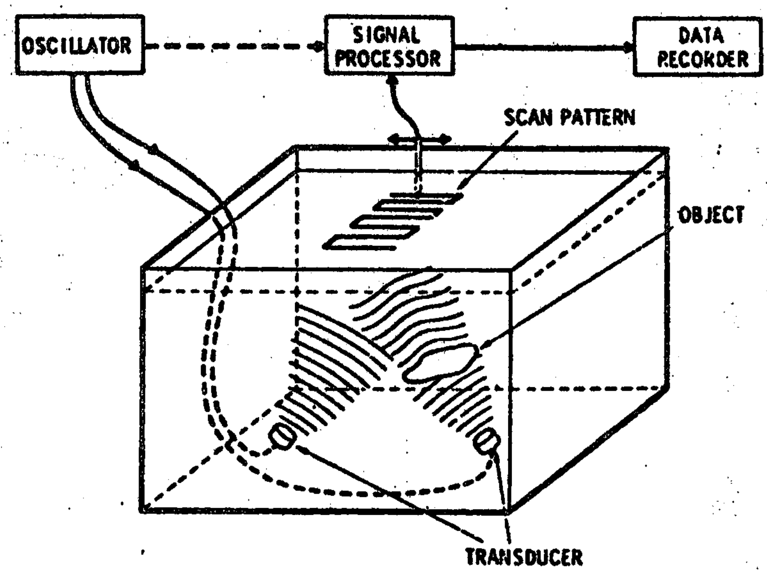

Fig. 1 : Principle of US-holographie

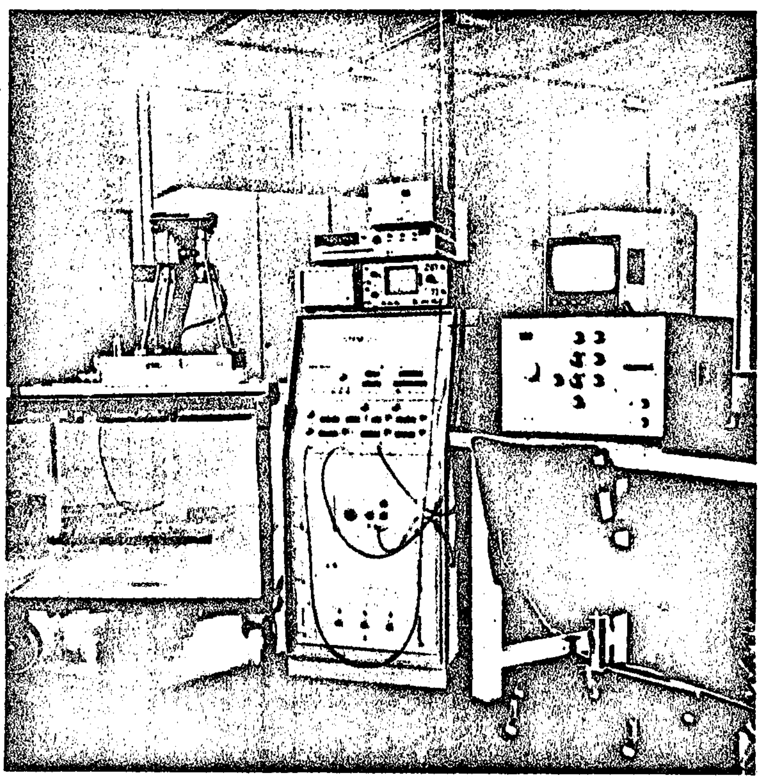

Fig. 2: Holographic equipment Holscan 200
of Holosonics, Inc.

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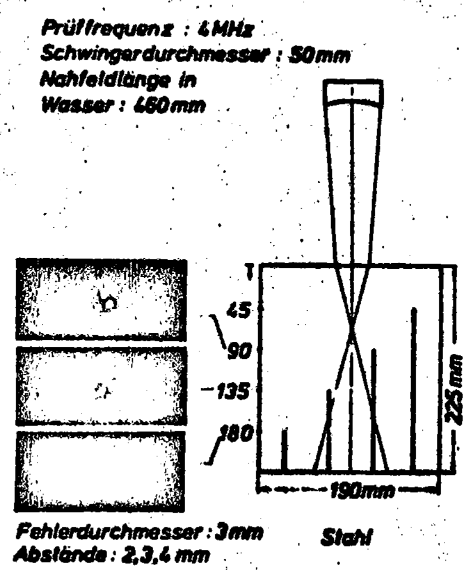


Fig. 3:
Testspecimen with flat bottom holes

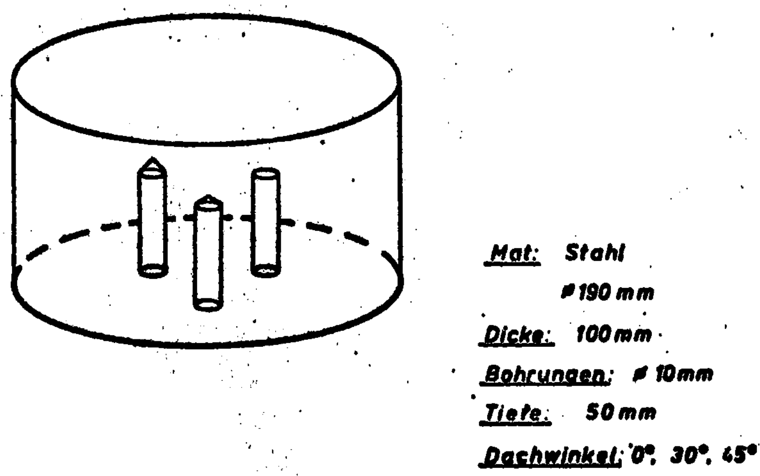
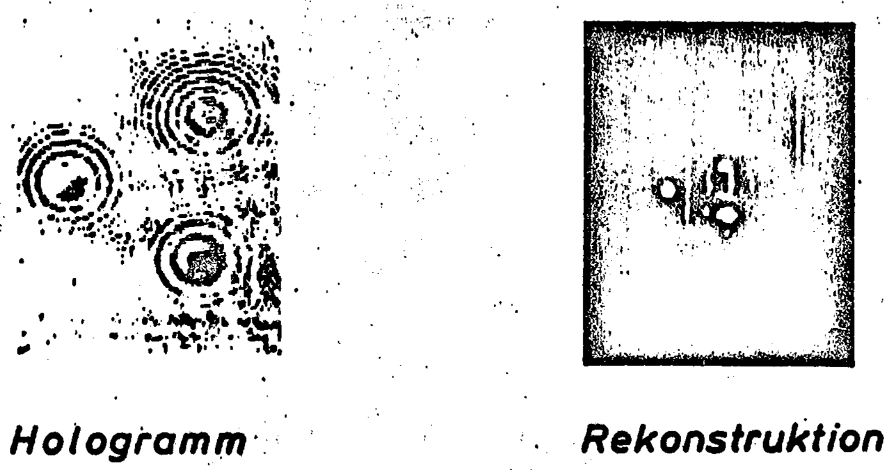
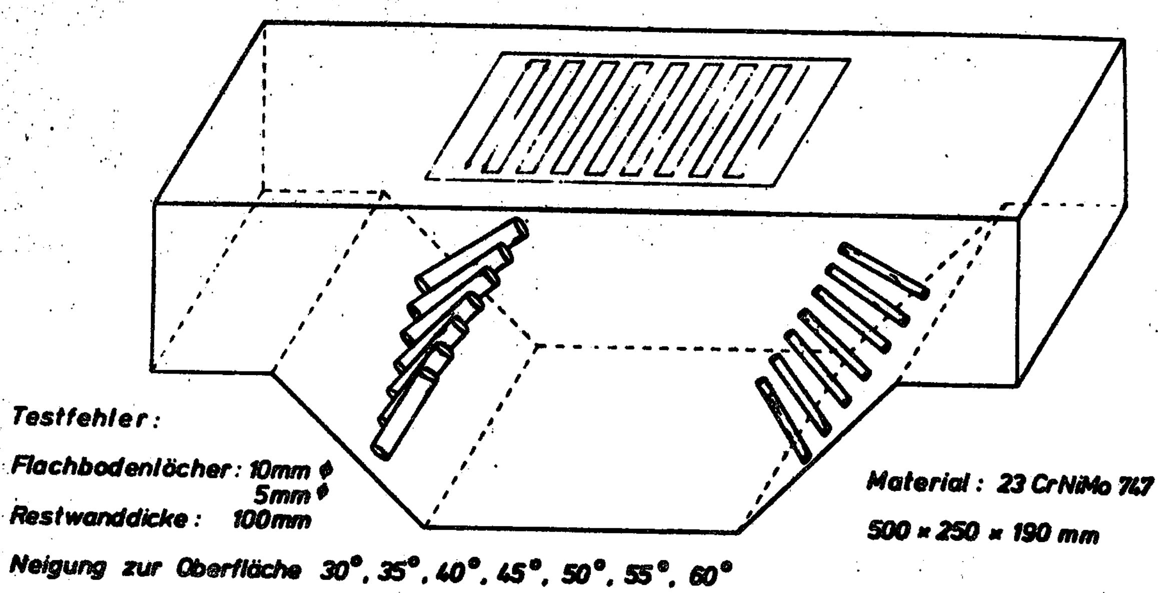


Fig. 4 a/b:
Testspecimen with holes with 30° and 45° slope



Bohrungen: 10mm ; Dachwinkel: 0°, 30°, 45° Tiefe: 50 mm



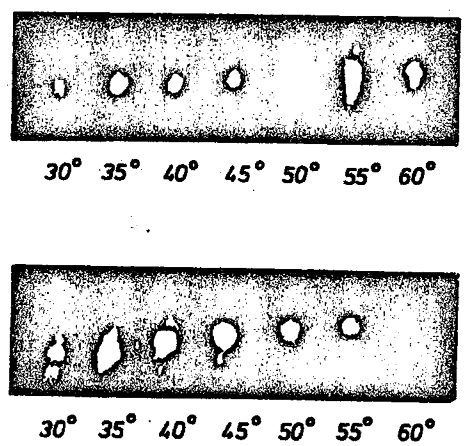


Fig. 5 a/b:
Investigation of the dependence of the flaw inclination with transverse wave beams

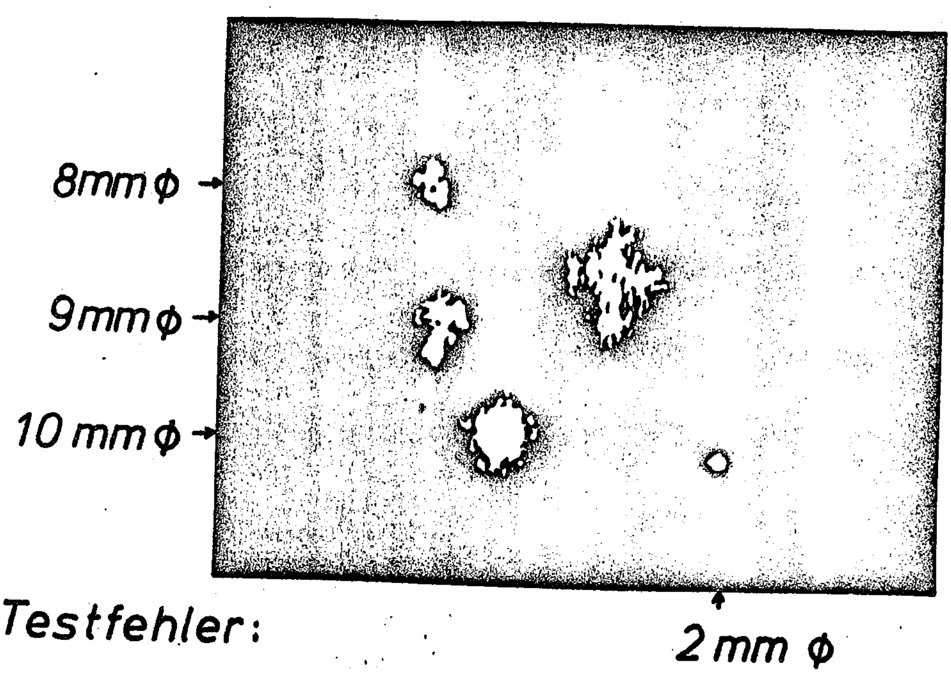


Fig. 6:
Investigation of a double clad specimen
(f = 1,25 MHz)

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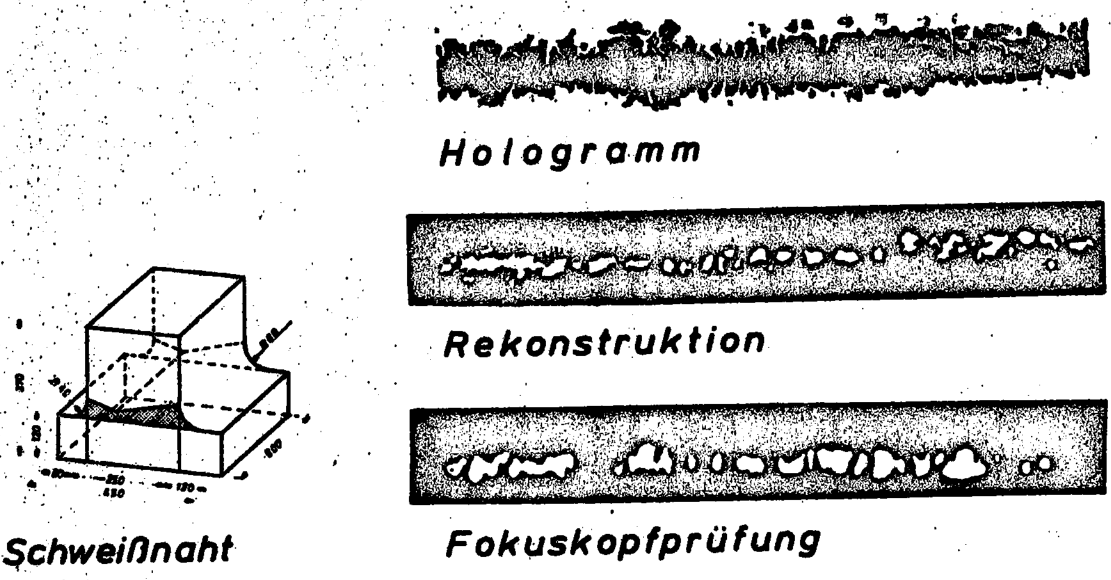

Fig. 7: C-scan image of the whole weld, the ultrasonic hologram and the reconstruction of the flaws.

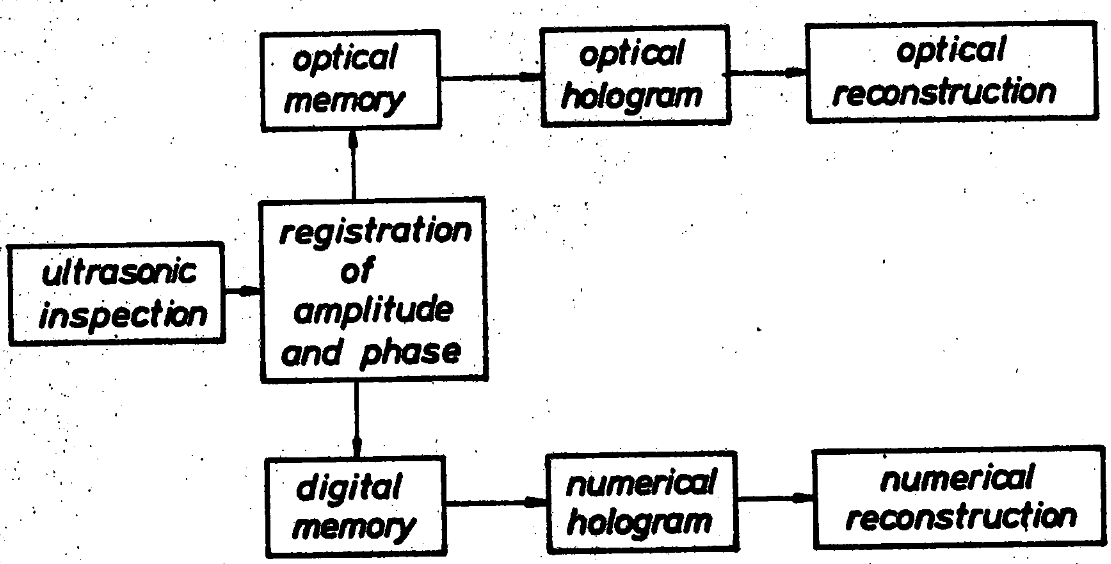

Fig. 8: Scheme of optical and numerical reconstruction.

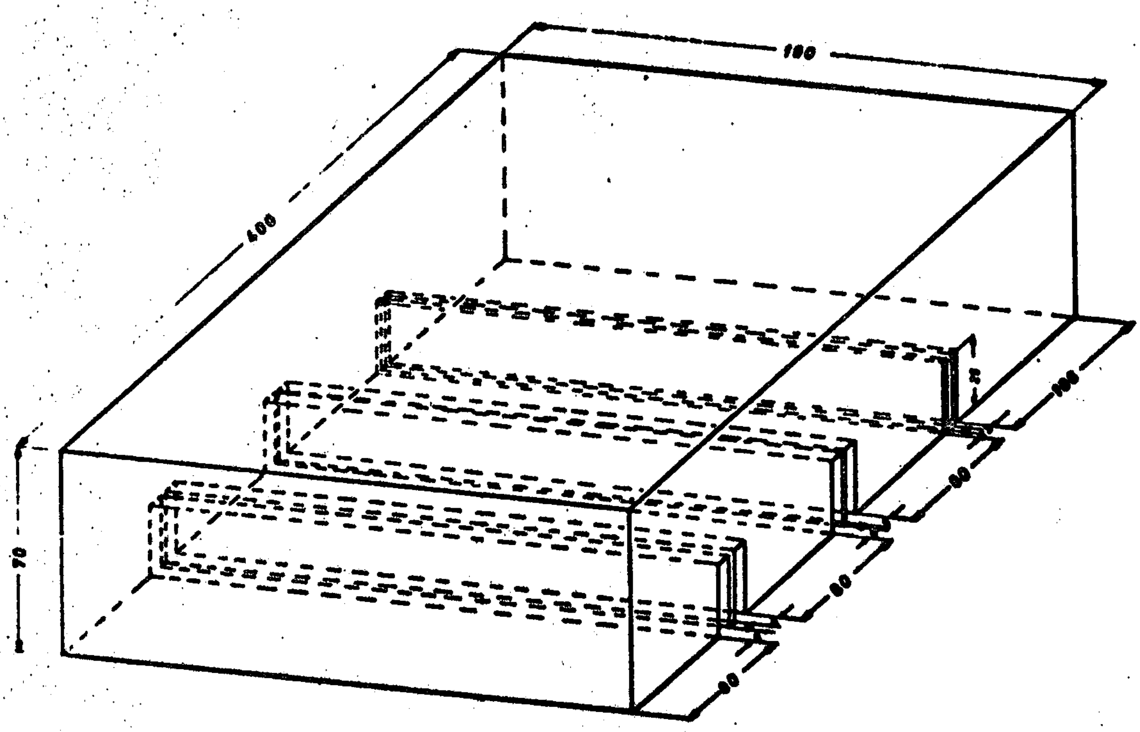

Fig. 9: Specimen for testing numerical reconstruction.

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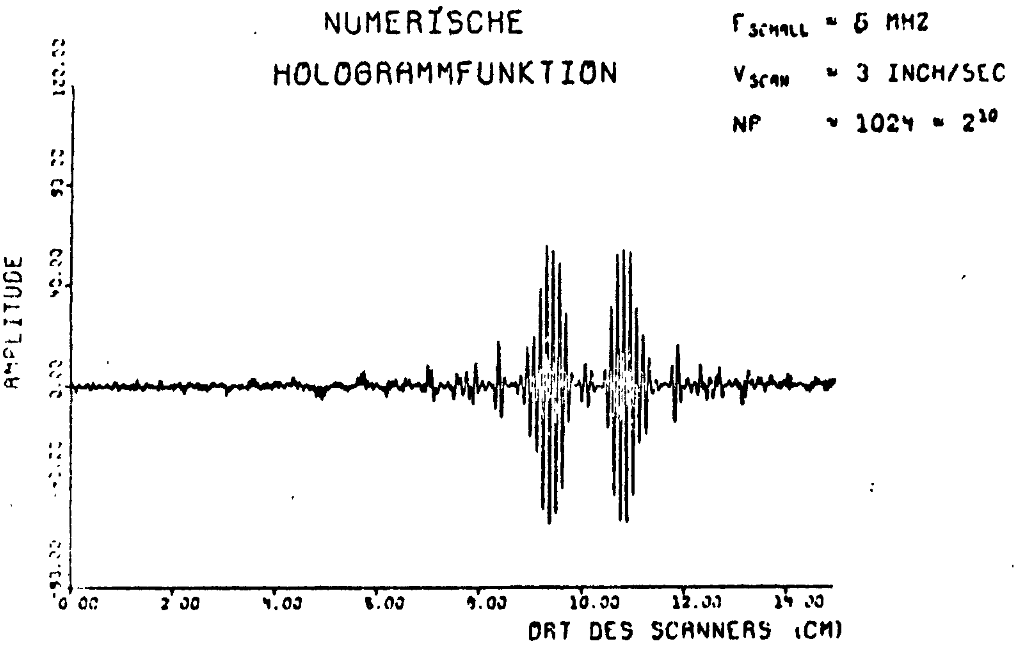

Fig. 10a: Hologram of the specimen of fig. 9

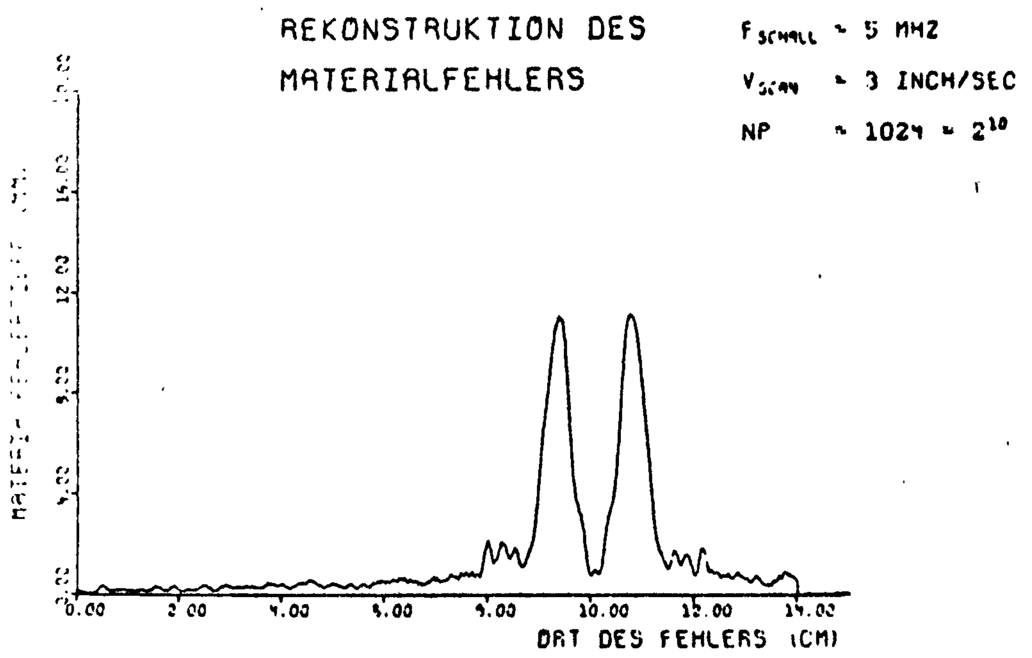

Fig. 10b: Reconstructed image (numerical reconstruction)

Classification: 11.2.3	
Title 1 (Original Language): Durchführung von Untersuchungen zur Rißerkennung an druckführenden Reaktorbauteilen mit Hilfe der optischen Holographie (RS 132-II.3.2, Jahresbericht A76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:**
Title 2 (English): Crack detection in pressurized vessels and reactor components by optical holography	**Project Leader:** Professor Dr.-Ing. F. Mayinger
Initiated (Date): July 1974 **Status:** Continuing **Completed (Date):** June 1978 **Last Updating (Date):** December 1976	

1. General aim

The aim of the activities is to develop a quick and reliable optical method in order to detect cracks in reactor vessels during fabrication and later in repeating tests.

2. Particular objectives

In the last few years a new nondestructive technique - holographic interferometry - was developed by which one can detect material defects. 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 reactor vessel steels using surface waves which are recorded by double-pulse holography.

3. Research program

At first, basic experiments with testpieces of simple geometric shape are made. The aim of these preliminary experiments is to find out the smallest crack that can be measured with holographic interferometry in connexion with the impact excitation of the testpieces. The indication of cracks is influenced by various parameters:

1. crack-parameters (shape, size, orientation and position of the crack)

2. testpiece-parameters (geometric shape, thickness and material)
3. test-method-parameters (impact-parameters sensitivity of the method)

Further experiments shall be carried out with testpieces 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. <u>Experimental facilities</u>

The principle of this holographic technique is to visualize the deformation of surface and bending waves due to small irregularities in the material. This waves cause surface deformations which can be measured with holographic interferometry.

To produce the waves it is usually necessary to strike the testsection 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, 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 testpiece shortly before the impact and thus the testpiece 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 report A 74.

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

The experiments with testpieces of simple geometric shape (plates) in connexion with the described technique (particular with the steel ball impact-generator) are finished. It is therefore possible to describe the detectable cracks as a function of different parameters. Furthermore, measurements were made with testpieces of more complex geometric shape like tubes and elbows. In addition, basic investigations were made to use a thin liquid layer above the surface of the testpiece as a signal amplifier.

This liquid layer levitates the amplitude of the generated surface waves. In fact due to the very small disturbance of waves at small cracks it is necessary to amplify this disturbance up to a range that can be measured optically.

Simultaneously, the holographic set-up was rebuilt into a more compact device. With the aid of a new electronic trigger-system it is now possible to switch on the two laserpulses exactly in comparison to the wave generating impact (fig. 1).

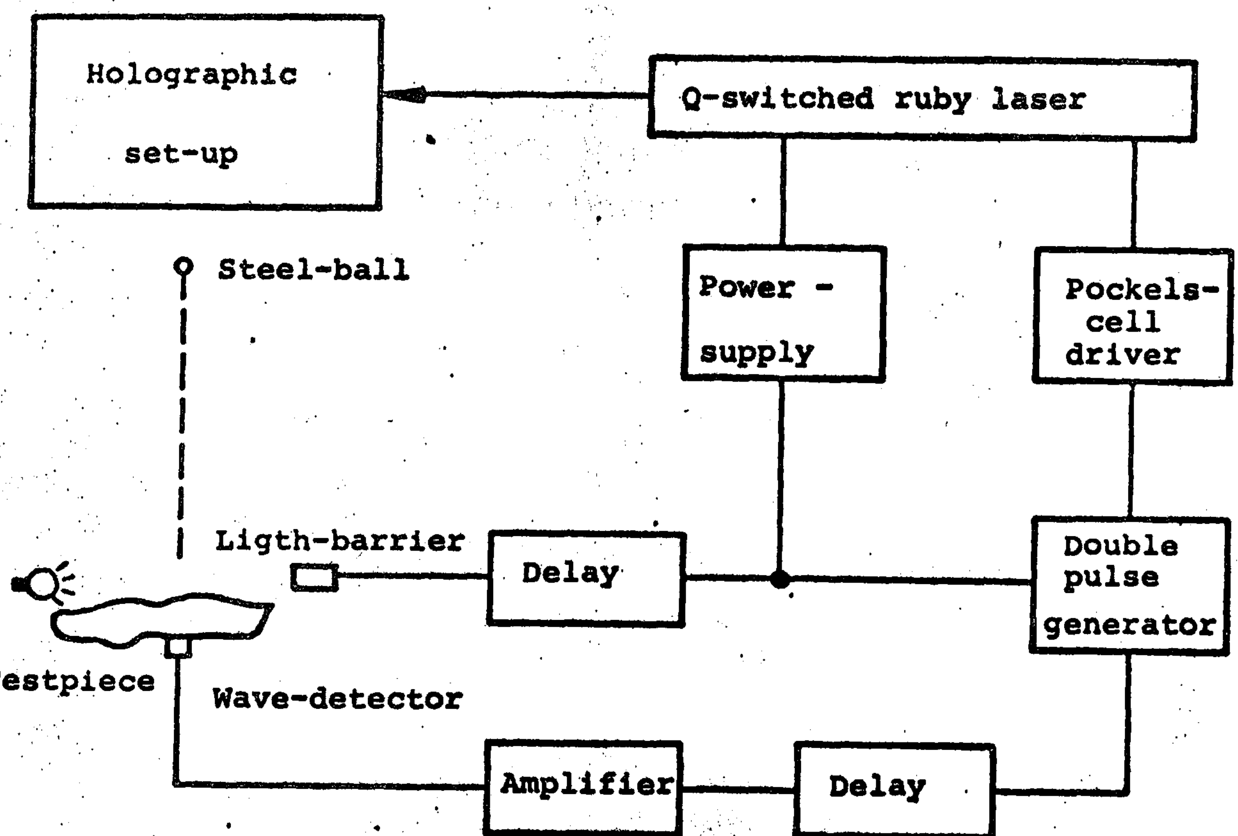

Fig. 1: Experimental set-up

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6. Results

Theoretical and experimental investigations have shown, that the detection of very small cracks in connexion with the described holographic method is limited due to the fact that the wave disturbance is extremly small for crack geometries of interest. Without an adequate magnification of the surface motion or an increased sensitivity of the measuring equipment cracks down to a length of few millimeters and a depth greater than the half of the material thickness are always detectable (fig. 2).

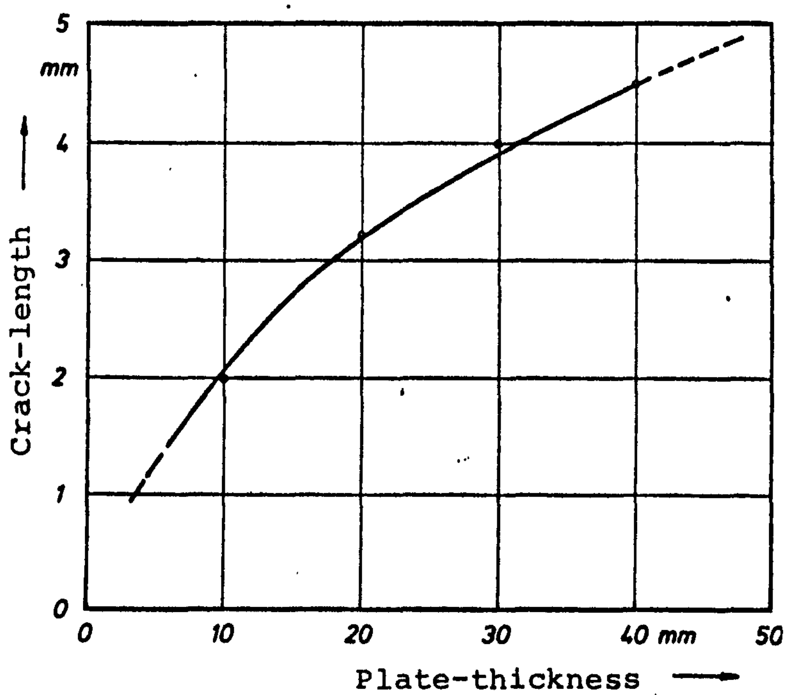

Fig. 2: Holographically detectable cracks without the aid of a liquid-layer (Impact generator: falling steel ball)

In addition it is difficult to obtain a reasonable surface deformation with testpieces thicker than 40 mm (see A 1975). The results with respect to the limited crack geometry and to the limited material thickness could be improved by the aid of a liquid or plastic layer placed upon the testpiece surface. This layer served in amplifing the extremly small wave distortion near small cracks up to a range that could be measured optically. It can be shown theoretically, that the levitation h of the liquid-air interface is

$$h = \frac{4\pi^2}{g} f^2 a_0^2$$

f: Frequency of the generated wave
a_o: Amplitude of the generated wave

With the aid of such a liquid-layer it is possible to detect disturbances of the waves resulting from very small material-voids.

The influence of the liquid layer can be seen in fig. 3 and fig. 4. The testpiece was a steelplate with a thickness of 40 mm. The surface-cracks were simulated by rows of 6 holes with a depth variing from 1 mm to 10 mm. One can see in fig. 3 that without the aid of a liquid-layer no distortion of the interference pattern are visible. But if the surface of the testpiece was covered with a thin water film, all cracks are clearly detectable (fig. 4). Similar results were obtained with a much more thicker steelplate, as it is shown in fig. 5 were a 85 mm plate with a liquid layer of water was tested. To examine another liquid-layer than water, experiments were made with gelatine on the surface of the 85 mm thick steelplate. The result is shown in fig. 6, were all cracks are surrounded by two or more interference-fringes. After these positive results experiments were made with even smaller cracks and as fig. 7 and 8 show, surface-cracks down to a length of 5 mm, a depth of only 0,5 mm and a width of 0,1 mm are clearly visible.

In addition, experiments were carried out with material voids which are located below the surface of the testpiece. As can be seen in fig. 9, these voids are also detectable with holographic interferometry. Because the disturbance of the wave at these cracks are big enough to be measured optically, it was not necessary to use a liquid-layer on the testpiece surface.

Further experiments with testpieces of more complicated geometric shape were made. The crack is located near the welding seam of the flange of an elbow. The interference pattern (fig. 10) show that the crack detection will be not influenced by the complicated geometric shape of this testpiece.

7. Next steps

To improve the detection of even smaller cracks, some investigations will be made with a ultrasonic wave generator, which produces waves in the range of a few MHz. More elaborated measurements with testpieces of any geometric shape in connexion with a new holographic difference-method will be tested to suppress the regular interference pattern and to make visible only the disturbance due to the crack.

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Dimension of cracks shown in fig. 3 - 6 : length 10 mm (6 holes 1 mm Ø in each row)

depth 1 - 10 mm

Fig. 3: Interference pattern of a 40 mm thick steelplate without any liquid layer

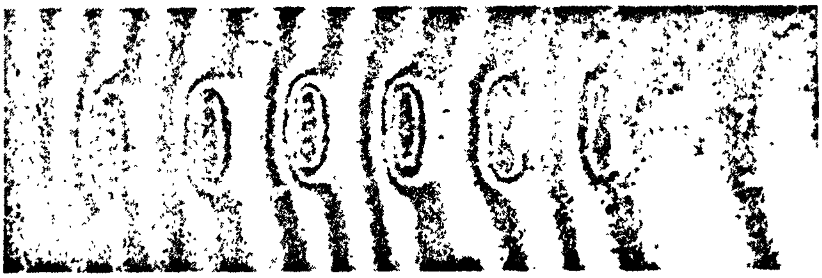

Fig. 4: Interference pattern of the same steelplate as in fig. 3, but with the aid of a thin water-layer. All cracks, even the smallest with a depth of 1 mm are clearly visible

Fig. 5: Interference pattern of a 85 mm thick steelplate. Liquid-layer: waterfilm 2 mm high

Fig. 6: Interference pattern of a 85 mm thick steelplate
Liquid-layer: gelatine 2 mm high

crack dimension: width : 0.2 mm
depth : 0.5 mm
length: 10 mm

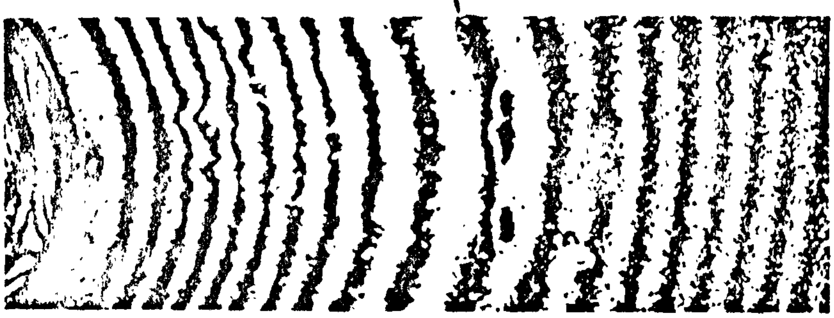

Fig. 7: Interference pattern of a 10 mm thick steelplate
Liquid-layer: gelatine

crack dimension: width : 0.1 mm
depth : 0.5 mm
length: 5 mm

Fig. 8: Interference pattern of a 10 mm thick steelplate
Liquid-layer: gelatine

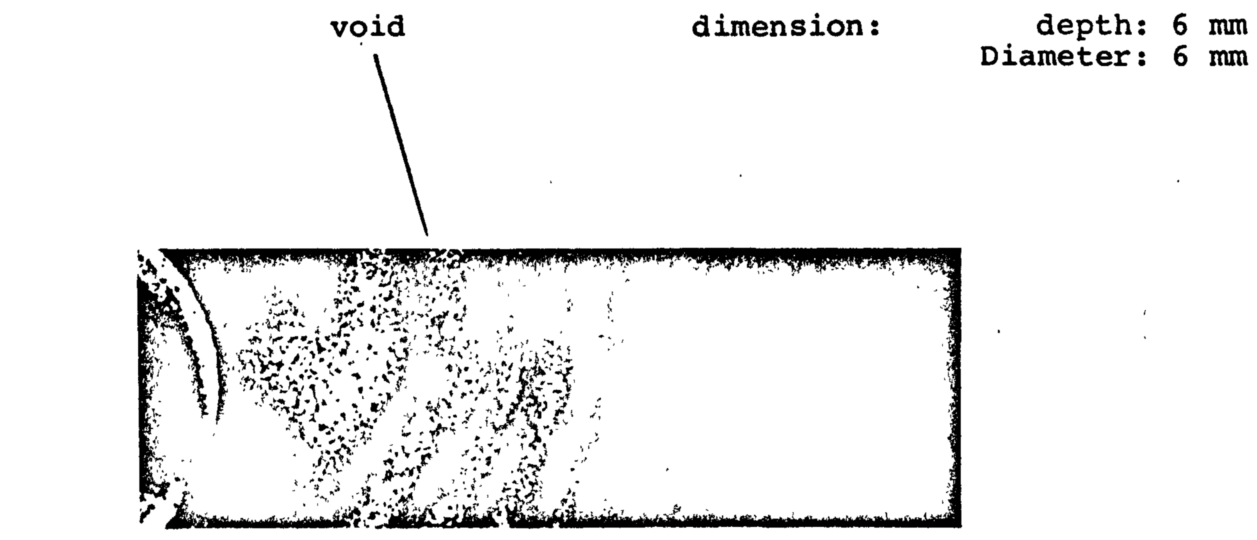

Fig. 9: Interference pattern of a 40 mm thick steelplate with a void below the surface

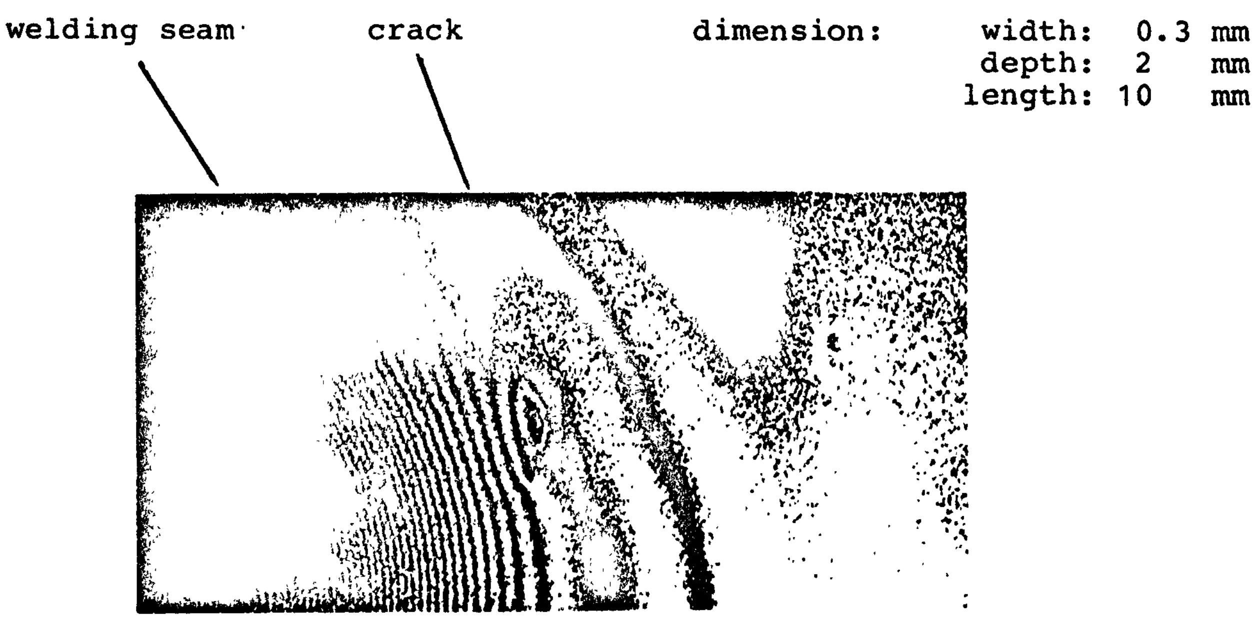

Fig. 10: Interference pattern at an elbow

8. Relations with other Projects

See Annual Reports A74, A75.

9. Reference documents

Annual report A74

Quarterly reports in the series GRS-Forschungsberichte

Report-period	Jan.	1976	-	March	1976	IRS-F-30
Report-period	April	1976	-	June	1976	IRS-F-31
Report-period	July	1976	-	Sept.	1976	IRS-F-33
Report-period	Oct.	1976	-	Dec.	1976	GRS-F-35

10. Degree of availability

The quarterly reports are available by Gesellschaft für Reaktorsicherheit (GRS).

Classification 11.2.3.	
Title 1 Anvendelse af Akustisk Emission	COUNTRY Denmark SPONSOR DAEC et. al. ORGANIZATION DAEC Risø Danish Wel. Inst.
Title 2 Industrial application of Acoustic Emission	Project leader Arved Nielsen
Initiated: 1970 — Completed: Status : progressing — Last updating	Scientists: W. Swindlehurst S.E. Iversen 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.

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5. Next steps

Development of more advanced discriminators and development of sof tware 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.

1383

Classification: 11.2.3.

Title: Akustisk Emission fra stålkonstruktioner	Country: DENMARK
	Sponsor: ECSC
Title: Acoustic Emission from steel structures	Organization: Res. Establ. Risø
Initiated date: 1977 Completed date: Status: In progress	Scientists: A. Nielsen, W. Swindlehurst, C. 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. Experimental facilities and programme: Testing machines and electronic analysis apparatus are available. A facility for pressure testing of vessels is being designed. Laboratory investigations on particular test pieces are planned to compare amount and particular features of the AE produced by different materials (Base metals, weld metals, HAZ) at different conditions (Heat treatment residual stress) under different testing conditions (Temperature, load configuration). Experiments on intermediate size vessels are then planned to investigate to what extent the results gained from the small test pieces are applicable to steel structures.

4. Project status: The testing facilities are ready for laboratory experiments to start.

5. Next steps: The observations from the laboratory experiments will form basis for a programme for vessel testing.

6. Relation with other projects: This project is related to industrial projects on application of the AE technique.

7. Reference documents:

RISØ-M-1896. Arved Nielsen "AE surveillance Methods. AE as a supplementary tool for inspection of reactor pressure components".

8. Degree of availability: As decided by the sponsors (European Coal and Steel Community, Danish Teknologirådet).

1385

Classification
11.2.2/11.2.3

Title 1	COUNTRY Denmark
Spændingsanalyse af primære trykbærende stålkomponenter: Sammenligning mellem beregnede og målte tøjninger og spændinger i en BWR pumpestuts.	SPONSOR DAEC, Risø ORGANIZATION DAEC,Risø
Title 2 Stress analysis of primary steelcomponents: A comparison between calculated and measured stress and strain in a BWR main circulation pump nozzle.	Project leader S.I. Andersen Scientists:
Initiated (date) 74.04.01 Completed: 1977	S.I. Andersen
Status: concluded Last updating(date) April 1977	

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	Classification 11.2.2/11.2.3.
Title 1 Spændingsanalyse af primære trykbærende stålkomponenter: Sammenligning mellem beregnede og målte tøjninger og spændinger i en BWR-pumpestuts.	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2. Stress analysis of primary steelcomponents: A comparison between calculated and measured stress and strain in a BWR main circulation pump nozzle.	Project leader S.I. Andersen Scientists:
Initiated (date) 1974.04.01 Status: progressing Completed: (date) 1976 Last updating (date) February 1976	S.I. Andersen P. Engbæk S. Krenk

Title 1 (Original language) Controllo della integrità dei recipienti in pressione mediante monitoraggio dell'emissione acustica.	Classification 11.2.3.
Title 2 (English) Assessment of nuclear pressure vessel integrity through acoustic emission monitoring.	Country ITALY Sponsor ENEL Organisation CISE
Date initiated October 1971 Date completed 1978 Last updating March 1977	Project Leader 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.

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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.

1391

PROJECT TITLE : RELIABILITY OF CLASSICAL NDT	12.3 11.2.3 LWR 11.1
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : JRC Ispra
DATE INITIATED : 1977 DATE COMPLETED : 1980	PROJECT LEADER : S.J. CRUTZEN

Classification 11.2.3.	
Title 1 Automatisch ultrasoon onderzoek van PWR-componenten	**Country** The Netherlands **Sponsor** PZEM **Organization** KEMA, MatEVal NDT Co
Title 2 Automatic ultrasonic examination of PWR-systems	**Projectleader** R. Huizing
Initiated 1975 **Status** Tests are being performed during shut-down February 1976	**Scientists** H. Jackson (MatEval) V.d. Berg) KEMA De Jong) Boer) Vroman)

1. General aim

Performing inservice ultrasonic inspection on PWR-systems in accordance with the APME-Code and Dutch authorities.

2. Particular objectives

Development of automatic positioning and scanning devices used to carry inspection equipment, such as ultrasonic wheels, to remote areas where manual acces is restricted for reliable inspection operations.
The data acquisition equipment is a special purpose mini-computer to provide optimum data; easy command of the data and mechanical systems; and convenient, easily interpreted, real-time display of all essential information.

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3. Experimental facilities and programme

Pre-tests have been performed in KEMA- and MatEval (Warrington) laboratories.

The programme consists of:

- ultrasonic inspection of circumferential welds and nozzle welds in a pressurizer;
- ultrasonic inspection of circumferential welds and nozzle welds of a steamgenerator;
- ultrasonic inspection of circumferential and longitudinal welds in mainloops.

4. Project status

Tests are being performed during shut-down February 1976.

5. Next steps

The main devices to carry the inspection equipment will not be removed and will be used for the next inspection period.

6. Relations with other projects

Not applicable.

7. Reference documents

Not applicable.

8. Degree of availability

Through the organization KEMA.

1395

NIL & MI - TNO	CLASSIFICATION 11.2.1/2.3
TITLE: Literatuurstudie van materiaalparameters die nodig zijn bij de interpretatie van defect-localisatie met AE-apparatuur	COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs and others ORGANIZATION: NIL, MI - TNO
TITLE: (ENGLISH LANGUAGE): Literature survey of material parameters which are necessary for the diagnosis of defects localized with AE techniques.	PROJECTLEADER: Kloots
INITIATED: September 1974 / LAST UPDATING: April 1977 STATUS: in progress / COMPLETED: May 1977	SCIENTISTS: Boerstoel V.d. Brink

Classification

11.2.3

Title 1 DEVELOPMENT OF ACOUSTIC EMISSION MEASUREMENT	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION REML RISLEY
Title 2	Project Leader P BENTLEY
Initiated 1969 Completed : 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.

1399

Classification	
11.2.3	
Title 1 DEVELOPMENT OF HOLOGRAPHY AND LASER SPECKLE PHOTOGRAPHY	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION SRD CULCHETH
Title 2	Project Leader D J V MARTIN
Initiated 1973 Completed : 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.

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.

Classification: 11.2.4.

Title: Trykprøvningsfacilitet	Country: DENMARK Sponsor: Res. Establ. Risø
Title: Pressure Testing Facility	Organization: Res. Establ. Risø
Initiated date: 1977 **Completed date:** 1978 Status: In progress	Scientists: A. Nielsen 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.

1403

Classification 11.2.4	
Title 1 Onderzoek naar het scheuruitbreidingsgedrag van onderplateringsscheurtjes in PWR's	**Country** The Netherlands **Sponsors** PZEM, GKN, TNO, KEMA **Organizations** KEMA, TNO
Title 2 Evaluation of the behaviour of undercladding cracks in a PWR	**Projectleader** R.M. van Kuijk
Initiated 1971 **Status** Completed 1975	**Scientists** A.P.A.M. Beyers L.B. Dufour H.C. van Elst

1. General aim

Crack extension behaviour of undercladding cracks in heavy section nuclear steel pressure vessel during service.

2. Particular objectives

Investigation of the crack extension in A508-02 under cyclic load conditions, load and temperature simultaneously. Reasurements of crack depths were performed by sampling and statistical approaches.

3. Experimental facilities and programme

The tests have been performed in TNO and KEMA laboratories.
The programme consisted of:

- making a testpiece of A508-2 containing undercladding cracks;
- development of apparentus for realizing simultaneously cycling load and temperature;

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- testing two testpieces with undercladding cracks under cyclic loading (1000 times) and constant temperature;
- testing two testpieces with undercladding cracks under cyclic loading (1000 times) and cyclic temperature from 70° - $300^{\circ}C$ simultaneously;
- sampling the testpieces after the tests, measurements of cracky depths, statistical approaches.

4. Project status

Completed end 1975. It was shown that undercladding cracks grow very little in the base material and no indications were found that the cracks will grow through the cladding.

5. Next steps

None.

6. Relations with other projects

Through TNO with similar projects (TNO, Breda

7. Reference documents

A serie of about 30 technical and progress reports have been prepared. All these reports are written in Dutch.

8. Degree of availability

Through the organization KEMA.

14o5

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 Organisation ENEL
Date initiated 1970 Date completed - Last updating April 1977	Project Leader 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 multi-axial 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.

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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.

1407

	Classification 11.3.2/11.3.4.
Title 1 Brudundersøgelse af Betontank-lågmodeller.	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2 Overload behaviour and ultimate load capacity of PCRV-closures for BWR-plants.	Project leader S.I. Andersen Scientists:
Initiated (date) July 1971 Status: progressing Completed: (date) 1976 Last updating (date) February 1976	N.S. Ottosen S.I. Andersen

1. General aim

To study the overload behaviour, failure mode 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.

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 includes testing of 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 of axisymmetric PCRV-structures, taking into account the effects of concrete creep, plasticity and cracking together with the plastic behaviour of the steel parts.

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4. Project status

9 closure models have been tested during the period between February 1972 and January 1976. Extensive calculations on the unreinforced specimens show good agreement with the experimental observations.

In general the computer programme has been verified thoroughly by the comprehensive test results.

5. Next steps

Final reports are under preparation.

6. Relations with other projects

The programme is part of a joint Nordic development work on a PCRV for BWR application.

7. Reference documents

[1] 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)

[2] Theoretical and experimental studies for optimization of PCRV top closures.
N.S. Ottosen, S.I. Andersen.
Paper H 3/6. Trans. of the 3rd Int. Conf. on Struct. Mech. in Reactor Technology, London Sept. 1975.

8. Degree of availability

Project information restricted.

1409

Classification
11.3.2/11.3.4

Title 1 Brudundersøgelse af Betontank-lågmodeller	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2 Overload behaviour and ultimate load capacity of PCRV-closures for BWR-plants.	Project leader S.I. Andersen Scientists:
Initiated (date) Completed: 1976 Status: concluded Last updating (date) April 1977	N.S. Ottosen 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

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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
To be published in 1977 in Journal of the Engineering Mechanics Division, Proceedings of the American Society of Civil Engineers.

Structural Failure of Thick-Walled Concrete Elements.
N.S. Ottosen, S.I. Andersen.
Paper H 3/d, 4. Int. Conf. on Structural Mech. in Reactor Technology. San Francisco (1977).

8. Degree of availability Project information restricted.

1411

	Classification 11.3.2/11.3.4
Title 1 **Brudundersøgelse af Betontank-lågmodeller**	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2 Overload behaviour and ultimate load capacity of PCRV-closures for BWR-plants.	Project leader S.I. Andersen Scientists:
Initiated (date) Completed: 1976 Status: concluded Last updating (date) April 1977	N.S. Ottosen S.I. Andersen

Title 1 (Original language) Sviluppo di soluzioni avanzate per i recipienti in pressione in calcestruzzo precompresso (soluzioni a pareti sottili).	Classification 11.3.2 11.3.4
Title 2 (English) Development of advanced solutions for Pre-stressed Concrete Pressure Vessels (Thin-wall solutions).	Country ITALY Sponsor ENEL Organisation ENEL, ISMES, CISE
Date initiated 1970 Date completed -- Last updating April 1977	Project Leader F. Scotto

1. 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.

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Title 1 (Original language) Sviluppo di soluzioni avanzate per i recipienti in pressione in calcestruzzo precompresso (soluzioni a pareti sottili).	Classification 11.3.2 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.

Title 1 (Original language) Sviluppo di soluzioni avanzate per i recipienti in pressione in calcestruzzo precompresso (soluzioni a pareti sottili).	Classification
	11.3.2 11.3.4

8. Degree of availability
Exchange of information can be arranged.

9. Budget
About $60 \cdot 10^6$ Lire/year.

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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 Sponsor ENEL Organisation ENEL, ISMES, CISE
Date initiated 1970 Date completed -- Last updating April 1977	Project Leader F. Scotto

1419

	Classification $\boxed{7.1}$/10.4/11.2/11.4
Title 1 BEAMDYN - Program til lineær 3-dimensionel statisk og dynamisk analyse af bjælkestrukturer.	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2 BEAMDYN - Program for the linear 3-dimensional static and dynamic analysis of beam type structures.	Project leader Per Lundsager Scientists:
Initiated (date) November 1975 Status: progressing Completed: (date) 1976 Last updating (date) February 1976	Per Lundsager

	Classification $\boxed{3.2}$/11.4.
Title 1 Flystyrt på containment-bygning	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
Title 2 Aircraft impact on containment building.	**Project leader** Per Lundsager Scientists:
Initiated (date) March 1975 **Status:** progressing **Completed:** (date) 1976 **Last updating** (date) February 1976	Per Lundsager S. Krenk

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	CLASSIFICATION 11.4.1
TITLE 1 ENCEINTE DE CONFINEMENT POUR PWR. DOUBLE ENCEINTE SANS PEAU D'ETANCHEITE.	COUNTRY FRANCE SPONSOR E.D.F. ORGANIZATION E.D.F.
TITLE 2 PRIMARY CONTAINMENT FOR PWR WITHOUT METALLIC SKIN FOR TIGHNESS.	Project Leader E.D.F./SEPTEN/M Scientists M. COSTAZ
Started 1970 Completed 1975	
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 à 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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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.

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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.

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Classification: 11.5	
Title 1 (Original Language): Untersuchungen zum Einfluß der Größe und Form von Kühlkanalblockaden auf die Kernnotkühlung in der Flutphase eines Kühlmittelstörfalles (PNS 4239 - I.1.3, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** **ORGANIZATION:** GfK Karlsruhe Projekt Nukleare Sicherheit
Title 2 (English): Influence of the Size and Shape of Coolant Channel Blockages upon Core Cooling in the Flooding Phase of a LOCA	**Project Leader:** S. Malang/IRB
Initiated (Date): 1972 **Status:** Continuing	**Completed (Date):** 1979 **Last Updating (Date):** December 31st, 1976

1. General Aim

The influence of coolant channel blockages upon the cooling effect during the reflood phase of a loss of coolant accidents 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 the blockages. Not considered is the interaction between deformation and cooling conditions.

3. Research Program

The investigations will be done with a single rod, a row of 5 rods, and a bundle of 5 x 5 rods. The used indirect electrically heated rods have the same geometry like the fuel rods and a stepwise cosine shaped axial power profile.

The experiments are performed in three major steps concerning the shape of blockages:

a) without blockages (reference tests),
b) grid plate,
The free cross section for the coolant is varied by different diameters

of the boreholes in the center of the coolant subchannels.

c) exchangeable balloons attached to the outer rod surface,
The outer diameter, the length, and the axial profile of the balloons are varied.

The mass flow of the flooding water and the back pressure at the outlet of the test section are kept constant for each test. The rod power is following the ANS-Standard for decay heat, where the level is a parameter. Test parameters are:

a) initial rod temperature,
b) mass flow of the flooding water,
c) temperature of the flooding water,
d) back pressure at the outlet of the test section,
e) level of the decay heat.

The cladding temperatures and the pressure drops are measured. At three axial levels high speed movies are taken which serve as an important tool for the inpretation of the two phase flow.

4. Experimental Facilities, Computer Codes

The main components of the test rig are:

- water tank,
- pump,
- throttle valve,
- lower plenum,
- test section with electrically heated rods,
- upper plenum with pressure-regulating valve,
- thyristor controlled power supply,
- digital data acquisition system,
- high speed film camera with a flash light equipment.

An available computer code was improved and adjusted for the calculation of the heat transfer coefficient using the measured clad temperatures.

5. Progress to Date

After the shake-down period of the test loop and the data acquisition system the experiments with a row of 5 heater rods were started.

The first test series without blockages has been completed. It serves as baseline experiments.

Tests with a grid plate to simulate coolant channel blockages have been started. In the first step which has been completed the reduction in the coolant channel cross section area was 62 %.

6. Results

The analysis of the experiments was started. A comparison between the tests with free coolant subchannels and those with subchannels partially blocked by a grid plate show that at the same mass flow rate the heat transfer was always better behind the blockages. However, in front of the grid plate the heat transfer was nearly uninfluenced. That result is not relevant for rod bundles in which only a part of the subchannels is blocked. For other types of subchannel blockages the result is not applicable without modifications.

7. Next Steps

The tests with a row of 5 heater rods and a grid plate simulating coolant channel blockages are continued. In the next step the influence of balloons attached to the outer surface of the rods is investigated. Tests with a 5 x 5 rod array are prepared.

8. Relation with other Projects

FLECHT-Tests, performed by Westinghouse
Low pressure tests, performed by KWU.

9. References

Semiannual Reports of the Nuclear Safety Project.

10. Degree of Availability of the Reports

Unrestricted Distribution.

Classification: 11.5	
Title 1 (Original Language): Blockierte Kühlkanäle (RS 194 - I.1.3, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Blocked Cooling Channels	Project Leader: H. Hein
Initiated (Date): 1. 1. 75 — Completed (Date): 31. 12. 77 Status: Continuing — Last Updating (Date): 31. 12. 76	

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 upsteam and downstream of the blocked region as a function of the geometry of the blockage.

3. Research Program

3.1 Planning and preparation of blockage tests
3.2 Modification of the test facility for BWR geometry blockage tests
3.3 Assembly of the bundle in the rod bundle container
3.4 Pre-tests of mass distribution measurement techniques in the core
3.5 Performance tests
3.6 Evaluation of tests

4. Test Facilities

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 tests will be performed on the 2nd BWR double bundle under the scope of R & D Task RS 36 C. The double bundle was prepared for the installation of blockage sheets and the type of blockage to be used was determined. The blockage provides for local blockages of 75 % and 39 % respectively. The blockage plates were fabricated.

The next step was the assembly of the blockage sheets in the 2nd BWR double bundle. The ζ-value of the blockage was determined for Reynold's numbers between 10^4 and 10^5, followed by the assembly of the double bundle in the supporting structure (cage) and rod bundle container. The instrumentation lines were led from the container and sealed.

The flooding and spray systems were connected to the feed lines. Partial start-up of the plant and check-out of the measurement technique was initiated.

The construction of the test rig for developing and evaluating special measuring techniques is completed. The main components such as pump and power supply have been ordered.

6. Results

Measurement of the resistance coefficient of the blockage in the bundle resulted in a ζ-value of 5 ± 0.1 within the Reynold's number range of 10^4 to 10^5.

The plant has nearly been completed so that start-up could be initiated.

7. Next Steps

- start-up of plant
- performance of 10 tests using the first blockage type
- evaluation of tests
- decision on 2nd blockage type
- tests using the 2nd blockage type
- erection of the model test rig
- performance of measurement technique tests.

8. Relation to Other Projects

Preparatory work has been performed within the scope of R & D Tasks RS 36, RS 36 A, RS 36 B, RS 36 C.

Experiments pertaining to fuel rod performance are being conducted by both the project "Nukleare Sicherheit" at GfK as well as within R & D Task RS 107 and RS 185.

9. References

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10. Degree of Availability

1435

Title 1 (Original language)	Classification
Effetti dello scoppio di tubo a pressione CIRENE.	11.5
Title 2 (English)	**Country** ITALY
CIRENE power channel pressure tube burst tests.	Sponsor : CNEN Organisation : CISE
Date initiated: October 1970 Date completed: 1978 Last updating: May 1977	**Project Leader** 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

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Classification: 12.1	
Title 1 (Original Language): **Statusbericht Qualitätssicherungssystem** **- Darstellung des Istzustandes -** **(RS 124 - II.3.1,** Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: **KWU Erlangen**
Title 2 (English): **Status Report of the Quality Assurance System**	Project Leader: **Dr. Kaden**
Initiated (Date): 1. 8. 75 Completed (Date): 31. 1. 77 Status: Completed Last Updating (Date): 31. 12. 76	

1. General Aim

In a study the present state of the quality assurance system for design, manufacturing and examination, procurement, shaping erection and commissioning of the components (NDES) including nuclear auxiliary and ancillary components and the containment will be described.

2. Particular Objectives

All components were considered, which are used in the nuclear steam generating system.

3. Research Program

a) Listing of various steps of the work for the representation of the present state.

b) Detailed description of the activities, the boundary conditions and the quality assurance securing.

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4. Experimental Facilities

No experimental facilities necessary.

5./6. Progress to Date and Results

The progress report was revised and discussed with members of the IRS. The revised version was completed.

7. Next Step

The work has been completed.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

TITLE 1 (original language) Programma di garanzia della qualità	Classification 12.1
TITLE 2 (english) Quality assurance program	Country: ITALY Sponsor: Franco Tosi Organisation: Franco Tosi
Date initiated January 1, 1972 Date completed June 6, 1975 Last updating	Project Leader 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 Sponsór, 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

1441

PROJECT TITLE : RELIABILITY STUDIES	CLASSIFICATION 12.1
SPONSORING COUNTRY : ITALY	ORGANISATION : CNEN
DATE INITIATED : October 1974 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.

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PROJECT TITLE : RELIABILITY OF CLASSICAL NDT	12.3 11.2.3 LWR 11.1
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : JRC Ispra
DATE INITIATED : 1977 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

31.719

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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 collabmration 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.Erienne (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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Classification: 12.3

Title 1 (Original Language): Untersuchungen über die Anwendbarkeit der Ultraschall-Impulsspektrometrie zur Verbesserung der Aussagesicherheit bei der Materialprüfung mit Ultraschall (RS 54 - II.3.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Bundesanstalt f. Materialprüfung
Title 2 (english): Ultrasonic Pulse-Echo Spectroscopy in Ultrasonic NDT	Project Leader: Prof. Dr. Mundry
Initiated (Date): February 10, 1972 Completed (Date): February 10, 1975 Status: finished Last Updating (Date): December 1975	

1. General aim

The project aims at the complete evaluation of the information furnished by ultrasonic echoes.

2. Particular objectives

By means of frequency analysis type, size and orientation of a defect in materials of larger thicknesses (reactor pressure vessels) shall be determined.

3. Experimental facilities and program

Spectrum analysis can be performed with two different devices. The first device is a swept frequency RF-receiver, which is plugged into an oscilloscope mainframe. The other device is a digital Fourier Analyser, which calculates real and imaginary part (or amplitude and phase) of a spectrum of a given pulse. Additionally a set of broad band (shock wave) probes, ultrasonic transmitter/receivers, a set of steel specimen, an immersion tank and an electronic switch for time windowing are available.

During the experiments the echoes of artificial defects of different type, size and orientation are to be analysed, to study the influence of these parameters on the frequency spectra.

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4. Project status

4.1. Progress to date

The project has been finished. During the last two months of its duration investigations on the influence of the angle of incidence on amplitude and phase spectra took place. The experiments were carried out in immersion technique so, that the test reflectors were turned to vary the angle of incidence from normal incidence in steps of 0.2°. The distance between transducer and reflector was held constant.

4.2. Essential results

A result of the measurements is shown in the picture. The angle of incidence for a 20 mm circular flat reflector was varied in the range of 0° to 10° in steps of 0.2 degrees. The upper part of the picture shows the time function of six of the recorded echoes in the range of 0° to 5° in steps of 1°.

The second line of drawings shows real and imaginary part of the spectra of the above pulses after Fourier transform.

Next the mountains of spectra are cut through along the drawn plane, which is normal to the frequency axis. In these two planes one can find the curves of the third line of drawings. The left curve gives the real part and the right curve gives the imaginary part of an amplitude distribution versus the angle of incidence. In a rough analogy to holography a reconstruction of the reflector may be obtained from that distribution by inverse Fourier transform. The width of the peak of the last curve is strictly proportional to the width of the reflector.

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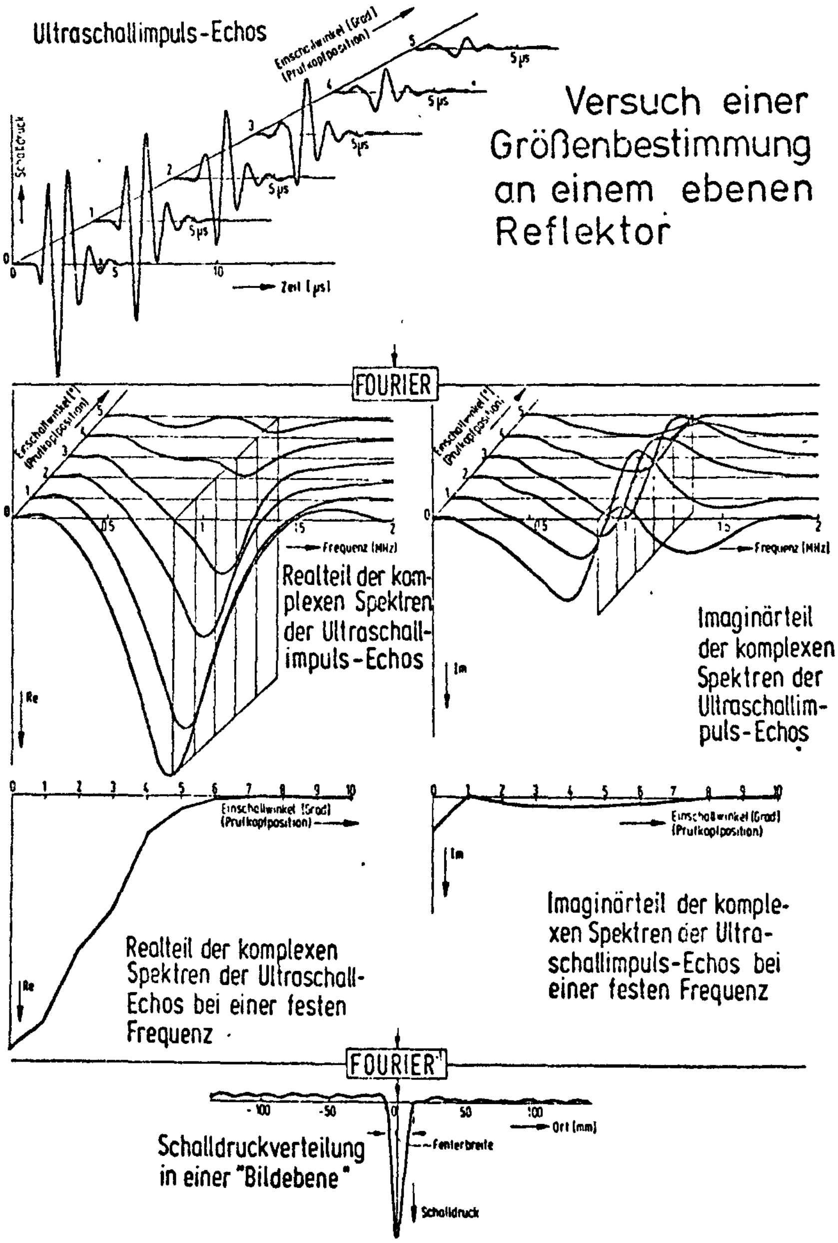
Versuch einer Größenbestimmung an einem ebenen Reflektor



1431

Classification: 12.3	
Title 1 (Original Language): Studie über den derzeitgen Stand der Schallemissionsanalyse (SEA), ihre Grenzen und Möglichkeiten auf dem Gebiet der Reaktorsicherheit (RS 0031 D - II.3.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Battelle-Institut Frankfurt/Main
Title 2 (english): Literature Survey of the State of the Art of Acoustic Emission, its Potentiality and its Limits in the Field of Reactor Safety	Project Leader: Dr. Eisenblätter Dr. Jax
Initiated (Date): July 1, 1975 — Completed (Date): November 30, 1975	
Status: finished — Last Updating (Date): December 1975	

1. General Aim

The aim of these evaluations is to present critically the status of knowledge and technology of acoustic emission for testing of thick-walled components, especially those of nuclear reactor vessels.

2. Particular Objectives

This literature survey shall be of use
- to present completely the results of the work done so far at Battelle Frankfurt during the past six years
- to compare this knowledge with the status of research in the Federal Republic of Germany and other countries
- to mark out the problems still existing.

3. Research Program

The results of our own experimental work which are represented in more than 20 technical reports (reference RS 0031 to RS 0031 D) have been taken into consideration as well as the relevant literature published lately.

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4. Project Status

The investigations are terminated. The corresponding report will be completed in February 1976.

4.2. Essential Results

First in literature the knowledge in seismology concerning the source mechanisms of earth-quakes and the propagation of earth-quake waves have been applicated on acoustic emission. Thus it was possible to understand the most important fundamentals of acoustic emission originated by different processes (crack formation, crack propagation, plastic deformation, friction) and to present them in a consistent form. Besides presenting these fundamentals, the scope of this survey was to show the still existing problems, especially in inspection of thick-walled vessels. Nowadays these are mainly concentrated in the classification of located defects. Related investigations conducted until today mostly do not furnish valuable information about this problem because of the following reasons:

- structures with unknown defects did not contain critical defects; those defects which have been found are not large enough to be identified unambigiously by other NDT methods; destructing methods are not possible,
- in the case of specimens with intentionally built-in defects mostly the natural defects occuring in structures have not been simulated correctly so far.

Therefore for the future special attention should be drawn on fundamental evaluations of thick-walled vessels with built-in natural defects.

5. Next Steps

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6. Relation with Other Projects

This literature survey represents the completion of the following projects: RS 0031, RS 0031 A, RS 0031 B, RS 0031 C.

7. Reference Documents

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Classification: 12.3	
Title 1 (Original Language): Zerstörungsfreie Prüfung des Gefügezustandes mittels Ultraschallrückstreuung (RS 102-16/1 - II.3.2., Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Ges. München, IzfP
Title 2 (English): Non-destructive structure evaluation by means of scattered ultrasound	Project Leader: Dr. G. Deuster Dr. K. Goebbels
Initiated (Date): 01.05.1973 Status: Continuing	Completed (Date): 31.12.1977 Last Updating (Date): December 1976

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 andvanced 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, the accumulation of impurities in certain material areas and the proof of inner tensions.

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:

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- 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 US 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 of beam angle) are under consideration.

3. Research Program

In 1976, there were the following points of special interest:

a) test of the new device for scattering measurements in an application test with about 2oo steel specimens;

b) determination of the impurities in several reactor pressure vessel specimens;

c) measurements on austenitic materials and welds for better understanding of the steel structures and for obtaining a better signal-to-noise level in ultrasound examinations;

d) measurements with several averaging methods applying the basic laws of the US scattering and the results of the measurements mentioned under c).

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 two apparatus:
  - at first a device built by Koppelmann[1], recording the analog scattering signal on a XY recorder.
  - A second apparatus 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 US pulse generating system differs from that one of the second apparatus (see above); the signal processing system is the same.

5. Progress to Date

The application test has been performed completely in 1976; the qualitative evaluation of the scattering curves allows to determine whether the structure is homogeneous or inhomogeneous; from measurements on austenitic materials there could be developed several averaging methods to improve the signal-to-noise ratio.

6. Results

- In the application test about 200 steel specimens are examined. The scattering curves were evaluated and the scattering coefficient and the grain size were determined from the attenuation coefficient. The error bonds are plus minus half an ASTM grain size class.

- The determination of impurity areas within the specimen was done by evaluating the scattering curves only in a qualitative manner. If there are any inclusions the sound reflexions outside this area are superposed to the scattering curve.

- To improve the signal-to-noise ratio when testing coarse-grained materials, especially austenites and its welded joints several averaging methods were applied. By varying the probe position, the excitation frequency or the beam angle, the interference pattern with respect to its maximum-minimum distribution over the

time axis is varied, too. If there is any signal from a defect it will stay with nearly the same amplitude at the same place in the pattern. By the superposition of several interference structures, the useful signal grows more than the noise whereby the signal-to-noise ratio is improved.
When applying the basic laws of the US scattering theory, the averaging curves can be interpreted in a quantitative manner; the flaw size of the defects can be determined by using the DGS method after Krautkrämer.

7. Next Steps

Beside a further examination of the several averaging methods there will be a main effort in improving the local resolving power when examining welded joints by means of focusing probes. In 1977 the project will be finished.

8. Relation with Other Projects

No essential relations with other projects are present, only the work concerning austenitic steels in RS 102-16/1 and RS 143 are tuned together.

9. References

K. Goebbels: Materialprüfung 18 (1976) 3, 86-88

S. Kraus, K. Goebbels:
Signalmittelungsverfahren zur Unterdrückung des kohärenten Untergrundes bei der Ultraschallfehlerprüfung grobkörniger Werkstoffe.
IzfP-Bericht 760717-TW

Classification: 12.3	
Title 1 (Original Language): Anpassung und Weiterentwicklung von elektrischen, elektromagnetischen und magnetischen Prüfverfahren für den Einsatz an Reaktoren (RS 102-18 - II.3.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Ges., IzfP
Title 2 (English): Adaption and continuation of the development of electrical, electromagnetic and magnetic test methods for the use at reactors	Project Leader: Dipl.-Phys. R. Becker Dr. W. Mohr Dipl.-Phys. G. Dobmann
Initiated (Date): 1.8.1973 Status: Part 1 finished, parts 2,3,4 continuing Completed (Date): 30.9.1976, Part 2,3,4: 30.9.1977 Last Updating (Date): 31.12.1976	

General aim

In the field of the electrical, magnetic and electromagnetic test methods the project has the general aim

- to transfer the status of knowledge into the concepts of techniques and devices,
- to expand the foundations in mathematics and physics as much as necessary,
- to establish rules and models for interpretation of measurement results,
- to adjust the techniques to the specific problems in reactor testing.

Based on the research results, special developments orientated towards application are to follow as individual projects.

1. Eddy Current Test Method

1.1. Particular Objectives and Research Program

We are working at two kinds of application:

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a) testing of installed heat exchanger tubes,
b) testing of the claddig, the basic material, the weld and austenitic/ ferritic junctions of the reactor pressure vessel for location of cracks near the surface.

In both cases a great number of parameters give measuring effects. The signals of defects which generally are the relevant parameters can be superimposed by signals of other parameters, so that the presence of the defects cannot be recognized or the indication of the defect depth can be falsified. Therefore the signals from the disturbing parameters must be suppressed and only the signals from the defects must be given out with the indication of the defect depth.

To solve this task the multifrequency method is to apply where is worked with several frequencies simultaneously. This method is to develop further with regard to the practical application and to test in several test situations which are simulated by models.

1.2. Experimental Facilities and Computer Codes

a) Evaluation and application of computer programs to determine the impedance of the test coil as a function of the electrical, magnetic and geometrical properties of the specimen, as well as a function of the dimensions of the coil; the specimen can be a tube, a rod or a slab consisting of several layers.

b) Construction of a laboratory test device which works simultaneously with up to 4 frequencies in the range of 10 - 500 kHz.

c) Application of this device for the above mentioned test situations and optimization of the evaluation method to obtain the read-out value.

1.3. Progress Today

a) Improvement of the test device (in 1.2.b) concerning stability and handling.

b) Continuation of the experiments with tubes, claddings and welded joints.

c) Completion of the computer program in 1.2.a).

1.4. Results

The project has been finished. In laboratory tests it could be demonstrated that the multifrequency method is adequate to solve the difficult problems in testing heat exchangers and other components of nuclear plants. A test device exists, which is the starting point for the construction of a prototype suited for the "in-situ" application. Just so there is a computer code which can be used to optimize the test equipment (frequency, design of the coils, etc.) for several test situations.

1.5. Relations With Other Projects

Concerning the testing of claddings and welded joints there is a relation to project RS 89.

1.6. References

Final report, in preparation.

2. Electrical Resistance Probe Method

2.2. Particular Objective

The method is used for crack-depth measurements of surface cracks in metallic materials. By contacting the body with potential probes an electrical flaw field is impressed. The associated potential field, measured on the surface is a function

- of the electrical conductivity,
- of the distance of the potential probes and
- of the surface geometry.

Every surface crack disturbs this geometry and therefore the potential distribution. The normalized voltage U/U_o measured above the crack is

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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.

2.3. Research Program

2.3.1. Numerical solution of two-dimensional electrical potential field problems.

2.3.2. Experimental tests of the numerically computed crack-depth functions for real reactor components.

2.4. Experimental Facilities, Computer Codes

2.4.1. Implementation of a finite-element FORTRAN IV program for numerical solution of Neumann boundary value problems.

2.5. Progress Today

The finite element program was improved, it allows the solution of potential problems for two-dimensional geometries with arbitrary boundaries and variable sources of the field.

2.6. Results

Application of the program on surface crack geometry in screw bolts of reactor pressure vessel yields to good agreements with model measurements. For CT-standard test specimens of fracture mechanics the form and manner of the electrical sources could be optimized for studying crack growth measurements.

2.7. Next Steps

The treatment of surface cracks in claddings and welded joints at a nozzel in reactor pressure vessel.
Development of a finite element program for three-dimensional geometries.

2.8. Relation with Other Projects: none

2.9. References

Dobmann, G.:
Potentialsondenverfahren. Berechnung und Messung zweidimensionaler Potentialverteilung bei homogener Anregung.
IzfP-Bericht 750102-TW

Zwischenbericht Potentialsondenverfahren. Numerische und experimentelle Untersuchungen zum Teildurchströmungsverfahren.
IzfP-Bericht 760214-TW

2.10. Degree of Availability of the Reports

Materialprüfung 18 (1976) 9, pp. 342-344

3. Magnetic Leakage Flux Method

3.1. Particular Objektive

For ferromagnetic materials which are flowed through by means of direct magnetic field produced 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 signals.

3.2. Research Program

3.2.1. The development of a magnetography signal measure and interpretation unit.

3.2.2. Theoretical examinations about the influence of surface crack parameters as crack-length, -depth and width and of the permeability by solving numerically a Fredholm's integral equation.

3.3. Experimental Facilities, Computer Codes

3.3.1. Fundamental experiments with magnetography are made.

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3.3.2. A FORTRAN IV program for iterative solution of integral equation is implemented.

3.4. Progress Today

First numerical results.

3.5. Results

The first iterative solution shows the influence of crack width in the distance of the local extremas in the vertical magnetic flux component.

3.6. Next Steps

The plan is the non-destructive testing of welded joints in reactor containments with magnetography-leakage-flux method. For this application a signal processor must be built.

3.7. Relations With Other Projects: none

3.8. References

Höller, P.:
Elektrische und magnetische Verfahren zur zerstörungsfreien Werkstoffprüfung.
In "Neuzeitliche Verfahren der Werkstoffprüfung", Verlag Stahleisen mbH
Düsseldorf 1973, Seite 139

4. Guided Ultrasonic Waves Electrodynamically Excited

4.1. Particular Objectives and Research Program

A technique will be elaborated which allows to inspect heat exchanger tubes for longitudinal and transverse cracks by guided waves. The guided waves will be excited and received contactlessly by electrodynamic transducers.
The contactless electrodynamic excitation shall also be used for free ultrasonic waves in order to develop an angle probe for austenitic specimens.

4.2. Experimental Facilities

The equipment for tube testing has been in essential the same as in 1975 with the difference that the transducer coils are shortened to improve the local resolution of flaw echoes. For the excitation of free ultrasonic waves a pulse magnetization technique was developed which allows the generation of a high short-time magnetization power in electromagnets of small size.

4.3. Project Status

4.3.1. Progress Today

Since the dispersion of guided waves generated in thin-walled tubes is very small in the frequency range of interest, short transducer coils with large spatial bandwidth could be developed in order to improve the axial local resolution of flaw echoes.

For the contactless electrodynamic generation of free ultrasonic waves meander-transducers with their fitting electromagnets were developed to be used as angle probes for ferritic specimens, the angle of incidence of the ultrasonic waves being adjustable by electronic frequency change. To suppress the side lobes in the radiation patterns some tapering techniques were tested.

4.3.2. Essential Results

Concerning tube waves the axial resolution of flaw echoes could be trebled. Concerning the contactless electrodynamic generation of free ultrasonic waves a dynamic range of 60 dB could be reached at an angle of incidence of 45°. By frequency change it was possible to vary the angle of incidence from about 25° to 50°. A Dolph-Chebychev-tapering involved a suppression of the side lobes of about 15 to 25 dB in the radiation patterns.

4.4. Next Steps

The development of a method for tube testing by guided waves is carried

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on in another project. The dynamic range of free ultrasonic waves generated by contactless electrodynamic transducers shall be improved in order to be available for austenitic specimens, too.

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Classification: 12.3	
Title 1 (Original Language): Automatisierung und EDV der US-Impulsechoprüfung (RS 102-17 - II.3.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Frauhofer-Gesellschaft, IzfP
Title 2 (English): Automation of Pulse-Echo-Testing and Electronic Data Processing of the Results	**Project Leader:** O.A. Barbian
Initiated (Date): 15.10.1974 **Status:** Continuing **Completed (Date):** 31.12.1977 **Last Updating (Date):** December 31, 1976	

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 threedimensional flaw picture dependent upon the different testing methods. These flaw pictures are compared with the specifications and standards.

2. Particular Objectives

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.

3. Research Program

The following steps have to be carried out:

- utilization of information included in echo-dynamics (local curve of transit time, local curve of amplitudes),

- utilization of information included in phase and amplitude of the signal,

- 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,

- considerations, 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 threedimensional flaw picture, which allows sections in any direction.

4. 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.

5./6. Progress to Date and Results

For the reconstruction of flaws in the far-field of piston diaphragms mathematical models have been developed which use the signal locus curves for evaluation. These models are able to consider simultaneously the indications of different ultrasonic test methods, as for example tandem and pulse-echo. By that means the distinction between plane and voluminous defects is possible. For plane defects the mathematical model has been expanded already so much that defect size and inclination can be reconstructed from the amplitude locus curves.

The IPW in Freiburg has started the development of the software for the mentioned evaluation and reconstruction. The following program parts are completed already:

- 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.

Concerning the ultrasonic testing electronic (hardware), the logarithmic amplifier, the multiplexer, the filters and the transmitter are completed and tested both, with bursts and with original ultrasonic signals. Moreover, the time synthesizer and the peak detector are finished and tested with bursts.

7. Next Steps

The ultrasonic hardware components will be completed and built up. Hardware and software will be tested with real and natural signals and - if necessary - improved.

8. Relations with Other Projects

This programme is partly based on the results of the reactor safety research RS 27/1 and RS 27/2 and will be carried through in coordination with the still running projects RS 2703 and RS 169.

9. References

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Classification: 12.3	
Title 1 (Original Language): Holografische Interferometrie (RS 102-22 -I.I.3.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: FhG-IfaM
Title 2 (English): Holografic Interferometry	**Project Leader:** Dr. W. Jüptner
Initiated (Date): 1.7.1975 **Status:** completed **Completed (Date):** 30.9.1976 **Last Updating (Date):** December 1976	

1. General Aim

Using holografic interferometry, the field of deformations of workpiece surfaces is registered by interferometric comparison of two states stored in a hologram. The appearing interference fringe system yields quantitative informations about the deformation in three axes. To produce the two stages the unloaded object has to be holographed, consequently it is subjected to the normal load and holographed again.

The field of deformation, obtained in this manner, can be used for calculation of the states of stress even of geometrically complicated workpieces. Furthermore by qualitative and quantitative evaluation weak points become evident. In fracture mechanical tests informations about the stress intensity could be obtained from experimentally determined plastic zones, and thus about the influence of crack displacement.

2. Particular Objective

It appears reasonable to apply the above mentioned advantages of holografic interferometry for safety controls of reactor components. However, before this method can be employed for components with possible interieur cracks or other damages, on the basis of this research project the following problems have to be solved:

a) Can defects be detected in components with thick walls?

b) Which are the conditions for critical stresses to be determined from the holografic fringe system?

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c) Is it possible to recognize critical defomations in the region of weld seams?

d) Which equipment is suitable for measurements whithout vibration protection?

These points shall be investigated in this project.

3. Research Program

3.1 Investigations to produce holograms without vibration isolation

3.1.1 Holografic equipment with an output coupled laser

3.1.2 One-beam holografic equipment

3.1.3 Holografic equipment with a phase reference mirror

3.2 Deformation measurements of a specimen with internal cracks

3.2.1 Construction of a tensile testing machine for holografic applications

3.2.2 Experimental evaluation of the defomation field.

4. Experimental Facilities

4.1 Holografic equipment
including an Ar-Ion Laser with an acousto-optical
output coupler

4.2 Holografic equipment
Calculator Tektronix Scientist 909
Tensile testing machine 120 ton

5. Progress to date

5.1 Investigations to produce holograms without vibration isolation

An important condition during the taking of a hologram is that for the period of the exposure of the hologram, the relation of the two lightways with respect to the object and to the reference beam is only allowed to change by less than one tenth of the light wave length; that means less than approximately 10^{-4}mm:

$$S_{ob} - S_{ref} \quad 0,1$$

S_{ob} : Change of object beam path

S_{ref} : Change of reference beam path

This condition may be satisfied in different ways, depending on the case and whether vibration isolation of the object is possible:

a) The hologram can be taken using a powerful laser, so that the object movement is small enough during the exposure / 1 /.

b) The laser output can be controlled by the object movement. This leads to a stroboscope-like exposure, but without a well-defined frequency.

c) The phase difference can be held constant while taking the hologram by a strong coupling of the two light waves. This means that the reference-wave will be modulated by the object / 2, 3, 4, 5 /.

Based on a theoretical evaluation the following methods were investigated experimentally.

5.1.1 Stroboscope-like exposure

The condition to use this method requires that the mean dislocation of the object from a reference position (not necessarily the rest position) decays with time. A hologram can be taken by exposing the photografic plate only while the object is in the reference position. To do this the position of the object must be exactly determined and when in the reference position, the laser light should be switched on.

This type of exposure is in effect "stroboscope-like". For the experiments a cw-Ar-laser with an acousto-optic output coupler inside the resonator was used. The laser is equipped with highly reflecting mirrors, so that without a signal the energy is stored in the resonator. A standing acoustical wave in the modulator deflects the beam according to a control signal and in this way the stored energy of the resonator is discharged in a pulse.

For determining the reference position of the object an acceleration gauge with low mass was fixed to the object and the signal processed by a special newly developed electronic equipment. When the acceleration signal (or respectively the derived path signal) reaches a limit which is variable depending on the application of the system, the modulation signal and subsequently the laser is switched off. By the aid of this device the holografic plate is only exposured while the object is in a region of 0.1 around the reference position. This technique was demonstrated by taking a hologram from a ship propellor without using vibration isolation.

5.1.2 Coupling of the light wave fiels by a stiff connection between the object and the holografic plate.

It is easier to get a constant phase difference between the reference and the wave at the location of the holografic plate, when the object and the hologram are rigidly connected together. Additionally of the object is illuminated through the photo-plate, the illumination wave isboth the reference beam and the object beam after the reflection on the object.

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By this method the object-plate system is decoupled from movements of the light source.

First experiments were made with simple models. In this case, the object with the holografic plate, was put on a normal tripod standing directly on the floor. The system was disturbed by the normal vibrations. A steel sheet which was vibrating in natural modes was then holographed.

5.1.3 Coupling of the object and the reference beam by a phase mirror.

In another method one can use a reference mirror fixed to the object in order to diminish the phase difference between the object wave and reference wave caused by the movement of a non-vibration isolated object during the time of exposure / 6 /.

If the light source impinging on the object and the reference beam are the same, the phase differences related to vibration movements will theoretically disappear.

This was proved with two interesting examples; firstly, the deviation of the working table of an electron beam welding machine under tensile load was investigated. To do this, the mirror of the reference beam was put onto the table. Then the table was moved and the interference fringe system was regarded in real-time.

Furthermore, the deformation of a welded sample was measured in a Testatron tensile test machine. Although it was necessary to expose the holograms for about 10 s, the good quality of the holograms proved the efficiency of the method.

5.2 The deformation measurements were made with fracture mechanics specimen of the steel 22 NiMoCr 3 7. The sample was chosen in such a way as to allow a comparison of the experimental values with those of theoretical predictions from fracture mechanics. The dimensions of the specimen were 30 x 120 x 520 mm^2, and the dimension B = 30 mm corresponds to the thickness of the HDR.

The experiments should show, whether defects could be detected by holografic interferometry, and whether the danger of the defects could be quantified with the aid of fracture mechanics.

The measurements were made in specially adapted holografic equipment with a newly developed 100 to-tensile-testing machine.

5.2.1 Construction of a tensile-test system for holografic investigations.

The forces needed to simulate conditions prevailing in practice are in the order of 1 MN. Testing machines with such high loading were not available in IfaM at the time of the planning of these experiments.

Therefore a hydraulic testing system was designed and built in IfaM. It consists of three major components:

a) Four hydraulic cylinders with a maximum pressure force of 0.3 MN per cylinder.

b) Two support crosses for the loading transfer from the cylinders to the sample.

c) The hydraulic aggregate.

The whole system was tested up to a load of 0.9 MN, which corresponds to the fracture loading of the notched sample, with very good results.

5.2.2 Experimental evaluation of the deformation field.

For these measurements, the specimen (30 x 120 x 520 mm^3) was prepared with internal cracks of variable length from 8 to 80 mm. With each crack length the specimens were loaded in steps of 0.05 MN (= 14 N/mm^2 related to the undamaged cross section) up to a load of 0.5 MN = 140 N/mm^2. This load approximates to that on the material in the HDR at a test pressure of 100 bar. The holografic-interferometric measurements were made using the double exposure method.

6. Results

6.1 Investigations in taking holograms without vibration isolation.

In order to conduct measurements directly at the reactor, it was necessary to investigate and refine the methods of taking holograms without the need for vibration isolation.

The following methods gave good results with respect to the taking of holograms and the making of interferometrical measurements:

6.1.1 Stroboscope-like exposure

6.1.2 One-beam holography

6.1.3 Phase reference mirror method

If the objekt is moving around a test position, it seems to be possible with these methods to compensate nearly all vibrations. But in all cases the techniques require more refinement before practical application is possible.

Furthermore, the investigations as part of this research work have shown, that it should be possible to measure deformations on reactor components by the means of holografic interferometry. During measurements on the pressure vessel, the holografic equipment is stiffly connected to the object. In this case the relative movements between the pressure vessel and the measuring equipment are small.

6.2 Deformation measurements with a flawed tensile test specimen.

The evaluation of the double-exposured holograms showed that when the crack in the middel of

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the specimen reaches 10 mm (= 17 % of the cross section). In this case the deformation reaches 0.5 µm. With the largest experimentally investigated crack length of 80 mm, the measured results were compared to those from theoretical calculations. The measured deformation was approximately 30 % smaller than the calculated one. This result can be considered to be good in the light of the approximation necessary for the calculation.

6.3 Evaluation of the results.

The described investigations in the region of deformation measurements postulate a new application for holografic interferometry. Metallic materials were not previously investigated in this manner. The measurements of the internally flawed specimens have shown the physical influence at the unbroken surface as a result of the defects. The results lead to the conclusion that it should be possible, with improved methods of evaluating, to detect small cracks using this method. Apart from this positive aspect, it must be stated, that with the present state-of-the-art of holografic interferometry, it is not yet practical to detect critical defects by this means.

However other applications may be possible. The investigations have also shown, that is should be possible to quantify the stress intensity in the region of a crack when the methods of holografic interferometry are promoted in this direction. Since the stress intensity in front of the defect is a value to describe the static and dynamic strength of the material, the potential danger of a defect can be quantified without other expedients. The future research should be directed to an investigation of this field of application.

9. References

/ 1 / Kiemle, H., D. Röss — Einführung in die Technik der Holografie. Akademie Verlagsgesellschaft Frankfurt/Main 1969

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/ 3 / Corcoran, V.I., R.W. Herron, I.G. Jamarillo — Generation of a Hologram from a moving target. Appl. Opt. 5, 668 (1966)

/ 4 / Neumann, D.B. H.W. Rose — Improvement of Recorded Holographic Fringe by Feedback Control. Appl. Opt. 6, 1097 (1967)

/ 5 / Waters, I.P. — Object Motion Compensation by Speckle Reference Beam Holography. Appl. Opt. 11, 630 (1972)

/ 6 / Jüptner, W. — Berichte zum Forschungsvorhaben RS 102 - 22

/ 7 / Jüptner, W. — Untersuchungen zur zerstörungsfreien Prüfung von Reaktorkomponenten mit Hilfe der holografischen Interferometrie. Abschlußbericht zum Forschungsvorhaben.

Classification: 12.3	
Title 1 (Original Language): Ermittlung von Schweißeigenspannungen mit Hilfe der Röntgenografie im ambulanten Einsatz (RS 13o - II.3.2, Jahresbericht A76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Allianz, Ismaning
Title 2 (English): The Determination of Welding Residual Stresses with the Aid of Ambulant X-Ray Diffraction	Project Leader: Dr. Christian Dr. Elfinger
Initiated (Date): 1.6.1974 Status: Finished Completed (Date): 31.5.1976 Last Updating (Date): June 1976	

1. General aim

The only method known today of determining the absolute value of residual stresses in metallic materials by non-destructive means is the x-ray diffraction stress measuring procedure. The aim of the work in progress is to clarify to what extent it is possible to incorporate the measurable residual stresses in the surface or in the surface-adjacent zones of welded joints into theoretical reflections on strength factors, and to perfect measuring methods and apparatus in such manner as to facilitate their ambulant application in respect of reactor components.

2. Particular objectives

The work is divided into two sections:

a) Investigations into the evolution of surface stresses and stress gradients in and in the vicinity of submerged arc welded joints.

b) Reconstruction of the x-ray diffraction measuring equipment for ambulant use.

3. Experimental facilities

For the x-ray diffraction measuring on specimen welds, two centre-point

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free x-ray goniometers of Messrs. Siemens were available. For the ambulant measurements on major components, a newly developed goniometer was used.

4. Progress to date

Determination of the distribution of welding residual stresses laterally to the welded joint and the stress gradients dependent on the depth in submerged arc specimen welds has been completed. On the basis of model conceptions, endeavours were made to discuss the formation of welding residual stresses in the weld material and the heat-affected zone.

The new design and construction of a mobile goniometer has been completed and the first test measurements carried out (Fig. 1).

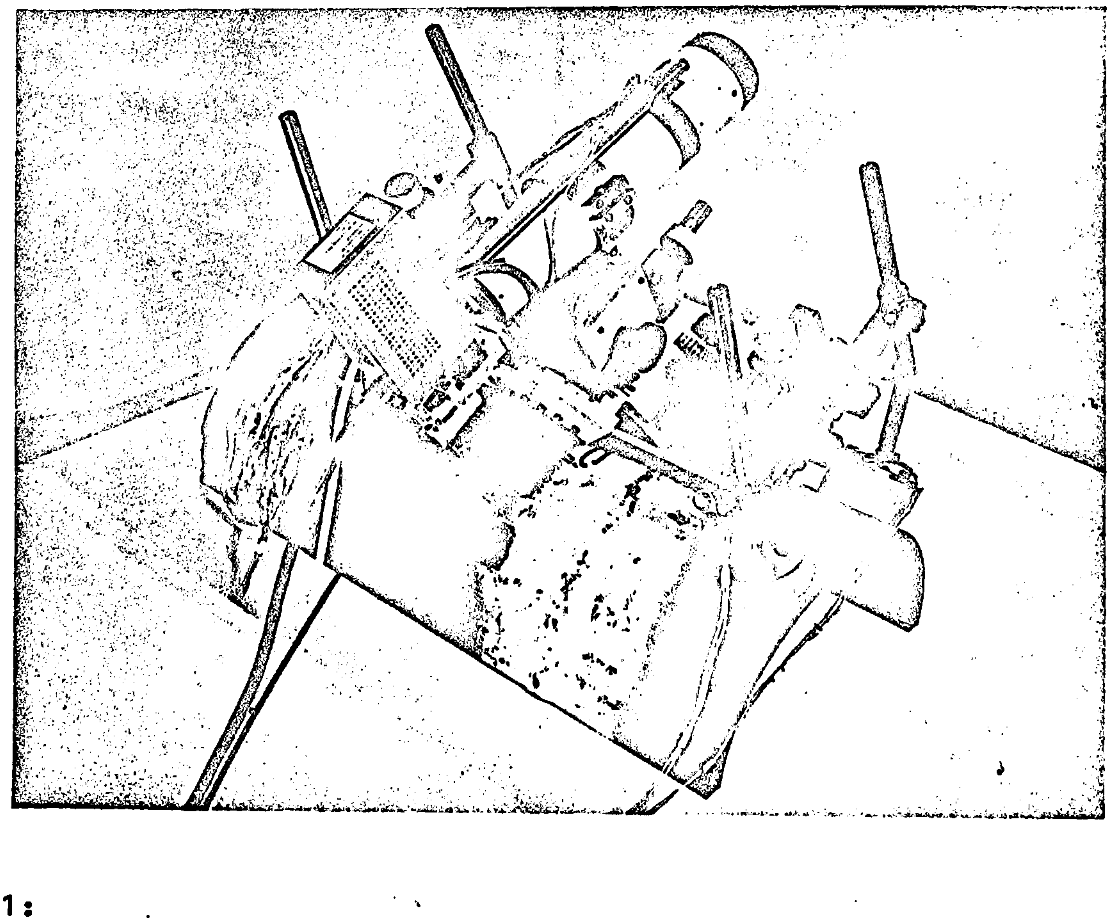

Fig. 1:

5. Results

The fundamental question of the work in hand, namely to what extent it is possible to employ x-ray diffraction measuring in determining residual stresses in weld seams, can be answered in the affirmative. In untreated weld seams, the maximum stress peaks occur in the weld seam surfaces, the stress gradient being dependent on the depth whilst in surface-adjacent zones it is almost nil or very low. Thus, in spite of the shallow penetration depth of the x-rays, material zones are reached, whose residual stress state ist characteristic for the weld seam to be measured. In mechanically treated weld seams, electrolytical removal of a maximum of o,3 mm suffices to also penetrate into zones, which are unaffected by the treatment and reveal the characteristic residual stress state of the original weld seam.

The newly developed x-ray goniometer for mobile use shown in Fig. 1 permits easy application of the x-ray welding residual stress measuring procedure on components of any geometry down to a curvature radius of 15o mm. By virtue of the rigid connection between component and measuring equipment, it is possible for the first time to quantitatively employ the x-ray measuring method on major objects.

The test measurements conducted also yielded new data and information on the formation of welding residual stresses in weld material and heat-affected zones.

For the newly developed goniometer, patent (No. 2633144.6) and registered design (No. 762344.5) have been applied for at the German Office of Patents in Munich.

The work is completed and concluded.

6. References

Christian H., Elfinger F., Klug W.: Ein Röntgengoniometer für den mobilen Einsatz. Materialprüfung 18 (1976) Nr. 1o, S. 388 - 39o

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Classification: 12.3	
Title 1 (Original Language): Holographisch-interferometrische Untersuchungen zur Material- und Bauteilprüfung von Reaktorkomponenten (RS 148 - II.3.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Dornier System
Title 2 (english): Holographic-Interferometric Investigations in the Field of Testing Pressure Vessel Materials and Components	Project Leader: Dr. Grünewald
Initiated (Date): 1.10.1974 Status: continuing	Completed (Date): 31.1.1976 Last Updating (Date): December 1975

1. General aim

The aim of the investigations is to establish whether and how the holographic interferometry can be applied to material testing in the field of nuclear safety.

2. Particular objectives

Theoretical and experimental investigations to clarify the possibility of finding faults in pressure vessels with thick walls.

3.1 Experimental facilities

Holographic test facility with a 4 W-argon-laser

3.2 Research program

Theoretical investigations:

- calculation of the deformation of pressure vessels under load and of the influence of local irregularities of different shape and position on the deformations
- calculation of interferograms of pressure vessels with irregularities under pressure load.

Experimental investigations:

- qualitative and quantitative holographic deformation measurements of steel tubes with faults under pressure load (tubes of 50, 100, 203, 419 mm diameter and a wall thickness

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of 1.6, 2.2, 6.3 and 10.6 mm; the extension of the faults amounts from 6 % to 30 % of the wall thickness), flange-lids and welded steelpieces under thermal load, and plastic deformation measurements around the top of a crack under stress.

4. Project status

4.1 Progress to date

The investigations listed under 3.2 have been carried out. Within these investigations different modes of load have been proved and the sensitivity of the system has been increased by means of varying the optical arrangement and testing procedure.

4.2 Essential results

As one essential result it can be stated that cracks in steel tubes with a diameter-to-wall-thickness relation of about 40 can be detected even when the size of the crack amounts to 6 % of the wall thickness and the crack is located at the inner side of the wall while during testing the outer side of the wall is inspected.
The sensitivity was obtained in an advanced testing mode. By optical means the normal interference pattern was suppressed and the high sensitivity for the detection of the irregularities was obtained.

The test pieces with cracks under a stress load led to interferograms with an irregular fringe-shape in the range of plastic deformation near the cracks. The calculation of deformations and strains from the interferograms are not yet completed.

5. Next steps

Evaluation and discussion of the results

6. Relation with other projects

The investigations were carried out in connection to the projects RS 132, executed by the TU Hannover, and RS 102-22, performed by the Fraunhofer-Gesellschaft, IFaM, Bremen.

Classification: 12.3	
Title 1 (Original Language): Entwicklung von zerstörungsfreien Prüfverfahren für Wiederholungsprüfungen an Reaktordruckbehältern (RS 2702 - II.4.2., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: MAN Nürnberg
Title 2 (english): Development of Non-Destructive Inspection Techniques for In-Service Inspection Tests of Reactor Pressure Vessels	Project Leader: J. Lindner
nitiated (Date): 1.10.1972 Status: finished Completed (Date): December 1975 Last Updating (Date): December 1975	

1. General aim

The general aim of this task is the development of inspection equipment with the aid of which in-service inspection tests can be carried out on pressurized components of nuclear reactors in the scope considered necessary by the authorities, such inspection tests to yield qualified test results without subjecting the inspection personnel to excessive irradiation. The inspection tests under this research programme are limited to ultrasonic techniques. In addition to the Federal Institute for Materials Testing (BAM) the task is being handled by the firms of Kraftwerk Union AG (KWU), Krautkrämer GmbH (KK) and Maschinenfabrik Augsburg-Nürnberg AG (MAN).

2. Particular objectives

The task is divided into the following objectives:

1. Basic investigation into means of inspection by ultrasonic techniques and the influence of boundary conditions; inspection studies on primary circuit systems.

2. Development of inspection systems for the various inspection areas.

3. Development, manufacture and testing of manipulating equipment for internal and external tests.

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4. Onward development of components of electronic systems and data logging systems for ultrasonic inspection work.

5. Testing of inspection systems.

3. Experimental facilities and programmes

Experimental facilities were set up to simulate internal and external test conditions for the purpose of testing the inspection system embracing the probes, the manipulator, the ultrasonic electronic gear and the data logging system. To this end, a large pressure vessel wall specimen with artificial flaws was devised and subjected to a voluminous inspection programme to establish the influence of various parameters on the detectability of flaws. In addition, there are experimental programmes to examine the influence of the cladding, the inclination of separations, the crack structure and the crack position in the material on flaw detectability.

4. Project status

4.1 Progress to date

A large number of tests were made to establish the influence of the cladding on the probability of fault detection and on fault indication behaviour. In addition, the influence of other parameters was examined, such as the angle and surface structure of planar reflectors and the influence of surface conditions of the test specimen. These investigations were made theoretically and were supplemented by experiments.

Ultrasonic probe systems were improved for better interpretation of the inspection results and higher probability of fault detection.

Ultrasonic electronics were matched to the increased demands on the probe systems. Appropriate computer programmes were set up to evaluate the data stored on tape.

Manipulators for internal and external inspection and for inspection of the spherical closure and closure head of the reactor pressure vessel were advanced to a level where most of the pressure vessel can be tested by remote control.

The whole system was tested by employing the equipment set forth in 3) and improved on the basis of experience gained.

4.2 Essential results

All equipment and processes have been improved to a stage where almost all essential areas of the reactor pressure vessel can be examined with a high probability of fault detection. Limitations need only be imposed in certain areas regarding the interpretation of inspection results.

5. Next steps

Work under this programme will terminate on 31st December 1975. As individual problems have not yet been solved completely, all parties are making efforts to obtain a continuation order covering an extension of the inspection zones of the reactor pressure vessel as well as an improvement in the interpretation of the inspection results. In addition, the parties want to set up a working programme for inspection of the primary circuit.

6. Relation with other projects

Reference is made to what has been said in Report A 74.

7. Reference documents

The participants of the present project issued the following comprehensive reports in German during the period under review:

Author	Title	Date
M.A.N.	Investigations into the inspection ability of inner nozzle radii during external inspection of the reactor pressure vessel	March 1975
KK	Noise level due to grain structure and size at spherical closure of a reactor pressure vessel	April 1975

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Author	Title	Date
KK	Ultrasonic inspection of perforated spherical bottom closure of light water reactor pressure vessels	April 1975
BAM	Manual ultrasonic single-probe echo dynamics on artificial reflectors in spherical closure of a reactor pressure vessel	May 1975
M.A.N.	Study of a telescopic mast for the central mast manipulator	May 1975
BAM	Tandem echo dynamics	October 1975
KK	Study of reflection behaviour of various inclin ed notches in connection with the inspection of the spherical closure of light water reactor pressure vessels.	October 1975

8. Degree of availability

A relatively small number of reports were issued. A limited number are still obtainable from the issuer or IRS.

Classification: 12.3

Title 1 (Original Language): Entwicklung und Bau einer Ultraschall-Prüf- und Auswerteelektronik einschließlich Vorverstärkerkasten RS 169 - II.3.2, Jahresbericht A 76	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Krautkrämer GmbH
Title 2 (English): Development and construction of an electronic test and evaluation system with ultrasound including pre-amplifier box	Project Leader: Ing.grad. G. Gutmann
Initiated (Date): July 1, 1976 Status: Continuing Completed (Date): December 1977 Last Updating (Date): December 1976	

1. General Aim

Development of ultrasonic evaluation electronics meeting today's technical safeguarding requirements, universally applicable with all types of reactors.
Due to the experiences made so far and to the technical development as it is foreseeable in the future, the following demands were made on this electronic evaluation system:

2. Particular Objectives

Maximum number of channels: 60
Maximum speed of the manipulating system to be reached: 100 mm/s
Scanning rate: 1 test per mm
And, the installation is to be controlled continuously for function and stability.

3. Research Program

- Drawing up of the detailed operating program with arrow diagram.
- Development of some new evaluation modules
  a) min/max-value-memory
  b) noise investigations
- Development and construction of the electronic evaluation system corresponding the demands made.
- Testing the interplay between the modules developped.
- Projection works with regard to the construction of the prototype.
- Construction of the prototype.
- Investigations for noise suppression at the prototype of multiple wire systems.

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4. Progress to Date

The works of the development program listed under 3. have all been accomplished but the last two points

- Construction of the prototype
- Investigations for noise suppression at the prototype of multiple wire systems.

The works on these two points had to be deferred because of details demanding clarification before starting on the definite construction of the electronic system, such as number of channels and possibly integration of modules and/or drafts of the research plan RS 102-17.

5. Results

All electronical modules, especially needed for this electronic evaluation system have bee developped and tested as function sample. The demands to be met on function, exactitude, and stability of the modules have been fulfilled.

While carrying on construction and development of the general aim the following principal requirements were satisfied:

a) Maximum number of channels: 60
b) Maximum speed the manipulating system can realize: 100 mm/s
c) Scanning rate: 1 test per mm
d) All-automatic and permanently effective control of function and stability.
e) Maximum stability tolerance permissible of ± 1.5 dB.

The aims set in respect of function and stability have been reached.

Production can be started following clarification of the definite extension steps on the prototype.

6. Next Steps

- Synchronisation with research plan RS 102-17 and start of the layo works for the construction of the prototype.
- Construction of the prototype
- Investigations for noise suppression at the prototype of multiple wire systems.

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Classification: 12.3	
Title 1 (Original Language): Schallemissionsmessungen an bruchmechanischen Proben (Ermüdungsrissen) (RS 191-II.3.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Battelle-Institut e.V.
Title 2 (English): Acoustic Emission Measurements on Fracture Toughness Specimens (Fatigue Cracks)	Project Leader: Dr. J. Eisenblätter, H. Jöst
Initiated (Date): January 1, 1976 Status: Continuing Completed (Date): March 31, 1977 Last Updating (Date): December 1976	

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 are conducted on one type of defect (fatigue crack). The examinations include the materials of the pressure vessel of light-water reactors, i.e. the ferritic steel 22NiMoCr37 and the cladding material austenite X10CrNiNb189 as well as materials of the sodium-cooled fast breeder reactor, i.e. the austenites X6CrNi1811 and X6CrNiMo1713, the ferritic steels 8CrMoNiNb910 and X12CrMo91, and "Incoloy 800" (X10NiCrAlTi3220).

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 is measured.

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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 optimization of the required correlations between measured and desired values
3.5 Investigation of the acoustic wave propagation in thick-walled structural parts

4. Experimental Facilities

Ad 3.2: Construction of a measuring facility for calibrating narrow- and broad-band AE transducers by means of the reciprocity method with two transducers, which permits absolute calibration (voltage/sound pressure).

Assembly of the measuring system for AE data such as count rate, total count and amplitude and energy distribution.

Assembly or construction of experimental facilities for detecting and monitoring crack propagation during fracture toughness testing by optical measurement of the crack growth, measurement of the crack opening displacement (COD) and examination of the fracture surfaces.

Ad 3.3: A servo-hydraulically controlled test unit (load capacity 220 Mp) which has been made suitable for AE measurements by special interference noise eliminating measures is available for the experiments.

Ad 3.5: Assembly of a measuring facility for determining wave propagation by means of conventional instruments such as pulse generator, amplifier, oscillograph and multi-channel source location unit. A steel plate with polished surface and with the dimensions 1800 x 900 x 80 mm has been prepared; in addition, a large vessel is available for the experiments.

5. Progress to Date

Ad 3.1: Preparation of modified compact tension specimens from 22NiMoCr37 and heat treatment of these specimens in order to obtain the required microstructures: 100 percent low temperature bainite; 40 percent bainite, balance ferrite and pearlite (grain size ASTM 7-8), 40 percent bainite, balance ferrite and pearlite (ASTM 2-3).

The hardness values determined and first investigations on the heat-treated specimens showed that the materials did not meet the usual requirements with respect to mechanical parameters. For this reason some specimens with the structure of 100 percent low temperature bainite were subsequently subjected to an additional heat treatment in order to achieve higher toughness values.

Design and construction of a special chucking device without pin joints (Fig. 1) to avoid the friction noise caused by the pins of usual chucking devices for compact tension (CT) specimens.

Ad 3.2: Evaluation of methods for the exact calibration of AE transducers, using two different procedures. A broad-band transducer made of lithium sulfate hydrate was calibrated in an under-water sound-field which was measured by a calibrated hydrophone and by the reciprocity method (cf. 4.). Routine calibration of a large number of narrow- and broad-band transducers consisted in comparing these transducers with the above-mentioned broad-band one.

During the fracture toughness experiments the AE signals were measured by narrow- and broad-band transducers and partly stored on magnetic tape. The AE count rate was directly recorded with coarse amplitude distribution analysis, while the magnetic tapes were evaluated subsequently. Some experiments also included source location measurements, measurements of the amplitude distribution and frequency spectra,

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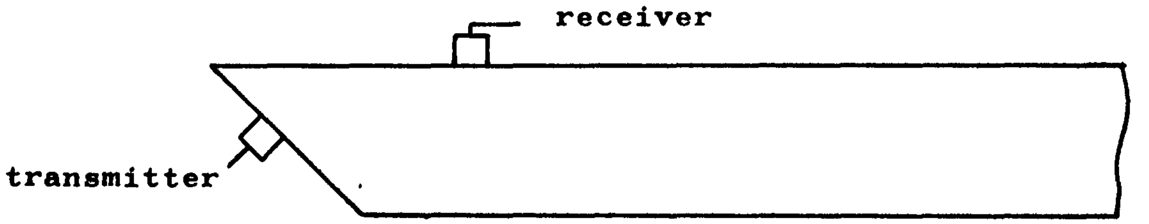
both under cyclic and tensile loading.

During each experiment the crack length was determined with the methods outlined in Section 4.

Ad 3.3: The load and the load frequency were varied during the experiments in order to be able to study the resulting effects on AE. A total of 17 fracture toughness specimens (modified 2" CT specimens) made from steel 22NiMoCr37 of different microstructure were investigated during cyclic and tensile loading, using two temperatures (RT and 60°C).

Ad 3.5: Transmission of confined transversal and longitudinal wave pulses of the frequency of 1, 2 and 4 MHz from an oblique end surface of the test plate (cf. drawing below).



Recording of body and surface waves on the surface.

Investigation of the following special cases:
- near field of ultrasonic probes (distance-dependent phase velocity)
- transversal waves from ultrasonic probe with angles of incidence in the region of the critical angle of total reflection of 33.2° (strong angle-dependence of the phase shift and of the shift of the reflected wave beam)
- transversal waves from normal point sources in the angular range between 34° and 37° (strong angle-dependence of the phase of the incident wave)

6. Results

Ad 3.2: When the elastic properties of the transmission medium (steel) are known, the reciprocity method enables absolute calibration of the transducer sensitivity(in dB resp. 1 V/µbar).

Figs. 2 and 3 show the sensitivity values of various commercial broad-band transducers (Fig. 2) and of several Battelle-developed and commercial narrow-band transducers (Fig. 3). Comparing the results achieved by calibrating broad-band and narrow-band transducers of various manufacturers shows that

- all these transducers are considerably less sensitive than stated by the manufacturers,
- the sensitivity decreases with increasing frequencies also in the case of broad-band transducers
- the resonant transducers designed by Battelle-Frankfurt, which have a natural frequency of 70 and 140 kHz, are by far the most sensitive ones.

It should be noted, however, that this calibration relates only to longitudinal waves.

Ad 3.3: Fig. 4 shows the fracture surfaces of the materials at different quenched and tempered conditions; the differences in toughness are clearly obvious. The fracture surfaces in Figs. 4a and 4b (40 % low-temperature bainite, balance ferrite and pearlite, ASTM 2-3 or 7-8) and Fig. 4c (100 % low-temperature bainite) are characteristic of brittle material behavior, while Fig. 4d shows the fracture surface of a tough material (additional heat treatment). The great difference in toughness values is also evident from the diagram in Fig. 5 which shows the load versus the crack opening displacement (COD).

Figs. 6 and 7 depict the AE count rate recorded at four respectively two different discriminator thresholds and the COD during rupture of a brittle specimen (corresponding to Fig. 5, curve a) and a tough specimen (corresponding to Fig. 5, curve b). Two regions in which AE occurs can be clearly distinguished. These two regions are separated at the upper load P_o during cyclic loading. AE in the lower load region, which has its maximum at about the medium load P_m, is definitely caused by friction between the crack surfaces. The increase in AE above P_o, which increases until fracture occurs, is attributed to micro-cracking processes at the crack

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tip. The amplitudes of these AE signals are considerably higher than those of AE originating from friction processes.

Fig. 8 shows a distinct increase in the AE count rate with decreasing load frequency during cyclic loading, which clearly demonstrates that the AE below P_o results mainly from friction (slip-stick effect). The load frequency was varied between 0.0016 and 1.66 Hz in the experiment.

Ad 3.5: The amplitude of the surface waves is about 50 dB lower than that of the incident body waves. Also in the special cases mentioned in section 5 intensive surface waves were not observed.

7. Next Steps

Ad 3.1: Procurement of the materials used for the sodium-cooled fast breeder reactor and preparation of specimens; construction of chucking devices

Ad 3.2: Calibration of transducers with shear and surface wave

Ad 3.3: Performance of the remainder of the planned experiments

Ad 3.4: Evaluation of the results with a view to establishing correlations between measured AE values and characteristic crack parameters

Ad 3.5: Additional investigations on the test plate, the waves being transmitted from a point source (as a function of the distance between transmitter and receiver), location of AE sources; investigations on a vessel

8. Relation to Other Projects

Continuation of the work under projects RS 0031 to RS 0031 D; cooperation with IzfP, Saarbrücken, and IfaM, Bremen (RS 196)

9. References

10. Degree of Availability of the Reports

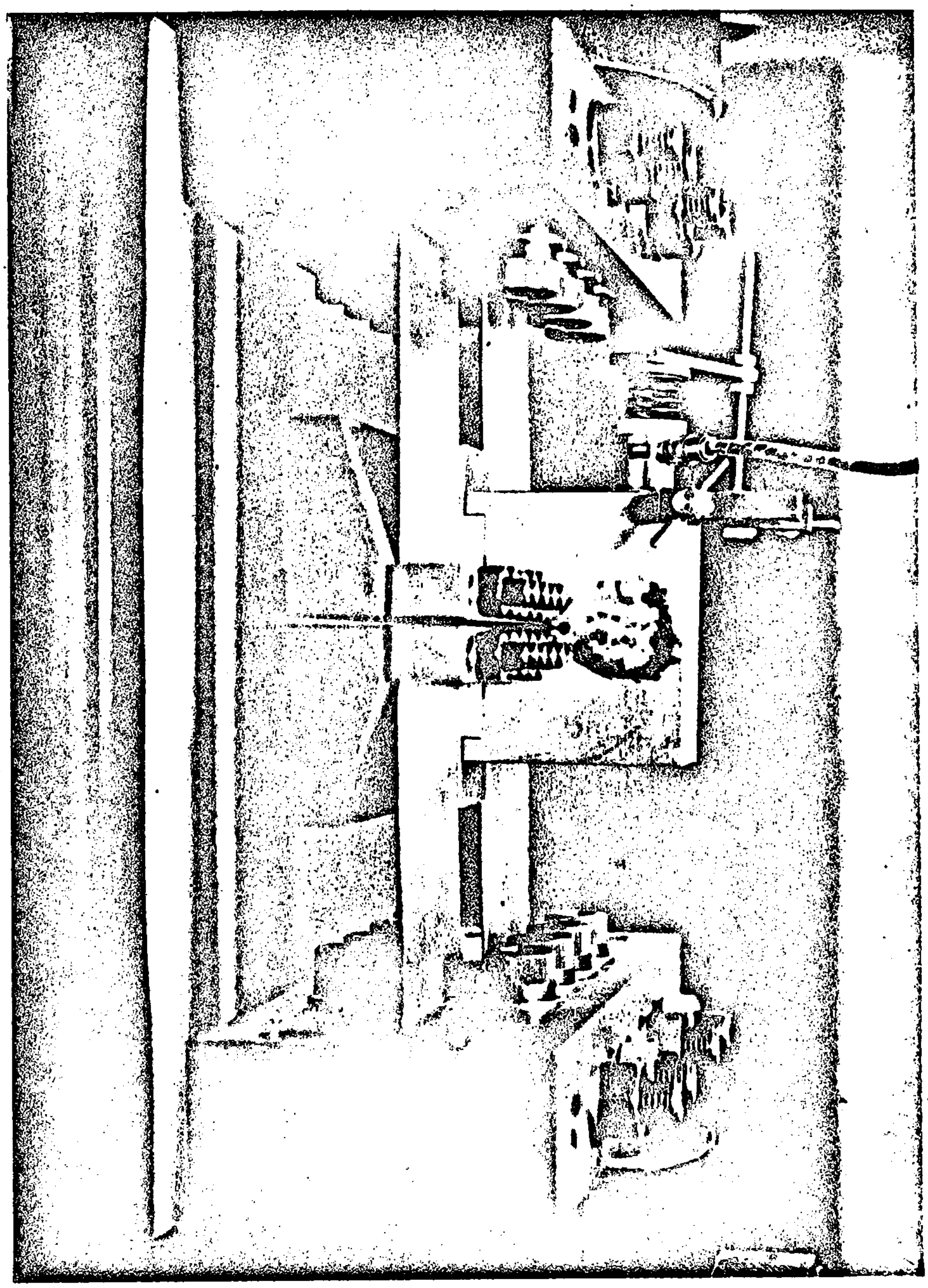

Fig. 1: Chucking device and specimen

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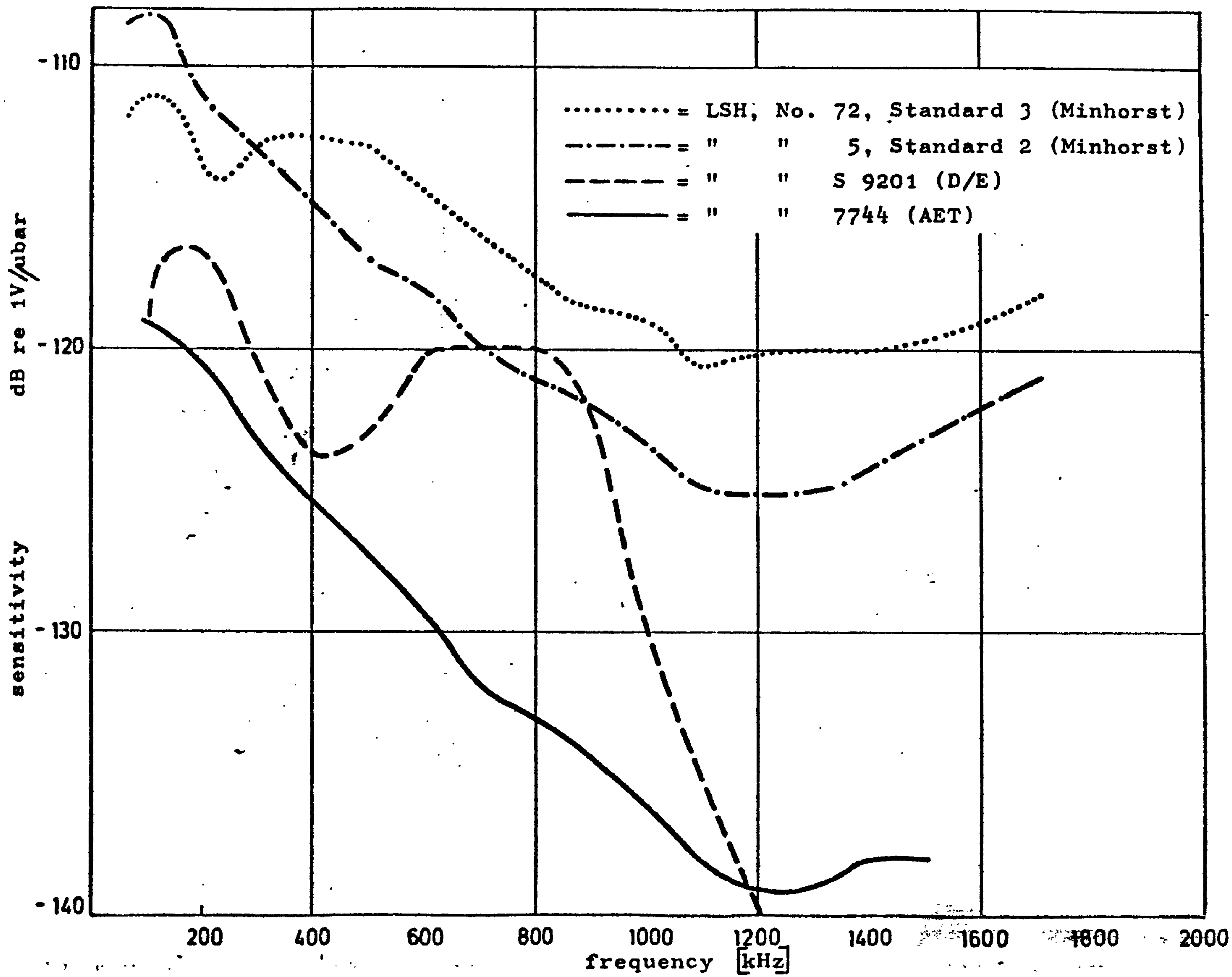

Fig. 2: Reciprocity calibration of broad-band transducers

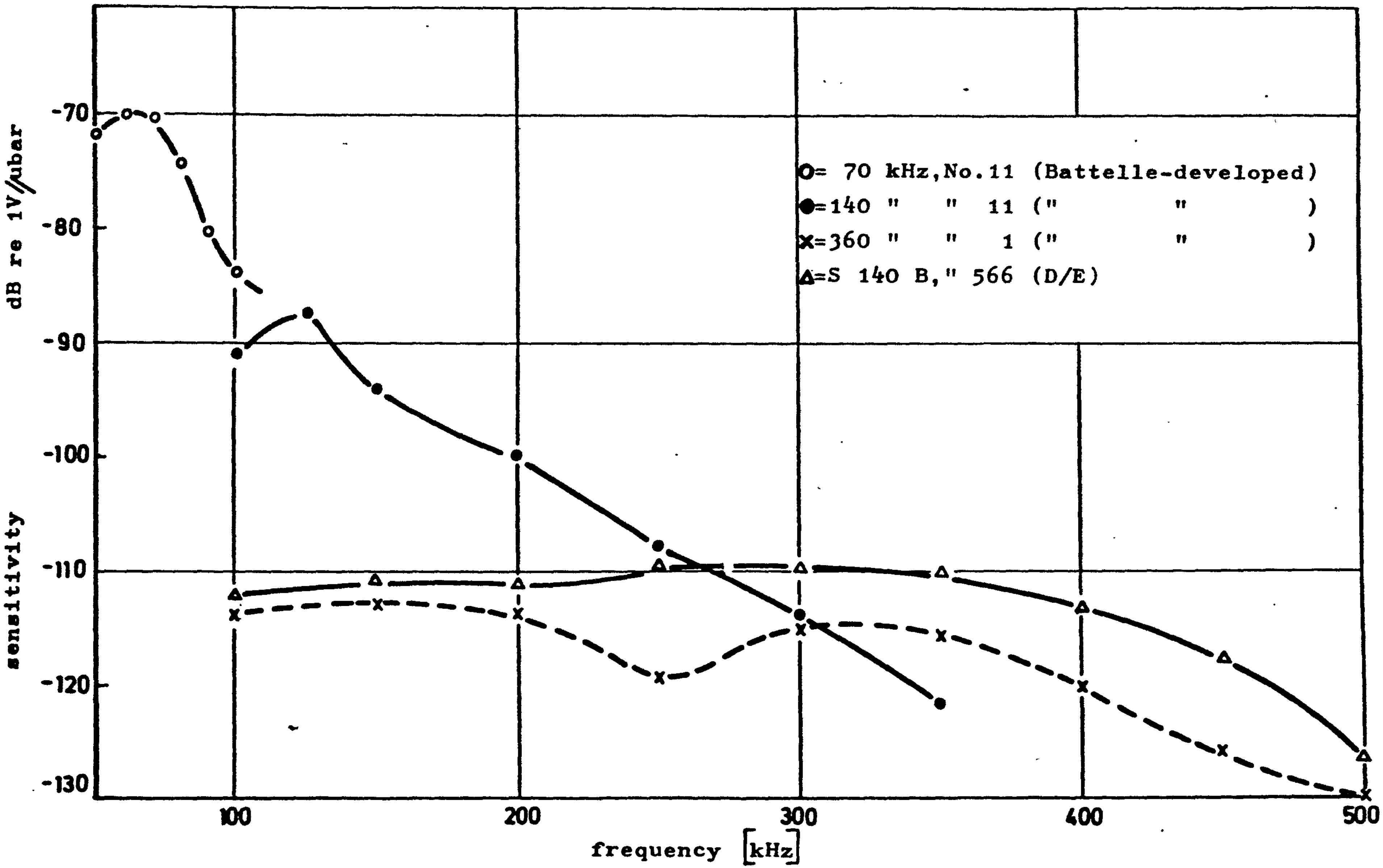

Fig. 3: Reciprocity calibration of narrow-band transducers

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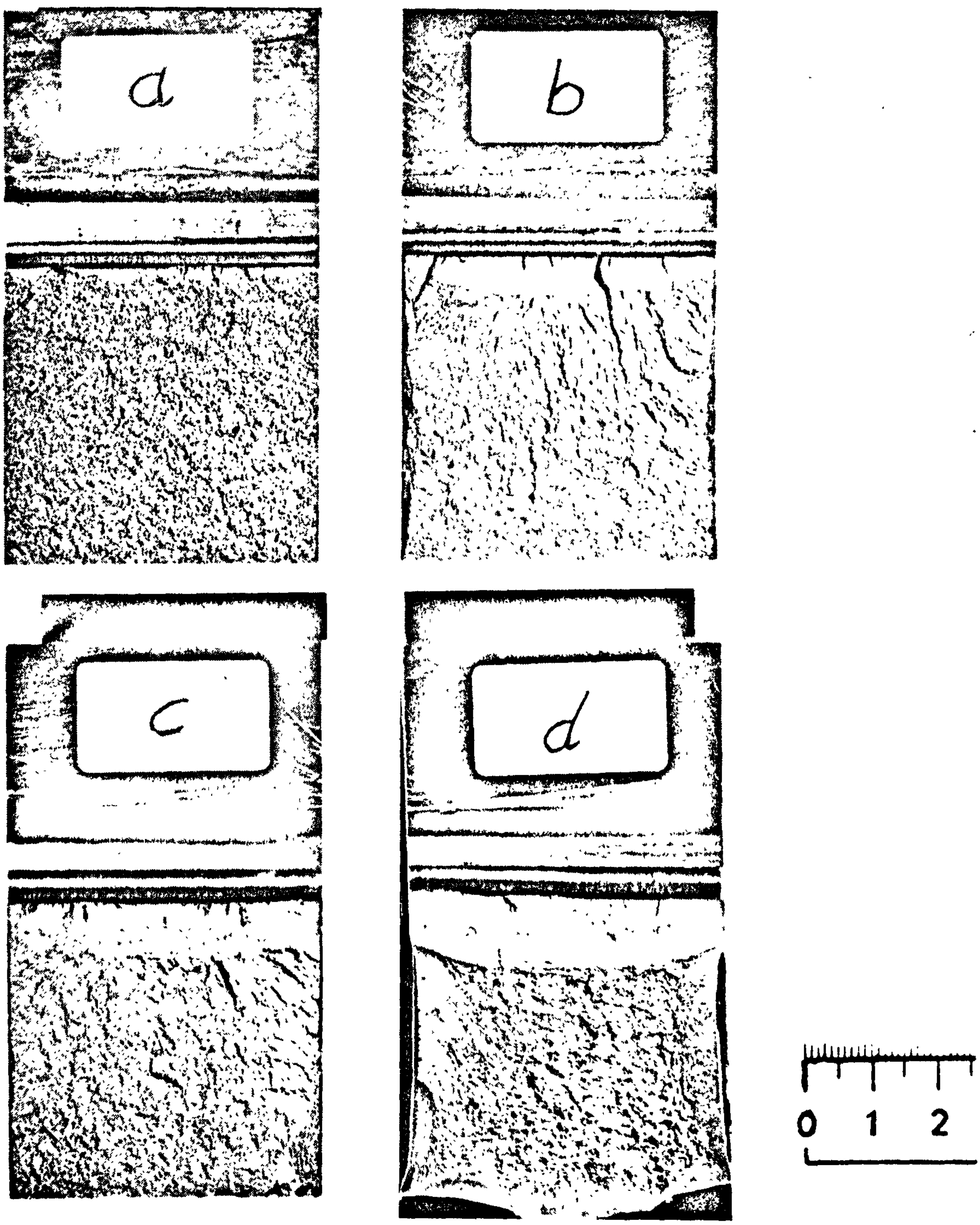

Fig. 4: Fracture surfaces of specimens from 22NiMoCr37 after different heat treatments

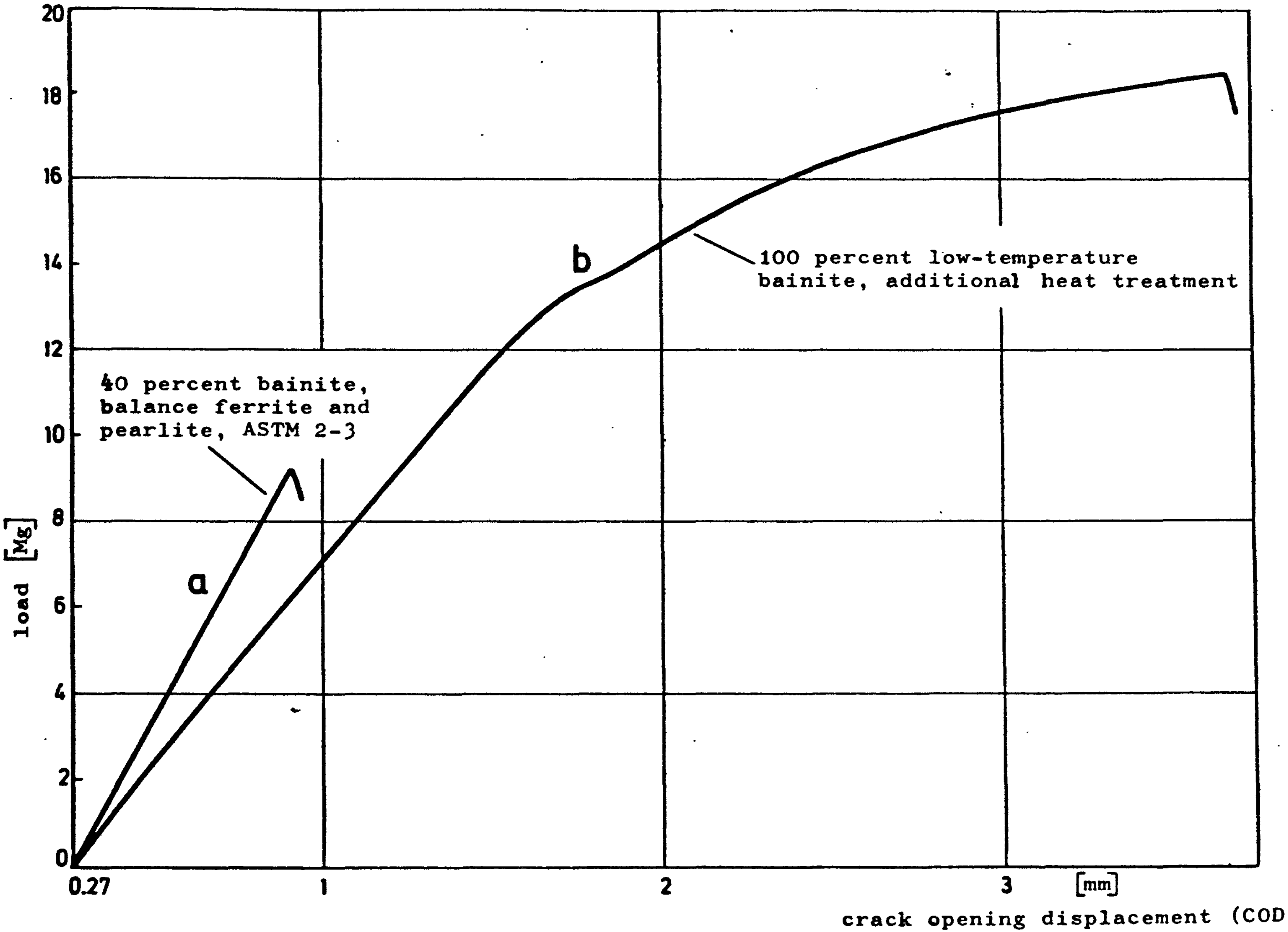

Fig. 5: Load-COD diagrams

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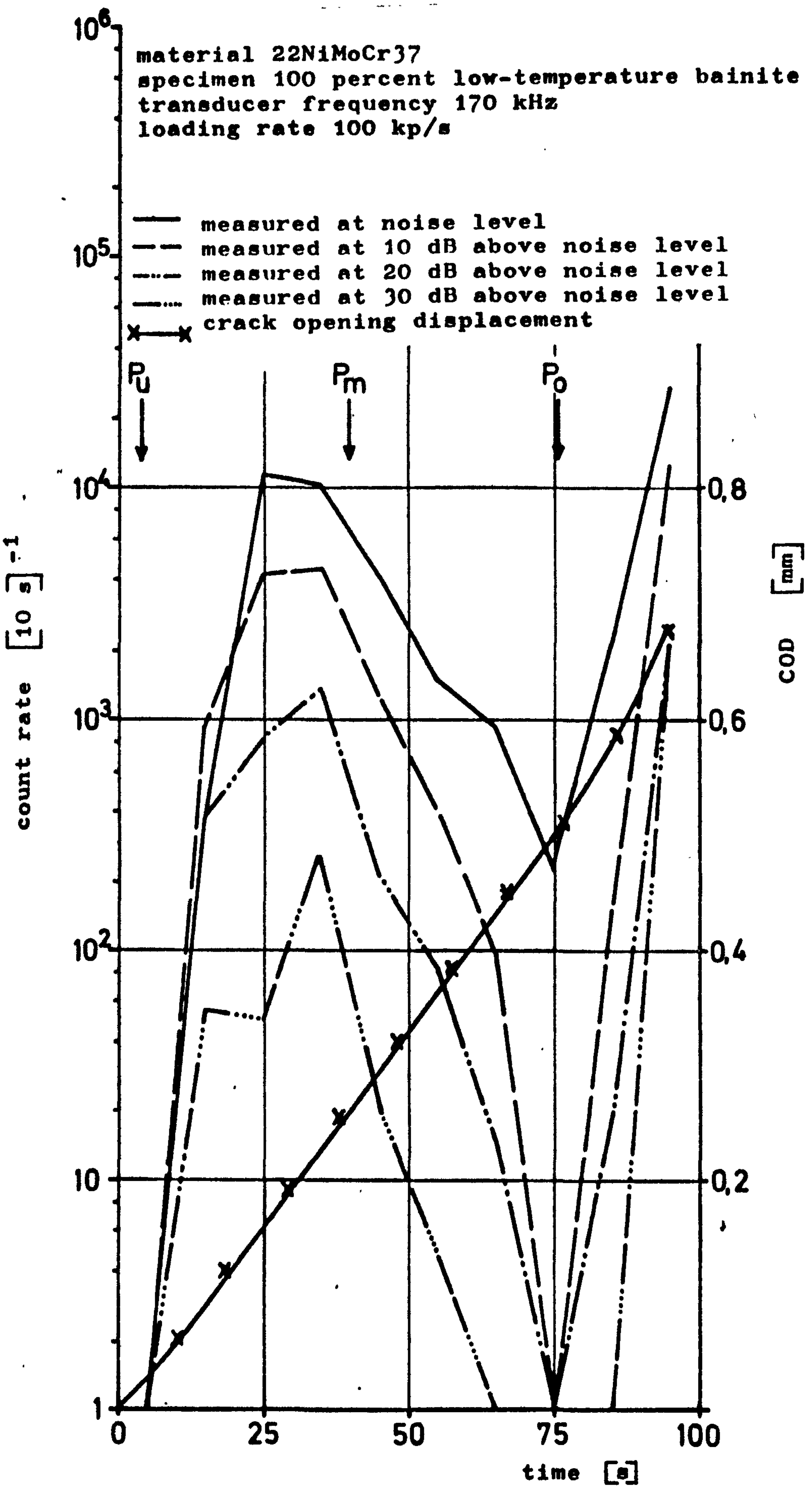

Fig. 6: AE and COD over time during tensile test of a pre-cracked specimen

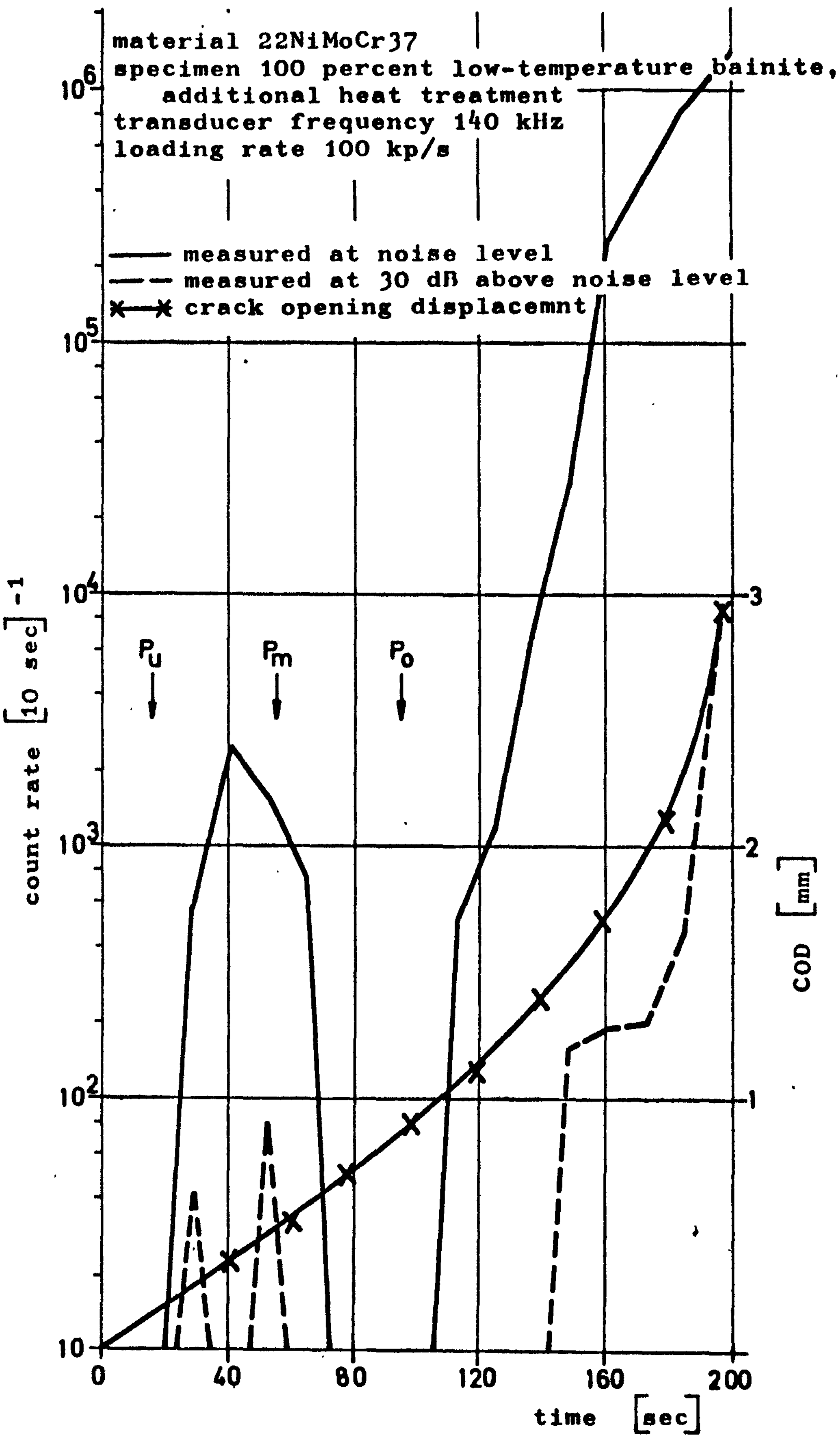

Fig. 7: AE and COD over time during tensile test of a pre-cracked specimen

1500.

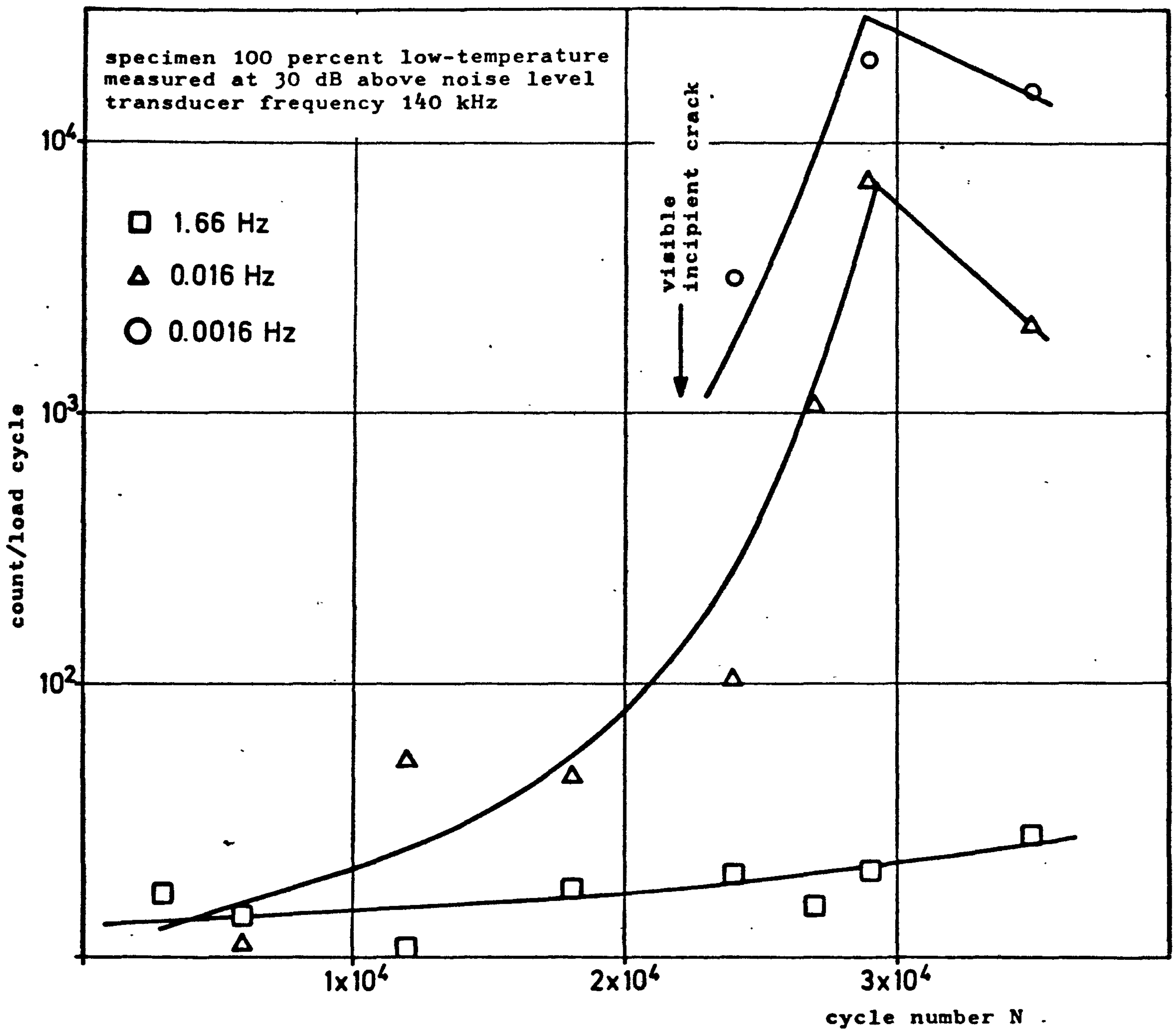

Fig. 8: AE at varying load frequency during fatigue

1501.

Classification: 12.3	
Title 1 (Original Language): Vorversuche zur Leckageüberwachung mit Hilfe der Schallemissionsanalyse (RS 193-II.3.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Battelle-Institut e.V.
Title 2 (English): Preliminary Investigations On On-Line Leak Detection By Acoustic Emission	Project Leader: Dr. P. Jax
Initiated (Date): January 1, 1976 Status: finished Completed (Date): January 31, 1977 Last Updating (Date): December 1976	

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 occuring 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 to be produced on a test loop of KWU, Erlangen, and the acoustic emission involved was to be analyzed.

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The research program covered the following investigations:

- 4 different leak types (crack-type leak, flange-type leak, valve leak, and borehole-type leak); pressure about 150 bar, temperature 300°C

- 2 leak sizes for the crack-type leak and the valve leak (about 35 and 180 kg/h) and systematic variation of the leak size for the flange-type and borehole-type leak; pressure about 150 bar, temperature 300°C

- Variation of several parameters affecting the background noise, such as the flow rate in the measuring section and the delivery of the circulating pump

- Systematic variation of temperature (320 to 90°C) or pressure (160 to 86 bar) for two leak types (borehole-type leak and flange-type leak) or one leak type (borehole-type leak)

On the basis of AE signals stored on magnetic tape, detailed analyses were to be made with respect to

- acoustic emission intensity and amplitude distribution,

- frequency spectrum,

- correlation between the various parameters.

4. Experimental Facilities

The test loop used was that of the experimental refuelling-machine facility made available by Kraftwerk Union AG (KWU), Erlangen. In this facility pressure and temperature can be controlled separately. The loop was connected with an austenitic tube (length about 10 m, outside diameter 267 mm) which constituted the actual measuring section. The devices for simulating the different leak types were mounted to this tube by means of nozzles and flanged connections.

5. Progress to Date

For the leakage experiments, transducers were mounted either directly or via flexible waveguides at five measuring points differently spaced from the leak. The signal level, represented by the rms value, was continuously recorded. For detailed analysis, the AE signals from all measuring points were stored on magnetic tape in analog form.

The research program outlined in Section 3 was carried out, and the planned analyses of the signals stored on magnetic tape have been largely completed.

Parallel to the leakage experiments, the attenuation of acoustic signals in the measuring section was determined at different frequencies. These measurements were made both with continuous noise sources (compressed air-stream, sinus signal fed into a transmitter) and with pulsive signals (square pulse fed into a transmitter).

6. Results

Acoustic emission was found to be a sensitive method for the detection of leaks. All leaks with leakage rates from about 20 kg/h to 250 kg/h were indicated in spite of the background noise of the experimental facility. As can be extrapolated from the results, the detection limit of AE, $N \approx 2$ to 6 kg/h, is markedly below the leakage rates examined. For borehole-type leak and flange-type leak a simple relationship was found to exist between the low-frequency (60 to 150 kHz) AE intensity R and the leakage rate L determined by means of an orifice-plate circuit ($R \sim L^b$, b = 1.5 or 1). Within a specific scattering range, it is thus possible already at the present state of research to make inferences as to the size of a leak.

Amplitude Distribution of the AE Signals

In all cases a continuous acoustic signal level is emitted, which differs basically from the burst signals observed in the case of crack growth and friction processes. At a given frequency, the sound pressure amplitude A varies around an rms value U according to a uniform statistical distribution function ρ. Thus, we have

$$\rho = 2/U \cdot \left(\frac{A}{U}\right) \cdot \exp\left(-\left(\frac{A}{U}\right)^2\right) \qquad (1)$$

The physically reasonable measurement parameter is the rms value of amplitude U that can be measured by means of a voltmeter and from which, at a given frequency, all other parameters of acoustic emission intensity can be derived by means of equation (1). In general, it is sufficient for characterizing the leakage noise to measure U as a function of frequency and time.

Frequency Spectrum

Analyses of the frequency spectra of acoustic emission confirmed that marked changes in the ratio of the rms values of the investigated frequency ranges 300 to 900 kHz and 60 to 150 kHz may occur, depending on leak type, leak size and temperature. In a first approximation 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 (decreasing leakage rate). A similar relationship of the frequency spectrum with the leakage rate was not detected for the flange-type leak. There is evidence that the frequency spectrum depends in particular on the temperature of the medium circulated in the test loop. In addition, it was found that a minor variation of the frequency spectrum at the beginning of a leakage experiment correlates with a small increase in temperature at the site of leakage (heating by the medium).

In summary it may be stated that, apart from the results obtained for the borehole-type leak, it is not possible as yet to give more details about the relationships between frequency spectrum and the experimental parameters.

A comparison of the frequency spectra recorded at various measuring points showed a systematic relationship to the distance of the transducer from the site of leakage. With increasing distance the higher-frequency components (400 to 900 kHz) of acoustic emission are more strongly attenuated, i.e. there is a marked shift of the measured frequency spectrum in favor of the lower-frequency components. It appears feasible to achieve a more exact location of leaks with the aid of this effect, which is due to the attenuation of the signals during their propagation.

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7. Next Steps

Preparation of the final report.

8. Relation with Other Projects

Parallel to the AE measurements in the ultrasonic frequency range (50 to 800 kHz) described in the present report, the Allianz-Zentrum für Technik, München, measured the frequency components of leakage noise below 30 kHz (research project RS 97).

It was to be clarified in this way whether the audible frequency range or the ultrasonic frequency range is better suited for leak detection.

9. References

10. Degree of Availability

1507

<table>
<tr><td colspan="3">Classification: 12.3</td></tr>
<tr><td colspan="2" rowspan="3">Title 1 (Original Language):

Verbesserte phänomenologische Beschreibung von Schallemissionssignalen und ihre Analyse auf eine bessere Fehlerbewertung.
(RS 196 - II.3.2, Jahresbericht A 76)</td><td>COUNTRY:
BRD</td></tr>
<tr><td>SPONSOR: BMFT</td></tr>
<tr><td>ORGANIZATION:
Fraunhofer-Ges.,
IzfP, IfaM, IFKM</td></tr>
<tr><td colspan="2">Title 2 (English):
Better phenomenological description of acoustic emission signals and their analysis with respect to a better evaluation of defects.</td><td rowspan="2">Project Leader:

Dr. J. Lottermoser

Dipl.-Phys.
T. Fischer

Dr. B. Voss</td></tr>
<tr><td>Initiated (Date):
1.2.1976
Status:

continuing</td><td>Completed (Date):
31.1.1979
Last Updating (Date):

December 1976</td></tr>
</table>

1. General Aim

The general aim of this project is

- the determination of the correlations between acoustic emission (AE) parameters and defects (stress intensity factor) and optimization of mathematical models for the description of these correlations,

- characterization of AE sources (cracks) in base metals and welded joints,

- determination of the wave modes which are responsible for the propagation of AE signals in thick-walled components,

- correction of AE parameters for signal disturbances on their path in a specimen or component and the transducer,

- better analysis of AE signals including determination of energy, amplitude, spectra and the statistical distributions of these parameters.

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2. Particular Objectives and Research Program

This work is done especially for applications on the reactor pressure vessels. The herein used materials are 22 NiMoCr 3.7, 20 MnMoNi 55 and austenitic steels, mainly X6 CrNi 18.11. Base metals as well as welded joints are to be investigated.
The work shall be done in three steps:

- laboratory tests on modified CT and SEN specimens,
- laboratory tests on specimens with production defects,
- application of the developed methods and models to large scale tests.

The research program is including the simulation of AE signals and the characterization of acoustic emission transducers by their spectra and directivity pattern.

3. Experimental Facilities

- hydraulic testing machines modified for low noise,
- special grips for SEN and CT specimens without frictioning parts,
- analog and digital devices for signal analysis in the frequency range from 0.1 to 2 MHz,
- computer for signal analysis (especially frequency analysis) and determination of statistical distributions.

4. Progress todate

4.1. Modification of testing machines for low noise. Construction of special grips for CT specimens.

4.2. Frabrication of specimens (including welding) and metallurgical analysis.

4.3. Measurement of the directivity pattern of AE transducers.

4.4. AE measurements on CT specimens (50 mm)

4.5. Optimization of fracture mechanic models to the geometries and materials used in these experiments.

4.6. Measurements for determining the propagation of ultrasonic pulses in 100 mm thick plates and smaller specimens including welds.

4.7. Simulation of AE pulses with a laser.

5. Essential Results

5.1. In fatigue of CT specimens with large $\Delta K \approx 1500$ N/mm$^{3/2}$ (K = stress intensity) we have mainly acoustic emission from friction of the crack flanks. For smaller values of $\Delta K \approx 700$ N/mm$^{3/2}$ there is no more AE of friction.

5.2. The micro-structure of the ES-welded joints was mainly heterogeneous due to the high heat input involved during ES-welding (fig. 1). The base material is well tempered martensite. In the outer regions, the heat affected zone contains pearlite and ferrite in a lamellar constellation. Closer to the weld seam, the propagation of pearlite and upper bainite increases. The region of the HAZ next to the seam has coarse grain. It consists exclusively of upper bainite with rather large stripes of cementite inside the ferrite grains. Also the weld zone itself is purely upper bainite, but this has a crystallized dendritic structure. From measurements of hardness it can be expected that the region next to the base material is the weakest. The coarse grain region next to the weld is sensitive to cleavage fracture.

5.3. In a tension test on a CT specimen we have generally two maxima of AE energy rate:

- a first peak is due to friction of crack flanks and depending on their roughness (fig. 2 for a welded joint with friction and fig. 3 without friction); the AE signals can be characterized by the distribution of energy per event given in picture 4a (specimen of base material).

- a second peak is due to plastic deformation, micro-crack formation and macroscopic crack growth. These AE signals can be characterized by the distribution of energy per event given in picture 4b (specimen of base material).

By means of frequency and pulse area analysis, different groups of signals were detected. According to the time of their appearance they were correlated to plastic deformation, microvoid formation, macroscopic crack growth and friction (fig. 5).
Samples having the fatigue crack in different regions of the weld seam and in the HAZ all showed cleavage fracture at room temperature. At 60 ^{o}C, shear lips with a thickness of a few mm appeared.

5.4. The volume of the plastical zone in front of the crack tip is given by an expression

$$V_{pl} = a_4 K^4 + a_6 K^6$$

This expression is calculated by taking account of the plane strain and plane stress conditions in the materials. The coefficients a_4 and a_6 are dependent on the thickness of the CT specimen and the yield point of the material. For a specimen of thickness 50 mm we shall always see a dependence of V_{pl} on K with 4th power but for smaller specimens (e.g. thickness = 7 mm) and lower yield point there should be an influence of both parts of the above expression.

5.5. For the pressure vessel we have to assume that the major AE sources are near to the inner surface of the vessel wall because the stresses are highest at the inner side. In general the acoustic emission signals are detected at the outer surface of the vessel wall. For this geometry we have done experiments which are described in picture 6. The results are to be seen in picture 7. We see that the amplitude of the surface wave is dependent on the roughness of the outer surface of the vessel wall. Large-scale experiments are mainly done in the frequency range of 100 - 300 kHz, i.e. with wavelengths of cm. So we cannot assume for reactor pressure vessels that the surface wave is important for the propagation of acoustic emission signals.

5.6. We have simulated an acoustic emission source with a laser heating up the surface of a steel or aluminium plate. The generated acoustic pulse has a short duration (<500 ns) and the directivity pattern given in picture 8.

6. Next Steps

- Continuing AE measurements on CT and SEN specimens with signal analysis,
- determination of the correlations between AE parameters and stress intensity factor and development of models,
- measurement of the directivity pattern of natural AE sources (cracks).

7. Relation With Other Projects

These works are done on the base of the results of the projects RS 31 to RS 31/3 and in cooperation with Battelle-Institut at Frankfurt (RS 191).

8. References

E. Waschkies:
Schallausbreitung bei der Schallemissionprüfung.
(1977)

9. Degree of Availability of the Reports

The report cited above is available from IzfP, Saarbrücken.

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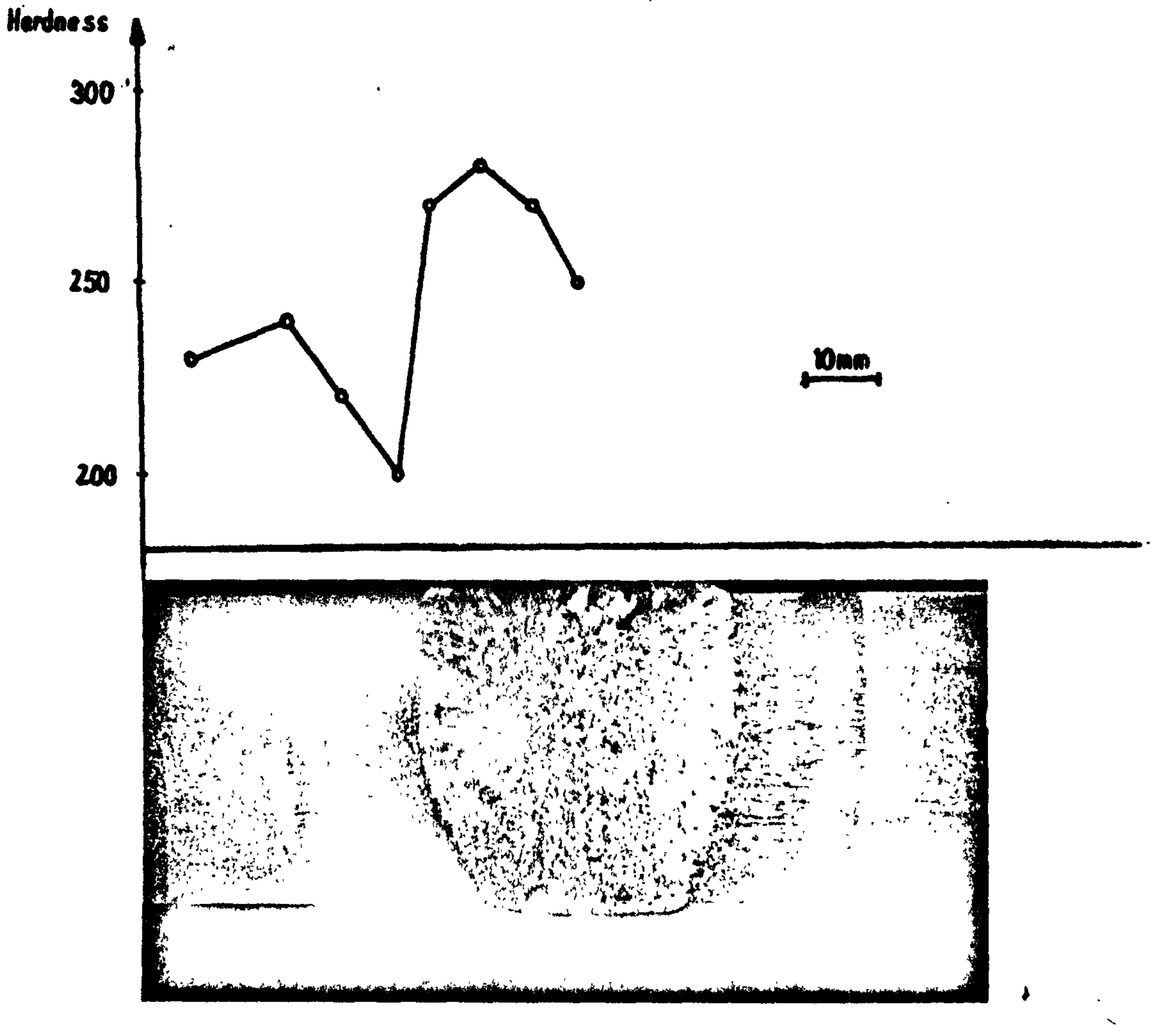

Fig. 1: photo-micrograph and hardness of a welded joint.

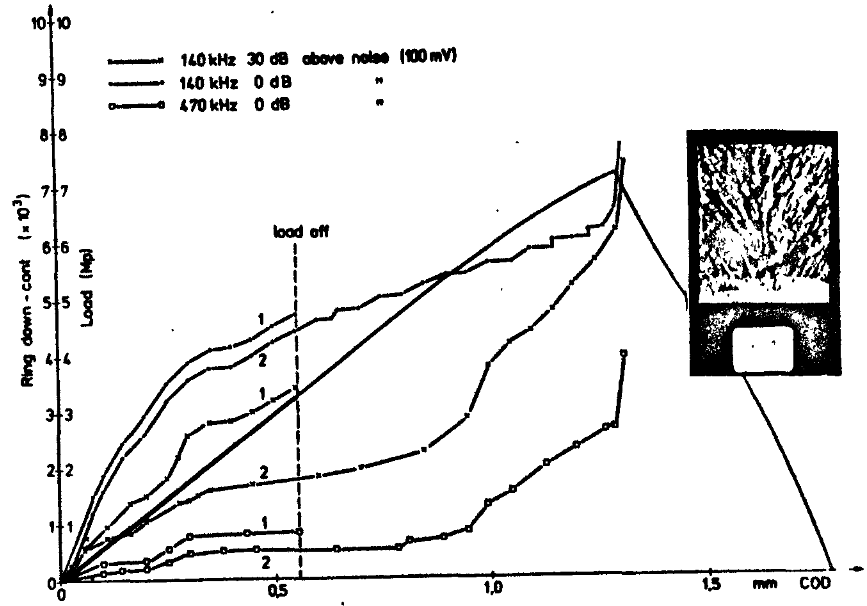

Fig.: 2

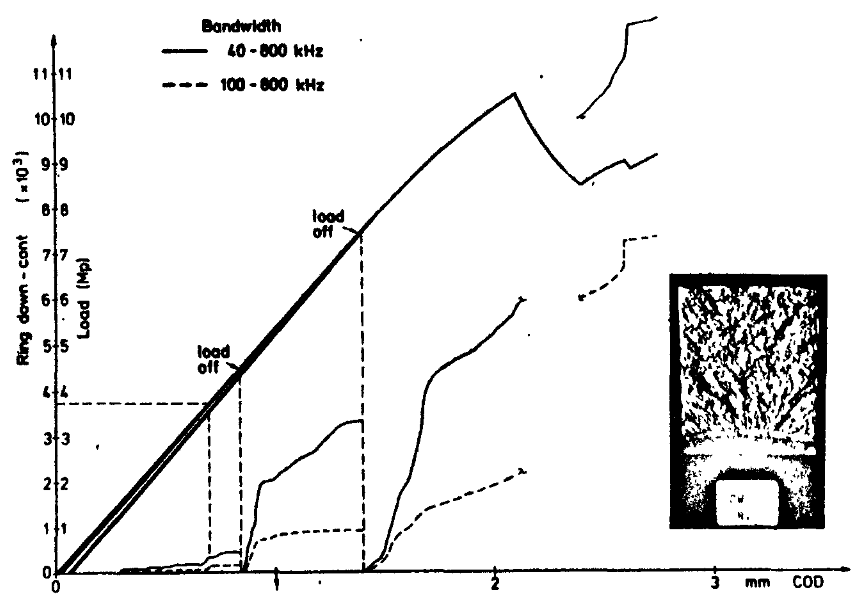

Fig: 3

Fig. 2: Ring-down-count and load against crack-opening for CT specimen (welded joint); rough surface of the crack.

Fig. 3: Ring-down-count and load against crack-opening for CT specimen (welded joint); smooth surface of the crack

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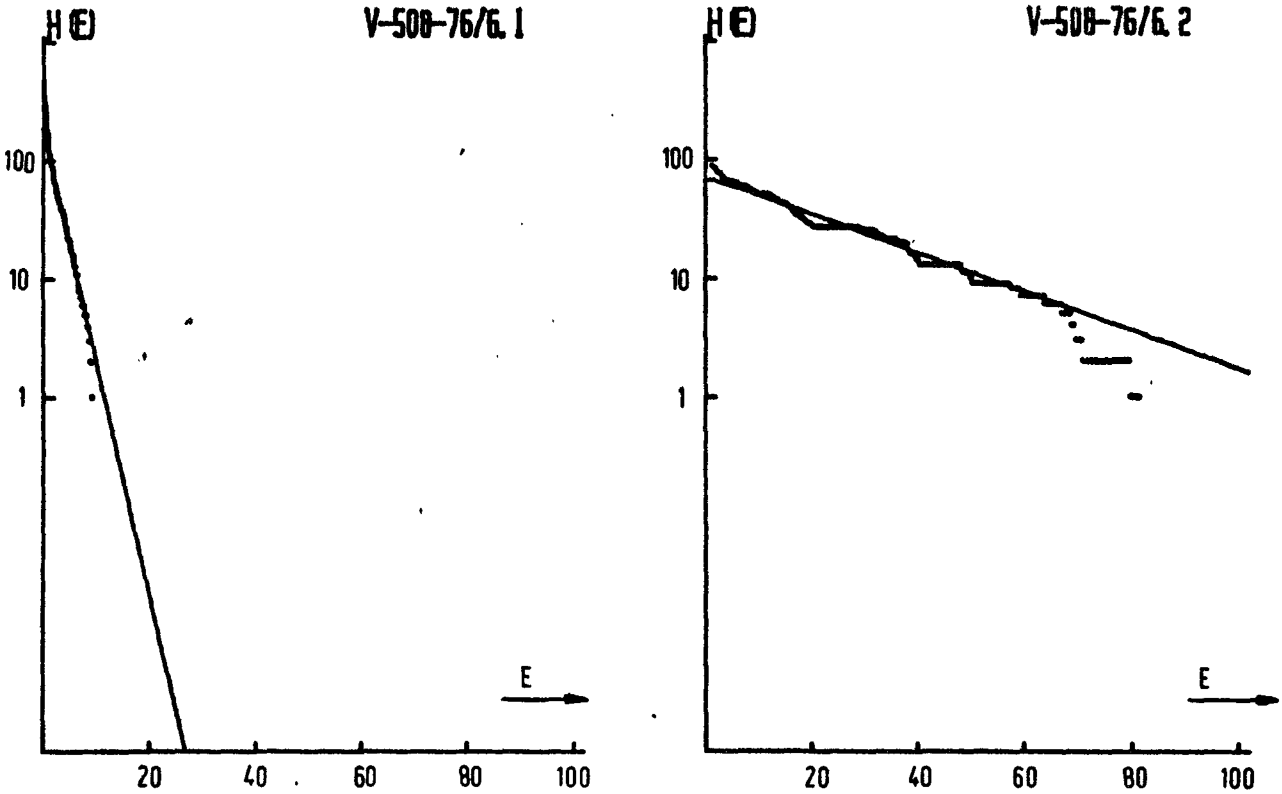


Fig. 4: Number H of AE events with energies equal to or greater than E (on arbitrary scale) in a tension test on a CT-specimen (base material 22 NiMoCr 3.7)
a) friction of the crack-surface
b) micro-cracking and macroscopic crack-growth.

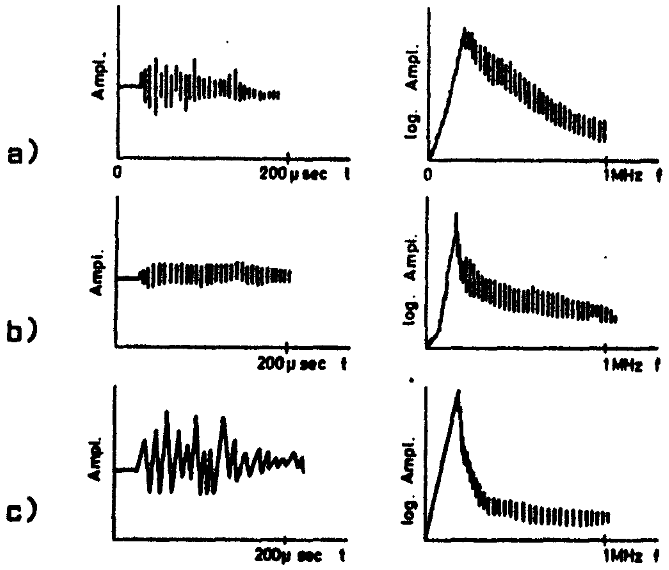
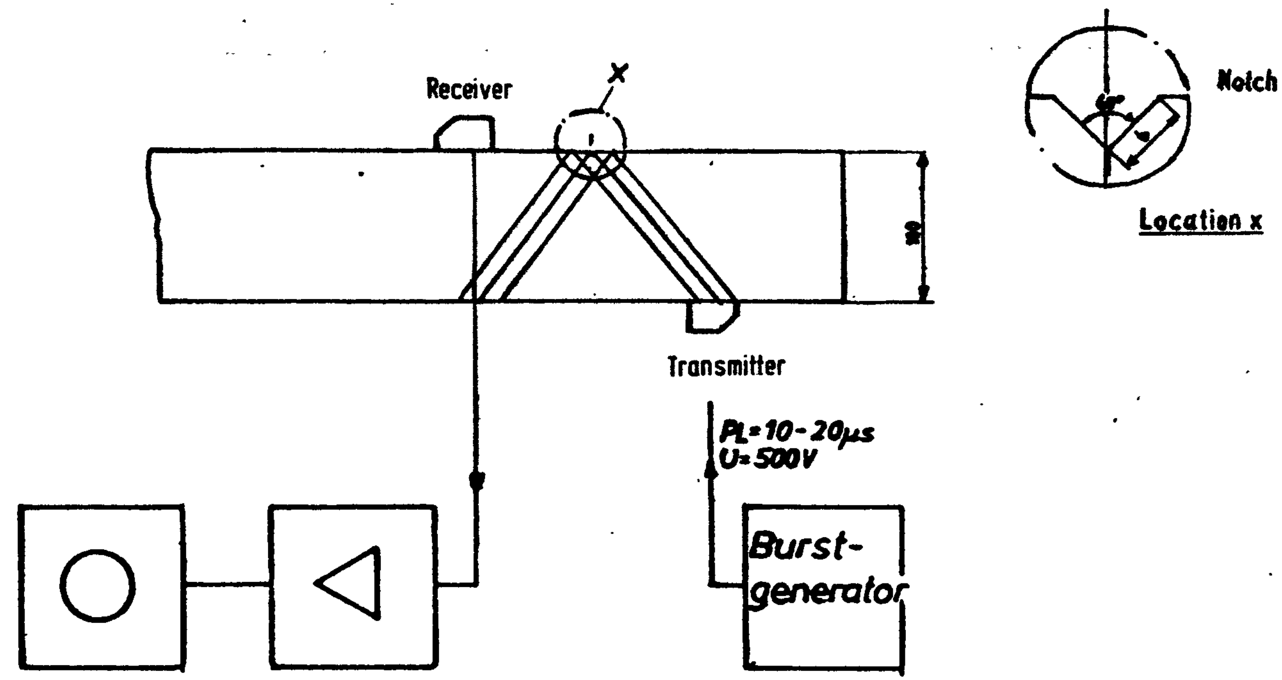


Fig. 5: typical AE signals and spectra:

a) friction of the crack surface

b) micro-cracking

c) macroscopic crack growth



IzfP

Blockdiagram for generating surfacewaves

Fig.6

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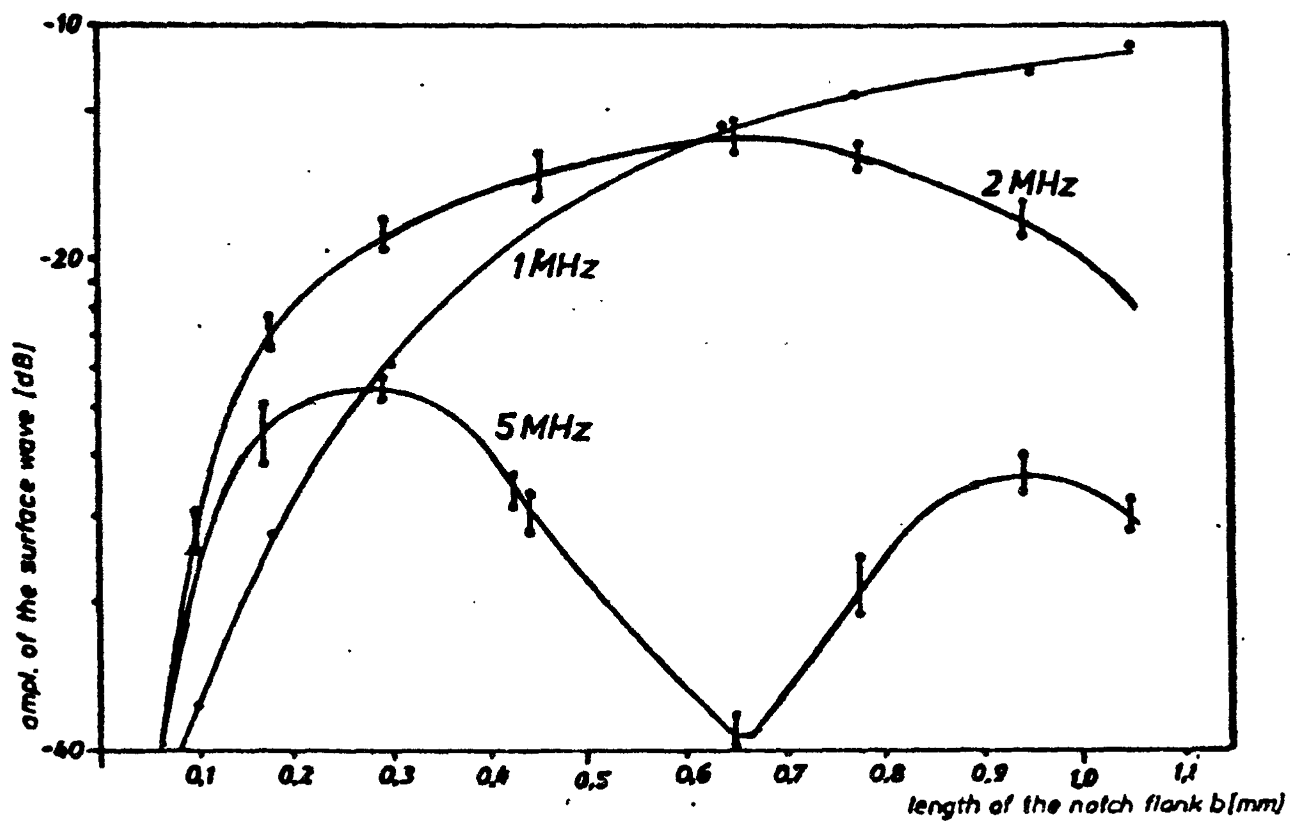

IzfP	Relative ampl. of the generated surface wave (in reference to the incident transverse beam) as function of the length of the notch flank.	Fig.: 7

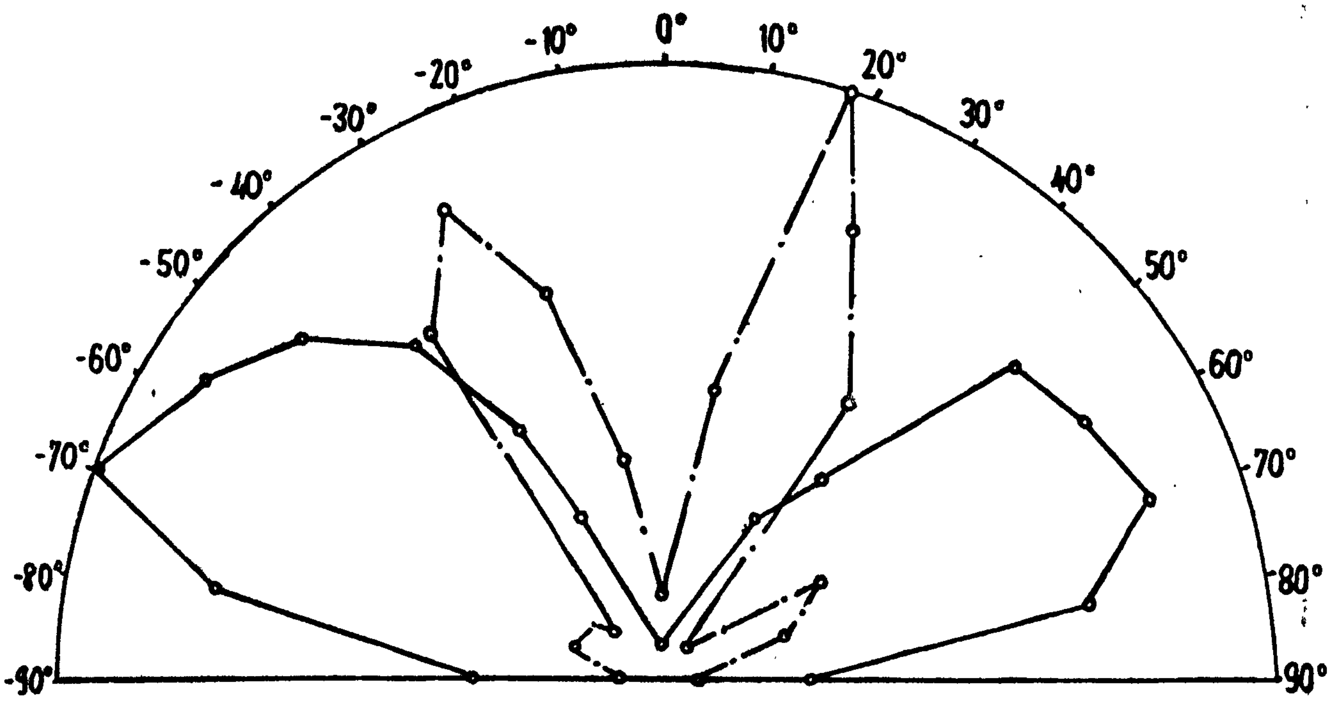

Fig. 8: Directivity pattern of a laser-generated acoustic source

——— longitudinal wave

—·— transverse wave

Classification: 12.3	
Title 1 (Original Language): Weiterentwicklung von Ultraschall-Prüfverfahren und dazu erforderlichen Einrichtungen für wiederkehrende Prüfungen an Reaktordruckbehältern (RS 2703 - II.4.2, Jahresbericht A 76)	COUNTRY: BRD FRG SPONSOR: BMFT ORGANIZATION: M.A.N., Nürnberg
Title 2 (English): Onward Development of Ultrasonic Inspection Techniques and Allied Equipment for In-Service Inspection of Reactor Pressure Vessels	Project Leader: Dipl.-Ing. J. Lindner
Initiated (Date): 1.2.1976 Status: Dec. 1976 Cotninuing Completed (Date): 1.2.1978 Last Updating (Date): Dec. 1976	

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 of 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:

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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.

1522

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

Actual test results from a large vessel wall specimen are available for a closer examination of perhaps significant indications with the aid of special methods of analysis. The following four methods of analysis were tested:

Single-probe dual-frequency echo dynamics
Tandem echo dynamics
Tandem echo dynamics with focus head
Line holography

Conceptual data and designs are available on the inspection systems to be developed for the individual areas. Prototypes are being built to test the capabilities of these systems. For the probe with adjustable sound parameters there are certain dimensional limitations because the probe is also proposed to be used in areas of restricted space.

Conceptual designs have been prepared for the various manipulators. Detailed designwork has been started.

For data logging and data display/printouts, work has commenced on the further development of software for on-line data processing and on increasing the data flow to all measuring channels.

6. Results

Most of the work is still in progress and no detailed results are as yet available.

The tests on the methods of analysis have shown that according to present knowledge focus technology and line holography appear to be particularly suitable methods for the analysis of indications. Computer codes are available for calculating the three-dimensional tandem echo dynamics.

7. Next Steps

Work under this programme will be continued in all areas with special emphasis on:

- Testing of new inspection systems on test specimens
- Study of various parameters as to the quality of indications for focus technology
- Testing of working models of new manipulator elements
- Onward development and testing of improved data display/ printout methods.

8. Relation with Other Projects

Reference is made to what has been said in Report A 74 with regard to RS 2702.

9. References

During the period under review the following summarized reports have been issued in the German language:

1524

Author	Title	Date
BAM	Three-dimensional Echo Dynamics for Tandem Technology	September 76
BAM	Fault Analysis on a Large Vessel Wall Specimen	October 76

10. 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, Köln.

1527

Classification: 12.3	
Title 1 (Original Language): Erhöhung der Auffindwahrscheinlichkeit von Fehlern durch Verbesserung des Signal-Rausch-Verhältnisses bei der Ultraschallprüfung (RS 190)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Battelle-Institut e.V.
Title 2 (English): Increasing the Probability of Flaw Detection by Improving the Signal-to-Noise Ratio in Ultrasonic Testing	**Project Leader:** R. von Klot
Initiated (Date): 1.1.1976 Completed (Date): 31.8.1976 Status: Continuing Last Updating (Date): December 31, 1976	

1. General Aim

The project is aimed at investigating whether it is possible in ultrasonic testing to improve the signal-to-noise ratio by application of the signal averaging technique.

2. Particular Objectives

It is to be investigated whether signal averaging can improve the signal-to-noise ratio of the echoes from flaws close to claddings (e.g. in reactor pressure vessels and primary pipe systems).

3. Research Program

The investigations were carried out on a clad specimen with a bore (diameter 5 mm) near the cladding and on a large vessel wall test specimen with two cracks originating directly below the cladding (crack depth 7.2 mm and 14.7 mm).

4. Experimental Facilities, Computer Codes

A commercial materials testing instrument and an averaging computer were used for the investigations. The exponential average value is computed by the following algorithm:

$$A_n = S_n/G + A_{n-1} \cdot (1 - 1/G)$$

1526

S_n	n-th input signal	A_n	stored result after n sweeps
G	weighting factor	A_{n-1}	stored result after n-1 sweeps

5. Progress to Date

The echo amplitude and the signal-to-noise ratio were measured. The velocity of the probes, the direction of the probe velocity relative to the defect (parallel $\parallel$ and perpendicular $\perp$), and the weighting factor were varied in these measurements.

6. Results

For the perpendicular direction of the probe velocity, the amplitude of the echo from the defect first remains constant and then drops rapidly as the weighting factor increase (Fig. 1). For the parallel velocity direction almost no drop occurs up to $G = 2^{10}$.

The signal-to-noise ratio first rises and then, after a maximum, declines as the weighting factor increases (Fig. 2). With growing probe velocity the maximum is shifted toward smaller values of the weighting factor and thus is reduced.

In the vessel wall test specimen the echo amplitudes for probe velocities of 2.0 mm/s and 4.0 mm/s are almost identical.

The signal-to-noise ratio passes through a maximum as a function of the weighting factor (Fig. 3). The maximum increases with rising probe velocity, but is not shifted. The signal-to-noise ratio can be improved by up to 10 dB.

7. Next Steps

The probe velocity is to be increased to about 50 mm/s by modification of the averaging computer. Defects close to claddings will be analyzed by single and tandem probes from the clad side and from the reverse side.

8. Relation with Other Projects

9. References

R. von Klot, A. Sahm: Erhöhung der Auffindwahrscheinlichkeit von Fehlern durch Verbesserung des Signal-Rausch-Verhältnisses bei der Ultraschallprüfung, RS 190, Report of August 1976 by Battelle-Institut e. V., Frankfurt am Main, to the Federal Ministry of Research and Technology (in German).

10. Degree of Availability of the Reports

The above-cited report is available upon request from Bundesministerium für Forschung und Technologie, Referat 313, Postfach 120 370, D-5300 Bonn 12.

1528

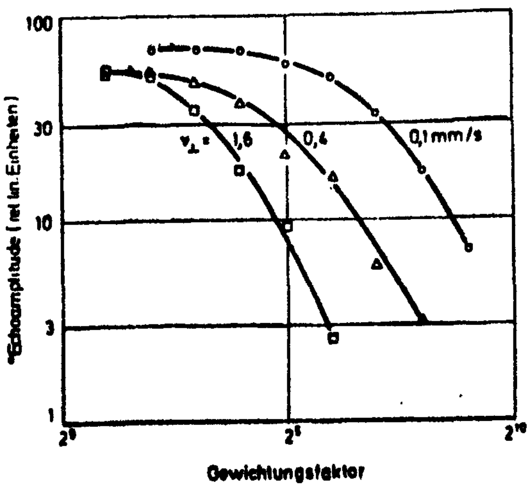

Fig. 1: Amplitude of the echo from the bore in the specimen as a function of the weighting factor for different probe velocities $v_\perp$. Shear waves 4 MHz.

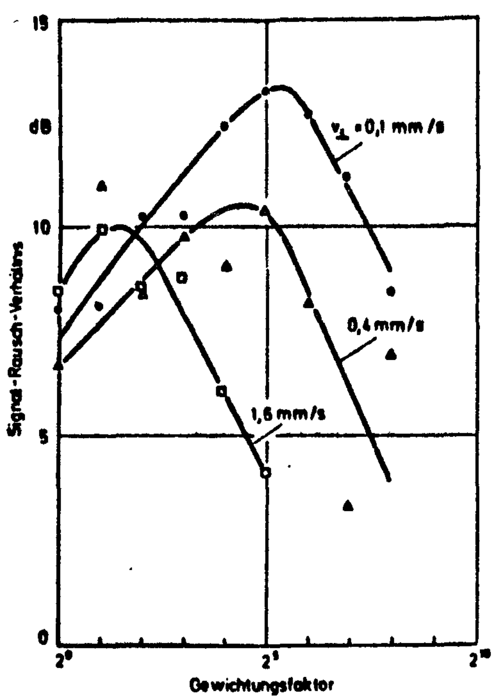

Fig. 2: Signal-to-noise ratio of the echo from the bore in the specimen as a function of the weighting factor for different probe velocities $v_\perp$. Shear waves 4 MHz.

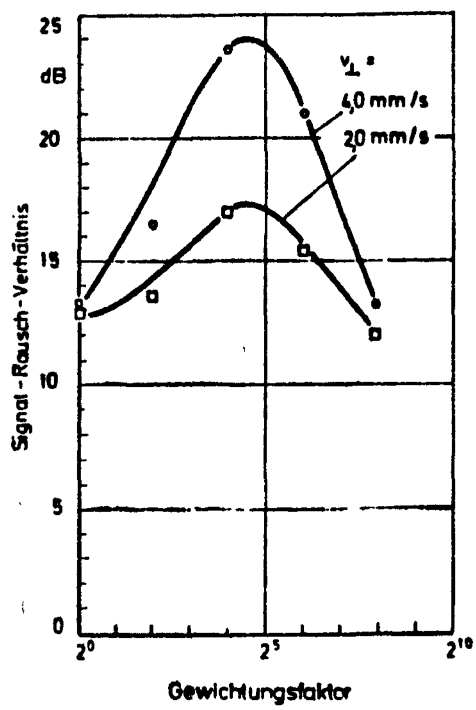

Fig. 3: Signal-to-noise ratio of the echo from the crack (depth 7.2 mm) in the vessel wall test specimen as a function of the weighting factor for different probe velocities $v_\perp$. Shear waves 1 MHz.

1529

Classification: 12.3	
Title 1 (Original Language): Studie über die Anwendung des Wirbelstromimpulsverfahrens zur Qualitätssicherung von Kernkraftwerkskomponenten (RS 229 - II.3.2, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** BAM, Berlin
Title 2 (English): Study on the application of pulsed eddy current methods to quality control of nuclear power plant components	**Project Leader:** Dr. G. Wittig
Initiated (Date): [illegible].10.76 **Status:** Continuing **Completed (Date):** 31.12.1977 **Last Updating (Date):** December 1976	

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 Objectivs

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.

1530

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

Program for numerical calculation of distributions of magnetic fields and eddy currents.

5. Progress to Date

Ad 3.1. Evaluation of present and procured literature.
Visit to KWU with discussions about test problems in question.

Ad 3.2. Numerical calculation of temporal and local distribution of the magnetic field and the eddy current density in a half-space with electrical conductivity of austenitic steel. Excitation by a plane magnetic field of variable duration.

6. Results

Program for calculation of field and eddy current distributions.
The project is in an initial phase in this time.

7. Next steps

Ad 3.1. Continuation of the evaluation of reports and of the procurement of informations about test problems.

Ad 3.2. Further calculations of field and eddy current distributions.

8. Relation with other Projects

- RS 89 - II.3.2
- RS 102- 18 - II.3.2

9. References

-

10. Degree of Availability of the Reports

-

1531

Classification: 12.3	
Title 1 (Original Language): Mehrfrequenz-Wirbelstromprüfung Phase 1: Aufbau eines Mehrfrequenz-Geräteprototyps (RS 231 - II.3.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Ges., IzfP, Saarbr.
Title 2 (English): Multifrequency - eddy current testing Phase 1: Construction of a multifrequency prototype	**Project Leader:** Dipl.-Phys. R. Becker
Initiated (Date): 1.9.1976 **Completed (Date):** 29.2.1977 **Status:** Continuing **Last Updating (Date):** December 1976	

1. General Aim

The project has the aim of building up a prototype for the inspection by means of the multifrequency eddy current method. Starting from the results of RS 102-18, section "eddy current", the equipment shall be applicable to the inspection in production lines as well as to the recurrent inspection of reactor components of the type BWR, PWR and FSR. The concept of the equipment shall harmonize with the conditions of the large application field. Over all, great importance must be spended upon safety against disturbances and upon easy handling.

2. Particular Objectives

The equipment consists of 2 parts:

- the transmitter-receiver-unit:
  The frequencies are given after multiplexing each after the other into the coil. The switching is so fast, that an inspection velocity of 3 m/s with a local resolution of 1 mm is achieved. The advantage of this proceeding compared with the simultaneous method is the small electronic expense and the flexibility concerning the quantity of frequencies and the frequency range.
  Respectively, in one of 3 ranges the frequencies can be tuned continuously and in any combination:

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range 1: 10 kHz - 600 kHz
range 2: 1 kHz - 60 kHz
range 3: 0.1 kHz - 6 kHz

Besides the oscillators the transmitter-receiver unit consists of the coil driver, the amplitude-phase-meter, the compensation network and the display to figure the test information (impedance of the coil). The coil is driven over a cable, which has a length of up to 30 m and can be an absolute or a differential type. In both cases the compensation is made in an automatical manner.

- the evaluation unit:
  This performs the separation of the parameters and produces the read-out value, which is settled of the disturbing signals and is only a function of one parameter to be chosen. The evaluation is realized by means of digital computer circuits and allows the simultaneous determination of several read-out values coordinated to different parammeters.

3. Research Program

The electronic circuits and components are to develop and to integrate within the prototype described in 2.

4. Progress to Date and Results

The block diagram of the equipment is worked out upon the experiences with the laboratory test device, which is arisen in the preceeding project RS 102-18.
The building up of the transmitter-receiver unit is short of conclusion.

5. Next Steps

Construction of the evaluation unit will be started and connection of the both units will be realized.

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<table>
<tr><td colspan="3">Classification: 12.3</td></tr>
<tr><td colspan="2">Title 1 (Original Language):
Untersuchungen zur Leistungsfähigkeit der akustischen Holographie, vor allem im Vergleich zu fokussierenden Prüfköpfen bei der ZfP
(RS 0230 - II.3.2, Jahresbericht A 76)</td><td>COUNTRY: BRD
SPONSOR: BMFT
ORGANIZATION: BAM</td></tr>
<tr><td colspan="2">Title 2 (English):
Investigations of the efficiency of acoustical holography especially in comparison to focussed beams in NDT</td><td rowspan="3">Project Leader:
Dr. J.Kutzner
Dr. H.Wüstenberg</td></tr>
<tr><td>Initiated (Date): 1. 10. 1976</td><td>Completed (Date): 31. 3. 1979</td></tr>
<tr><td>Status: Continuing</td><td>Last Updating (Date): December 1976</td></tr>
</table>

1. General Aim

Development of simplified methods of
- acoustical holography with numerical reconstruction
- flaw sizing by scanning with focussed beams.

Application of the developed methods to the analysis of ultrasonic indications under practical circumstances.

2. Particular Objectives

-

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.

1534

3.3 Holography in tandem-technique
Investigations of the efficiency of this method, experiments on test-blocks.

3.4 Construction and testing of focussing probes
Checking of the available construction method, manipulator for focussing probes, adaption of the focussing probes on the curvature of the surface.

3.5 Experiments with the probe-systems developed for holography- and focussing 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 Some theoretical investigations to the resolution, construction of a two transducer-probe, first experiments.

To 3.2 Testing of the developed reconstruction- and interface-software for linear scanning in normal-, oblique- and tandem-incidence.

To 3.3 Software for linear scanning.

To 3.4 -

To 3.5 -

6. Results

To 3.1 The resolution is determined mainly by the divergency of the probes.

To 3.2 The software-development for linear scanning is completed.

To 3.3 The software-development for linear scanning in tandem-technique is also completed.

To 3.4 -

To 3.5 -

7. Next Steps

To 3.1 Continuation of the theoretical investigations, testing and further development of the constructed probe.

To 3.2 Theoretical and experimental investigations to get informations about the distance of the scanning points. Writing a technical report to the equipment for linear acoustical holography (hardware and software).

To 3.3 -

To 3.4 Checking of the available construction method by reference tests.

To 3.5 -

8. Relation with other Projects

Real-time ultrasonic holography (IZfP-Saarbrücken), reactor security research program RS 2703.

9. References

-

10. Degree of Availability of the Reports

IRS, Köln

131-31-10a/4111-10

Classification : 12-3

1537

Titre	Pays :
Détectabilité par ultrasons des défauts en compression.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais)	Organisme exécuteur : CEA/DTECH-STA
Ultrasonic detectability of flaws under compression.	Responsable : A.C.PROT (DSN-SETSSR)
Date de démarrage : 01/07/77 Date prévue d'achèvement 31/12/78 Etat actuel : A lancer Dernière mise à jour : 23/11/76	Scientifiques : R. SAGLIO (STA)

Objectif général :

Etude de l'influence du champ de contraintes sur la détectabilité des défauts lors d'un contrôle par ultrasons.

Objectifs particuliers :

Il s'agit d'étudier, sur éprouvettes de traction représentatives (forte épaisseur), l'influence du champ de compression sur la détectabilité de défauts connus (décelés par ultrasons en l'absence de contrainte) et d'apprécier comment varie le dimensionnement de ces défauts en fonction de la contrainte appliquée.

Prochaines étapes :

- Détection de défauts réels introduits volontairement dans des éprouvettes de traction.
- Etude de l'évolution de ces défauts au cours d'essais de fatigue par traction-compression.
- Dimensionnement pendant les phases de traction puis compression.

Relation avec d'autres études :

Etude probabiliste de la rupture d'une cuve de réacteur nucléaire.

131-31-10b/4111-10

Classification : 12-3 1539

.Titre	Pays :
Détection des fissures dans les rebords internes de tubulures.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Crack detection of the nozzle inner edges.	Organisme exécuteur : CEA/DTECH-STA
	Responsable : A.C.PROT (DSN-SETSSR)
Date de démarrage : 01/01/77 Date prévue d'achèvement : 31/12/78 tat actuel : A lancer Dernière mise à jour : 23/11/76	Scientifiques : R.SAGLIO (STA)

Objectif général :

Le rebord interne des turbulures est une zone particulièrement sollicitée dans une cuve de réacteur. L'étude consiste à rechercher la méthode optimale pour détecter et éventuellement dimensionner les défauts qui peuvent prendre naissance pendant le fonctionnement.

Objectifs particuliers :

- Evaluer de manière critique les méthodes actuellement envisagées en particulier ultrasons et courants de Foucault.
- Etudier la faisabilité d'un contrôle en ultrasons focalisés.
- Etudier la possibilité d'application aux inspections périodiques. de cuves de réacteurs nucléaires.

Installations expérimentales et programme :

Utilisation des moyens actuels de la STA et en particulier de la maquette de piquage SENA.

Prochaines étapes :

Démonstration de la possibilité de contrôle par traducteurs ultrasonores focalisés ou/et courants de FOUCAULT (fin 1977)

131-31-10c/4111-10

Classification : 12-3

Titre	Pays : FRANCE
Etude de l'influence des divers revêtements sur le contrôle ultrasonore par l'extérieur des circuits primaires de réacteurs.	Organisme directeur : CEA/DSN
Titre (anglais) Study on the effect of coatings on the ultrasonic testing of primary circuits of nuclear reactors from the outside.	Organisme exécuteur : CEA/DTECH-STA Responsable : A.C.PROT (DSN-SETSSR)
Date de démarrage : 01/01/77 Date prévue d'achèvement :31/12/78 Etat actuel : A lancer Dernière mise à jour : 23/11/76	Scientifiques : F.DUBOIS (STA)

Objectif général :

Il s'agit d'évaluer l'influence des peintures actuellement utilisées pour la protection des cuves de réacteur, sur les possibilités de contrôle par ultrasons.

Objectifs particuliers :

- Il est nécessaire d'évaluer les modifications introduites par l'environnement (température, irradiation) sur les qualités d'adhérence et de résistance à l'abrasion de peintures afin de connaître l'influence possible sur le couplage des traducteurs ultrasonores.
- Eventuellement cette étude peut déboucher sur un "meilleur choix".

Prochaines étapes :

- Réalisation d'éprouvettes métalliques représentatives.
- Essais ultrasonores avant influence de l'environnement.
- Essais en température et sous irradiation.
- Essais ultrasonores comparatifs.
(Mise en route du programme fin 1977)

131-31-10d/4111-10

Classification : 1543 12-3

Titre	Pays : FRANCE
Programme européen de contrôle de tôles fortes par ultrasons.	Organisme directeur : CEA/DSN
Titre (anglais) European program of ultrasonic testing of heavy section steel-plates.	Organisme exécuteur : CEA/DTECH-STA Responsable : A.C.PROT (DSN-SETSSR)
Date de démarrage : 01/03/77 — Date prévue d'achèvement : 01/07/78 Etat actuel : A lancer — Dernière mise à jour : 23/11/76	Scientifiques : R.SAGLIO (STA)

Objectif général :

Participation à un programme européen de contrôle de tôles de forte épaisseur du programme HSST, mises à disposition par les USA.

Objectifs particuliers :

Détection des défauts volontairement introduits dans les tôles en utilisant la technique des traducteurs focalisés et éventuellement l'holographie acoustique.
Un tel essai devrait constituer une base pour la qualification de la méthode mise au point au CEA, puisqu'il sera suivi de la découpe des tôles et d'une corrélation entre examen ultrasonore et macro- ou micrographies.

Installations expérimentales et programme :

Les installations sont celles actuellement disponibles à la section des techniques lancées (STA).
- Cuve de contrôle par immersion (locale ou non).
- Pantographe
- Appareils et traducteurs à ultrasons.

Etat de l'étude :

Avancement à ce jour :

Les types de contrôles sont définis. Les tôles doivent être mises à disposition du 28 février au 30 juin 1977.

.../...

1544

Prochaines étapes :

Contrôle des 3 tôles HSST (2ème semestre 1977).

131/31/20/4111-10 Classification : 12-3

Titre	Pays : FRANCE
Etude des filtrages optiques de films radiographiques	Organisme directeur : CEA/DSN
Titre (anglais) Study of optical filtering of radiographic films.	Organisme exécuteur : CETIM
	Responsable : A.C.PROT (DSN-SETSSR)
Date de démarrage : 19/5/76 Date prévue d'achèvement :19/11/77 Etat actuel : en cours Dernière mise à jour : 23/11/76	Scientifiques : FLAMBARD PARASKEVAS

Objectif général :

Recherche d'un procédé de filtrage optique permettant d'augmenter le contenu d'informations délivrées par les films radiographiques.

Objectifs particuliers :

Amélioration du contraste dans la détection soit de fissures mal orientées soit de présence d'un bruit de fond (pièce radioactive et/ou en ambiance radioactive).

Etat de l'étude :

Avancement à ce jour :

L'étape est en cours de finition. Un document sera diffusé qui permettra après analyse (étape 2) de lancer l'étude proprement dite.

Prochaines étapes :

1) Etablissement d'un document faisant le point des connaissances en la matière (fin 76)
2) Choix des orientations (fin 77)
3) Lancement des études correspondantes.
4) Choix du procédé à développer
5) Etude de ce procédé.

131-31-30/4111-10

Classification : 12-3 1547

Titre	Pays :
Détermination fine de la dimension de défauts par mesure de temps de parcours ultrasonores.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais)	Organisme exécuteur : Ecole Centrale
Accurate flow size determination by measurement of ultrasonic delay time.	
	Responsable : A.C.PROT (DSN-SETSSR)
Date de démarrage : 01/01/77 Date prévue d'achèvement :31/12/77 at actuel : A lancer Dernière mise à jour : 23/11/76	Scientifiques : P.AZOU D. de VADDER

Objectif général :

Il s'agit d'une étude analytique associée à des essais de laboratoire, destinés à démontrer la faisabilité d'une méthode de dimensionnement fine des défauts décelés par ultrasons.

Objectifs particuliers :

Montrer analytiquement et vérifier expérimentalement que les dimensions d'un défaut peuvent être atteintes par mesure des temps de parcours de l'onde ultrasonore au cours du balayage du défaut par le traducteur ultrasonore.

131-31-40

Classification : 12-3

Titre Moyen d'inspection de circuit primaire de réacteurs.	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) Nuclear reactor primary circuit inspection means.	Organisme exécuteur : CEA - DPR -STEPPA Responsable : A.C.PROT (DSN - SETSSR)
Date de démarrage : 1/1/77 Date prévue d'achèvement : 31/12/78 Etat actuel : A lancer Dernière mise à jour : 22/11/76	Scientifiques : J.P.VERTUT (DPR)

Objectif général :

Les dispositifs mis au point pour le contrôle de cuves ne constituent qu'une partie de la solution des problèmes d'inspection.
Il reste à définir des moyens propres à atteindre de manière autonome des endroits contaminés et peu accessibles.
Cette étude vient en appui de celle commencée en 1976 par la section EMH.

Objectifs particuliers :

Il s'agit essentiellement d'étudier un dispositif autonome susceptible de se rendre dans des endroits inaccessibles ou difficilement accessibles. (soudure de tuyauteries, boite à eau de G.V., cuve de réacteur rapide) pour effectuer des inspections de divers types : ultrasons, courants de Foucault, visuel par télévision etc...) et d'en démontrer la faisabilité.

Etat de l'étude :

Avancement à ce jour :

Des schémas de principe ont été étudiés par le STEPPA au cours de l'année 1976 et serviront de base à l'étude envisagée.

Prochaines étapes :

- Etude d'un dispositif autonome (été 77)
- Maquette de faisabilité. (fin 77)

131-31-51/4111-612

Classification : 12-3 1551.

Titre Développement de méthodes de contrôle par ultrasons par l'extérieur des circuits primaires PWR.	Pays : FRANCE Organisme directeur : CEA
Titre (anglais) Development of ultrasonic testing methods from the outside of PWR primary circuits.	Organisme exécuteur : CEA/DTECH Responsable : M.CONTRE (DTECH-STA)
Date de démarrage : 01/76 Date prévue d'achèvement : 10/80 Etat actuel : en cours Dernière mise à jour : 1/4/77	Scientifiques : M.SAGLIO

Objectif général :

Permettre le contrôle par l'extérieur du circuit primaire des réacteurs PWR et plus particulièrement des cuves.

Objectifs particuliers :

Etudier les méthodes de contrôle par ultrasons propres à permettre un suivi précis des défauts éventuellement détectés par l'intérieur, sans nécessiter le démontage des internes.

Installations expérimentales et programme :

- Examen des modifications des plans de Génie Civil permettant cette inspection par l'extérieur.
- Etude des méthodes de contrôle
- Eventuellement étude d'un avant projet de machine d'inspection.

Etat de l'étude :

1) Avancement à ce jour :

- Etude des problèmes d'immersion locale - réalisation de joints d'étanchéité.
- Diminution des encombrements des traducteurs par utilisation de miroirs.

2) Résultats essentiels :

Mise au point d'un prototype expérimental de faisabilité de contrôle par l'extérieur en immersion locale.

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Prochaines étapes :

Miniaturisation de la cuve d'immersion locale
Etude de traducteurs adaptés aux géométries et permettant de satisfaire aux exigences d'encombrement.
Avant projet de machines d'inspection.

Titre	Pays :
Surveillance en continu par émission acoustique du circuit primaire des réacteurs à eau.	FRANCE
	Organisme directeur : CEA
Titre (anglais)	Organisme exécuteur : CEA/DTECH
Continous surveillance of water reactors primary circuits using acoustic emission	Responsable : CONTRE (DTECH-STA)
Date de démarrage : 01/76 Date prévue d'achèvement : 12/80 Etat actuel : En cours Dernière mise à jour : 1/4/77	Scientifiques : R. SAGLIO ASTY

Objectif général :

Définir et construire un matériel d'émission acoustique spécifique adapté à la surveillance en continu du circuit primaire des réacteurs à eau, c'est-à-dire susceptible de fonctionner dans l'ambiance d'un réacteur en fonctionnement.

Objectifs particuliers :

Permettre en 1977 la surveillance du dôme du pressuriseur de FESSENHEIM 1.

Installations expérimentales et programme :

Equipements électronique :
- Microprocesseur
- Echelle de comptage
- Electronique d'acquisition.

131-33-60/4112-60 Classification : 12-3 1555

Titre	Pays : FRANCE
Utilisation de la neutronographie pour l'inspection en service des réacteurs nucléaires et le contrôle non destructif des composants du coeur.	Organisme directeur : CEA/DSN
Titre (anglais) Use of neutron radiography in in-service inspection nuclear reactor and non destructive testing of core composants.	Organisme exécuteur : CEA/DSN-SEESNC Responsable : M.HOUELLE
Date de démarrage : 01/01/73 Date prévue d'achèvement : 31/12/78 Etat actuel : Etude en cours Dernière mise à jour : 08/11/76	Scientifiques :

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 serviront à effectuer les essais d'examens de pièces.

Objectifs particuliers :

Valeur du procédé pour :
1) L'examen de pièces épaisses en acier.
2) L'examen d'éléments combustibles de réacteurs à eau.
3) L'examen de matériels divers (ensembles colles).

Installations expérimentales et programme :

Réacteur pour neutronographie MIRENE.

Etat de l'étude :

1) Avancement à ce jour :

Le réacteur MIRENE a divergé en novembre 1976 et sera disponible pour des examens de pièces à partir de décembre 1976.

2) Résultats essentiels :

Les essais d'examens effectués antérieurement sur une installation semblable à MIRENE ont donné les résultats essentiels suivants : Ce type de réacteur installé près de deux réacteurs de puissance, PHENIX et RAPSODIE, permet un contrôle neutronographique satisfaisant du combustible nucléaire. Des essais effectués avec cette même installation ont montré qu'il était possible de contrôler des pièces en acier de 5 cm d'épaisseur.

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Prochaines étapes :

Essais d'examen de pièces test en acier et de pièces diverses de réacteur électrogène, susceptibles d'être démontées et transportées pour le contrôle dans une installation du type MIRENE.

Relation 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.

Documents de référence :

"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).

"Radiography with Neutrons", M.HOUELLE, C.MERCIER, H.REVOL - British Nuclear Energy Society - Conference at the University of Birmingham - 10/11 september 1973.

"Neutronographies de pièces épaisses en acier", M.HOUELLE - Rapport SEESNC N° 123 - février 74.

"Rapport provisoire de sûreté du miniréacteur pour neutronographie MIRENE", - Rapport SEESNC N° 122, 1974.

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133-46-61/4114-01

Classification : 12.3

Titre	Pays :
Utilisation de calculateurs dans les systèmes de protection.	FRANCE
	Organisme directeur : CEA
Titre (anglais)	Organisme exécuteur : CEA/LETI (Grenoble)
Computer protection system	Responsable : Mr. TOURNIER
Date de démarrage : 1/1/75 Date prévue d'achèvement : 1/1/79 Etat actuel : Etude en cours Dernière mise à jour : 21/12/76	Scientifiques :

Objectif général :

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

Objectif particulier :

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 protections, 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.

Prochaines étapes :

a) Elaboration des règles pour la programmation d'un tel système.

Elles doivent contenir et la modularisation stricte des programmes à utiliser et une liste des éléments de programmation à ne pas utiliser à cause de la fiabilité des programmes.

b) Programmation des autotests des calculateurs .

Elle doit être effectuée en étudiant et optimisant l'efficacité de la détection des défaillances. Pouvoir vérifier si toutes les défaillances simples peuvent être détectées avec ce programme de test, elles doivent être simulées ou provoquées au niveau du matériel.

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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 (3) 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, ambiguït des défaillances, probabilité d'occurence des défaillances particulières, etc...) d'encore améliorer le système entier.

Documents de référence :

"Propositions d'un système de protection utilisant des calculateurs. Applications au contrôle des réacteurs nucléaires", - Note Technique LETI/MCTE N° 1206, HD.MAIER, P.DARIER.

13. SYSTEMS OPTIMISATION, STANDARDISATION, NEW CONCEPTS

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Classification: 13	
Title 1 (Original Language): Verminderung der Primärkreiskontamination durch Einsatz eines Elektromagnetfilters. Versuche zur prinzipiellen Durchführbarkeit (RS 171 - II.1.6 , Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
Title 2 (English): Reduction of the contamination of primary system components by electromagnetic filters	Project Leader: Dr. Neeb
Initiated (Date): 1. 7. 75 Status: Completed Completed (Date): 30. 9. 76 Last Updating (Date): 31. 12. 76	

1. General Aim and 2. Particular Objectives

The possibilities for reduction of corrosion product activity and contamination by means of electromagnetic filters, which have been used in fossile hated plants have been studied. Some tests were conducted in order to find out, whether electromagnetic filters could be used in the primary system of a PWR.

3. Research Program

a) Tests on the effeciency of electromagnetic filters were conducted, using low concentrations of corrosion products in highly cleaned water at temperatures up to 300 °C

b) Investigation of the influence of boric acid and LiOH with concentrations as used in the coolant system of a PWR

c) Metal release rates of the balls of the electromagnetic filter in boric acid and LiOH-solutions, as far as possible in this experimental facility

d) The possibility of backflushing has been tested with respect to the corrosion product oxides, seperated on the balls of the electromagnetic filter

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4. Experimental Facilities

The experimental studies have been carried out in the Pressurized Water Chemistry Loop (DCK) of KWU-Erlangen. A high-pressure electromagnetic filter has been fabricated and was used in connection with the DCK. Measurements by normal and radio-tracer analytical techniques were done by the KWU laboratories.

5. Progress to Date

The radio-tracer tests of radioactive corrosion products were continued, using added corrosion products from KWO and KWB-A. Samples were taken at the entrance and the outlet of the electromagnetic filter, the Fe content and the distribution of radio-tracers were investigated.

The backwashing of the corrosion products, which were deposited at high temperatures, was tested aeveral times.

The influence of boric acid on the deposition of particles was tested at a boron content of 2000 ppm.

The activity distribution was measured on the electromagnetic balls and the pipe walls when the electromagnetic filter was switched off.

6. Results

The radio-tracer tests needed a long time, because the activity was rather low. The result was that only a rather low deposition rate was recognized.

The presence of boric acid has no influence on the deposition of particles on the electromagnetic filter.

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7. Next Steps

The work has been completed.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

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Classification: 13	
Title 1 (Original Language): **Vergleich verschiedener Druckbehälterkonzepte für Leichtwasserreaktoren** (RS 219 - II.1.6, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Battelle-Institut e.V., Ffm.
Title 2 (English): **Comparison of Different Pressure Vessel Concepts for Light-Water Reactors**	**Project Leader:** **G. Langer**
Initiated (Date): September 1, 1976 Status: Continuing Completed (Date): June 30, 1977 Last Updating (Date): December 31, 1976	

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 will be carried out essentially in two steps.

3.1. Collection of the Necessary Data

The information necessary for comparison will be collected by a literature survey. It will be supplemented by data obtained in interviews and discussions with manufacturers and consultants.

3.2. Technological Assessment of the Individual Concepts

The individual concepts will be assessed by utility value analysis.

4. Experimental Facilities, Computer Codes

5. Progress to Date

The following work was performed in the report period:

- Drawing up of a detailed plan of the report
- Completion of about one fourth of the volume of the research program outlined in Section 3.1
- Performance of the first investigations of Section 3.2

It is planned to include the following concepts in the research program:

- Thick-walled steel pressure vessel
- Multi-layer pressure vessel
- Pressure vessel produced by build-up welding
- Prestressed concrete pressure vessel
- Prestressed cast-iron pressure vessel
- Prestressed cast-steel pressure vessel

6. Results

The comparative criteria used in the study require detailed technical data. Because of the very low level of development, however, these are not available for all the concepts listed above. This applies to the concepts of the pressure vessel produced by build-up welding and the prestressed cast-steel pressure vessel.

7. Next Steps

Continuation of the work according to the research program outlined in Section 3.

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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14. PROBABILISTIC METHODS OF SAFETY ANALYSIS

Classification: 14	
Title 1 (Original Language): Die Berechnung der Zuverlässigkeit großer komplexer Systeme nach der Methode der relevanten Pfade (RS 106 - I.1.8., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: TU Berlin
Title 2 (english): Calculation of Reliability Data for Complex Systems Using the Success Paths Method	Project Leader: Prof. Dr. Memmert
Initiated (Date): 1.4.1973 Status: finished	Completed (Date): 30.9.1975 Last Updating (Date): December 1975

1. General aim

A computer program is to be written to determine reliability data for complex systems with components which

1. are unrepairable,
2. can be repaired immediately if failure occured,
3. can be maintained in periodical intervals,
4. or whose occurrence probability of failure is given by a time independent value.

The results to be obtained are

a) the occurrence probability of the first systemfailure Q(t),
b) the unavailability U(t) and its mean value $\bar{U}$,
c) the mean time of systemfailure as a function of time,
d) the steady state values, mean time to repair MTTR, and mean time between failures MTBF, if the system is in steady state.

2. Particular objectives

The program should employ analytical methods to find the cut-sets of the system andto compute the reliability data of these sets, which constitute first order approximation to the data of the system. It should be possible to use fault-trees and logical block diagrams and to use a cut-off method for large systems (more than 3.000 cut-sets).

3. Experimental facilities

Not relevant.

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4. Project status

4.1 Progress to date

A report on the theoretical background of ARP II was given last year /1/2/3/4/, so that now the application of this analytical reliability program to an example is of interest.
The largely simplified version of a nuclear power plant energy supply is shown in fig. 1 and the corresponding fault-tree in fig.2.

1	: short circuit in the main circuit connection (self-indicating)
2 and 6	: transformer failure (maintenanced)
3 and 4	: switch opens unintentional (self-indicating)
5	: generator or reactor failure (self-indicating)
7	: short circuit in the 6 kV supply (self-indicating)
8 and 9	: switch does not open on demand.

As the top-event (Gate 16) the break-down of the 6 kV supply is defined.
To compute this example with ARP II the following card-deck is required (real values in the format E 10.0 and integer values in the format I 3).

1. card:	9,16,7	max. number of components max. number of components and gates max. number of gate-entrances + 2
2. card:	10,1,1,8	and-gate 10(1) with entrances 1 and 8
3. card:	11,1,5,9	and-gate 11(1) with entrances 5 and 9
4. card:	12,0,1,2	or-gate 12(0) with entrances 1 and 2
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17. card:	16,0,10,11,15,6,7	or-gate 16(o) with entrances 10,11,15,6 and 7
18. card:	100	the computation is carried out using a hundred time steps

19. card: 100.0 time step length is a hundred hours
20. card: $1.0 \cdot 10^5$ mean time to failure of component 1

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29. card: $1.0 \cdot 10^4$ mean time to failure of component 9
30. card: 100.0 (first column) mean time to repair of component 1

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39. card: (second column) 500.0 maintenance period of component 9

4.2 Essential results

The results are presented in fig. 3. 2,5 sec of computing time were required on the CDC 6500 of the TU-Berlin.

5. Next steps

Work was finished at Sept. 1975.

6. Relation with other projects

At present no other work is being done on reliability problems in connection with the reactor safety program of the BMFT. The author's present report is related to the following analytical computer programs:

ARMM-69 and GAMM (both programs use the theorem of Bayes as a basic method),
SICHERHEIT (utilizing the Truth-Table-Method), and
SAP (semi analytical with repair and inspection).

7. Reference documents

1 Quarterly reports in the series IRS-Forschungsberichte

2 Richter, G. and Memmert, G.: Berechnungen von Zuverlässigkeitsdaten komplexer Systeme mit analytischen Methoden. TUBIK 28, Report of the Institut für Kerntechnik, TU-Berlin (Oct. 1973)

3 Richter, G.: Die Berechnung der Zuverlässigkeit großer komplexer Systeme nach der Methode der relevanten Pfade. (Dissertation am Institut für Kerntechnik, TU-Berlin, Febr.1975)

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4 Kamarinopoulos, L. and Richter, G.: Vergleichende Untersuchungen verschiedener Methoden zur Berechnung der Ausfallwahrscheinlichkeit komplexer Systeme (Atomkernenergie 26/2),

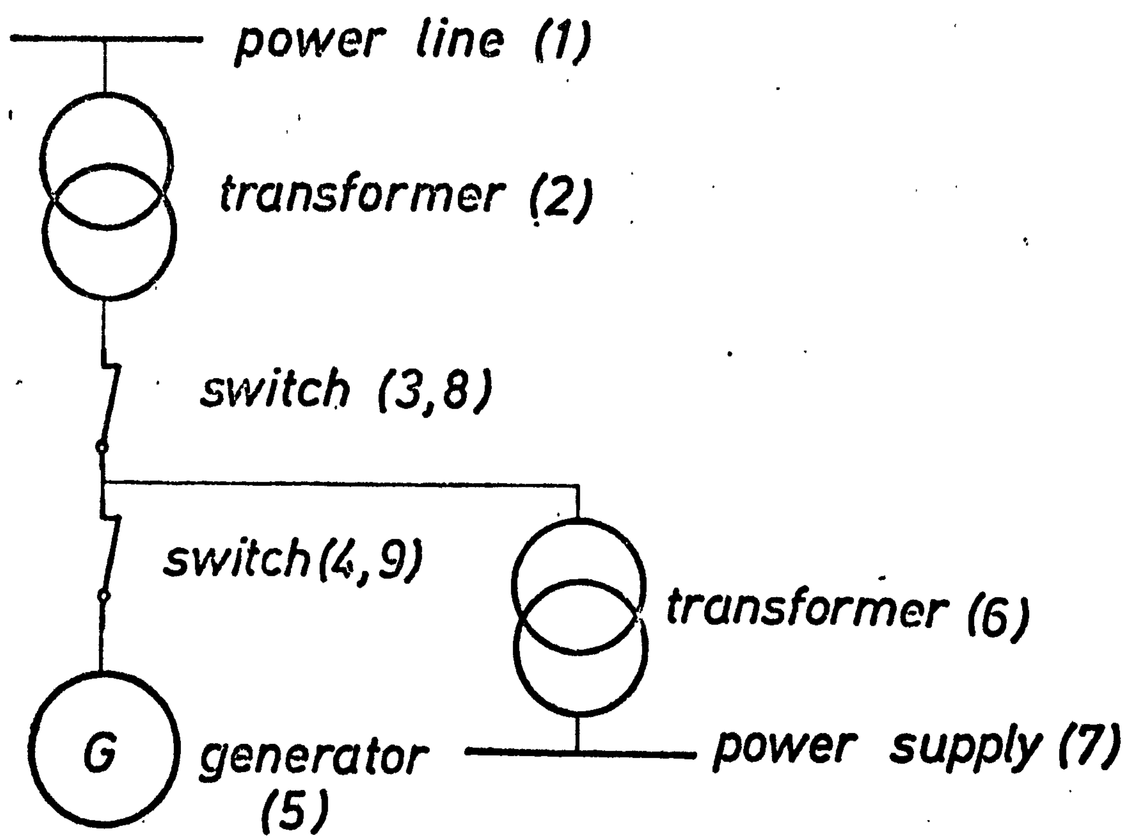

Figure 1

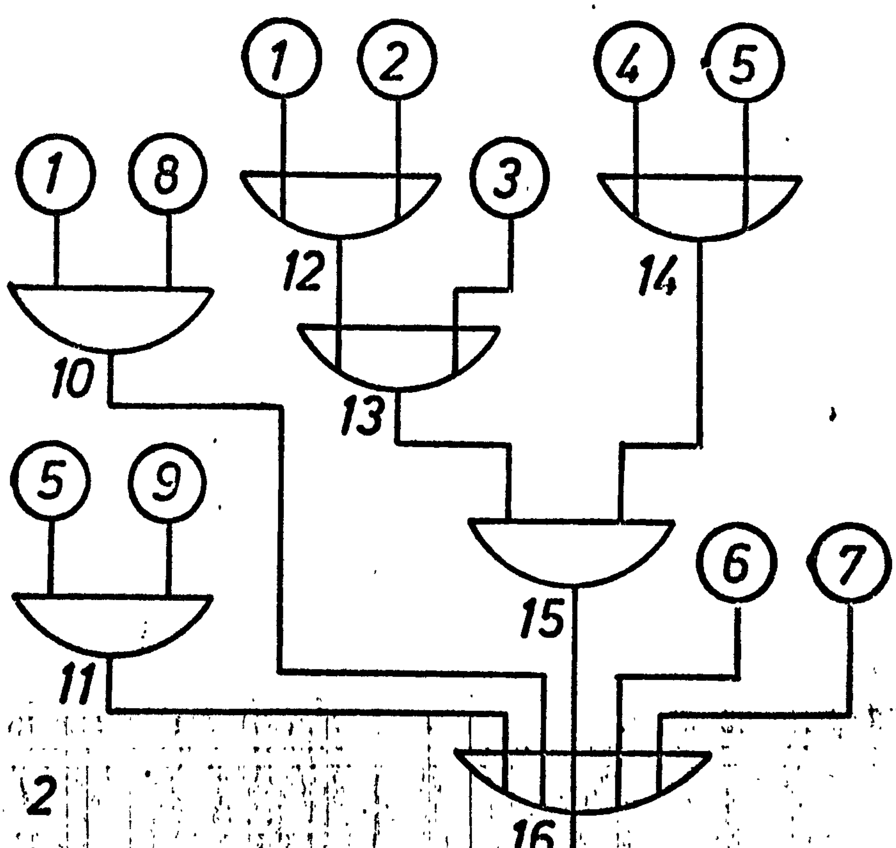

Figure 2

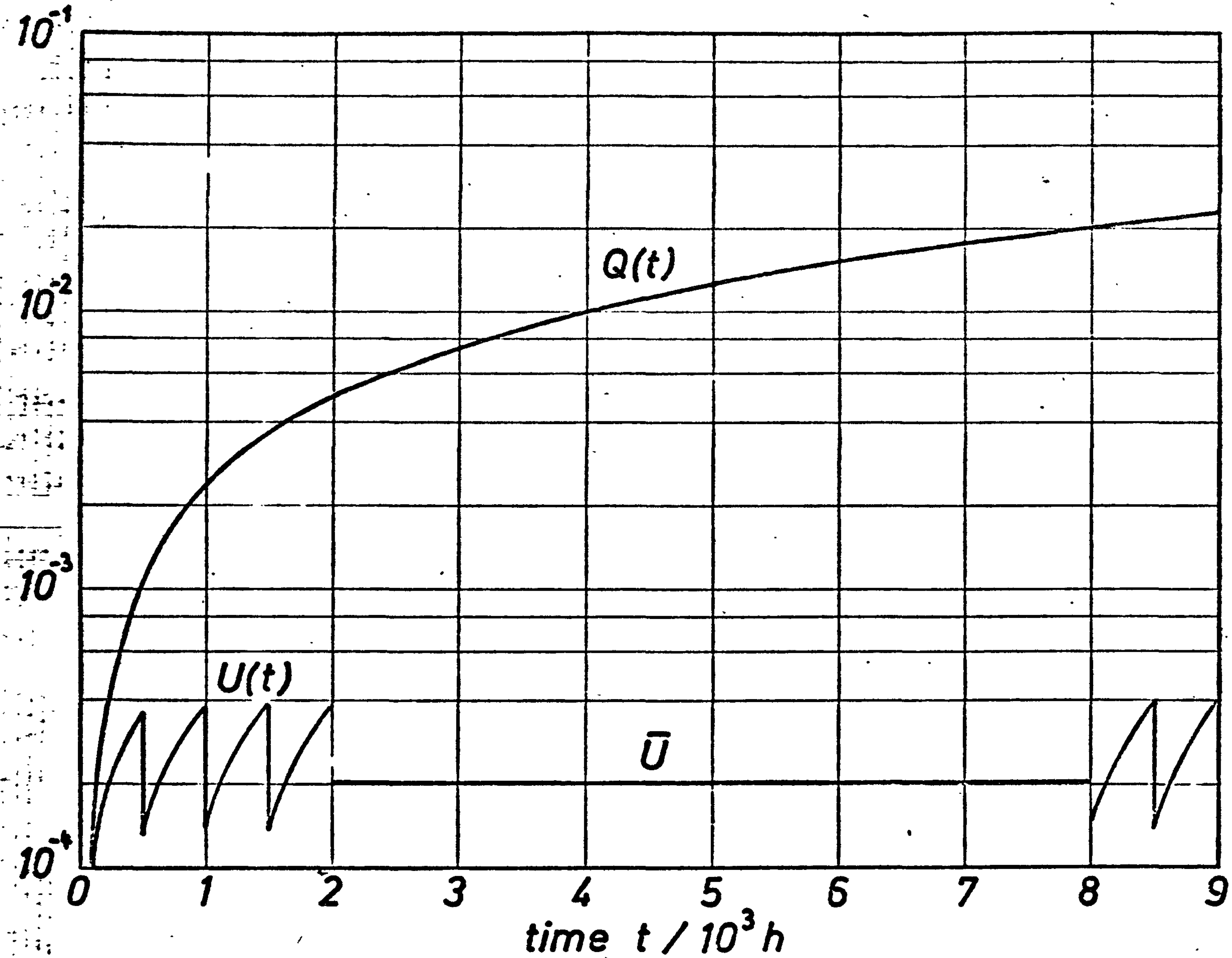

Figure 3

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Classification: 14	
Title 1 (Original Language): Sicherheits- und Zuverlässigkeitsanalysen kerntechnischer Anlagen (Teil 2) (RS 134 - I.1.8, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Ingenieurbüro Buck
Title 2 (English): Safety and Reliability Analysis of Nuclear Plants (Part 2)	Project Leader: Dr. W. Buck
Initiated (Date): 1 October 1975 Status: Completed Completed (Date): 3o September 1976 Last Updating (Date): January 1977	

1. General Aim

This research program concerns the safety and reliability of nuclear plants. Aims include the development and application of a new method to calculate the influence of physical and chemical events, during an accident or off-normal condition, on the reliability parameters of systems.

2. Particular Objectives

The new method comprises a technique to analyze and calculate:

- events leading to an accident or off-normal condition,
- events during and following the accident or off-normal condition.

The resulting reliability parameters for the short-term range investigated should be much more accurate than those calculated by the ordinary method.

In the LWR field the method should be particularly applicable to:

- Loss-of-coolant accidents,
- Anticipated transients without scram,

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- Reactivity-initiated accidents.

The first application was: Reactivity-initiated accidents in a BWR/6 in the hot standby condition.

3. Research Program

The research program comprised:

- Investigation of critical events,
- Development of a method (including a computer program) to perform a probabilistic analysis. In this analysis the components of systems may not only fail completely but also partially,
- Development of a dynamic model (including a computer program). The results of the probabilistic analysis are the input to this model.

4. Progress to Date

The development of the general method was completed. As a first application, reactivity accidents in a BWR/6 in the hot standby condition were investigated.

The critical event leading to these accidents is characterized by the following assumptions:

- A control rod is not coupled to the control rod drive,
- This rod is stuck in position,
- The operator intends to withdraw it.

Applying a sensitivity analysis, it was found that the following parameters had to be taken into account for the dynamic model:

- Reactivity worth of the control rod,
- Mean burn-up of the reactor,
- Rod-drop velocity.

5. Results

The resulting failure probabilities are:

1. Probability for cladding failure: $2.8 \cdot 10^{-4}$.
2. Probability for UO_2-melting: $5.6 \cdot 10^{-5}$.
3. Probability for prompt fuel dispersion: $3 \cdot 10^{-6}$.

The analysis allows the following conclusions:

- The new method developed and applied here yields very differentiated results,
- These results are much more conservative than those calculated by the ordinary method, and they are much more accurate,
- The method may also be very suitable for use in investigations of loss-of-coolant accidents and anticipated transients without scram.

6. References

During 1976 the following reports were edited concerning this research project:

1) General Description of a Method to Perform Reliability-Investigations in Nuclear Reactors in the Short-Term Range Following an Accident or Off-normal Condition. Ingenieurbüro Buck, 23-2/8.2. (1976).

2) SHOTER, a Model to Calculate the Failure Probability of Nuclear Reactors in the Short-Term Range Following an Accident or Off-normal Condition. Ingenieurbüro Buck, 23-3/8.2. (1976).

3) Reliability-Investigations in the Short-Term Range in BWR/6 Following an Reactivity-Initiated Accident. Ingenieurbüro Buck, 23-4/8.2. (1976).

7. Degree of Availability of the Reports

The reports are written in German. They may be available from Der Bundesminister für Forschung und Technologie, Postfach 12 03 70, D-5300 Bonn 12, W.Germany.

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Classification: 14	
Title 1 (Original Language): Prozeßmodelle und Reaktorregelung (ATT 085 A - I.1.8, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMI **ORGANIZATION:** LRA, Garching
Title 2 (English): Process Models and Reactor Control	**Project Leader:** i.V. Dr. D. Wach Dr. A. Höld
Initiated (Date): **Completed (Date):** **Status:** continuing **Last Updating (Date):** December 1976	

1. General Aim

Development of theoretical process models in order to describe the dynamic behaviour of different types of power reactors (together with their conventional parts).

Application of modern control theory, as developed in recent years for complex multiple input/output systems, to on-line control of large nuclear power plants by process computers.

2. Particular Objectives

Since several years the institute is occupied with the development of theoretical process models (and corresponding digital codes) which are able to describe either the frequency response and stability or the transient behaviour of different types of nuclear power reactors (BWRs, PWRs, sodium cooled reactors) and their conventional parts (heat exchangers, steam generators).

In addition, efforts have been made to apply the optimal control theory to nuclear power plants. The work covered two main objectives, the simu-

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lation of an optimal core power distribution control loop and the development of an optimal control for a BWR nuclear power plant.

3. Experimental Facilities and Program

Not relevant.

4. Project Status

4.1 Progress to Date

a) Frequency response models:

No further development work has been done on this subject.

b) Nonlinear transient model for U-tube steam generators:

Within the scope of establishing a nonlinear transient model for the description of the dynamic behaviour of a PWR nuclear power plant after large excursions (see also project "Dynamics of Large Water Reactors") the derivation of an extensive transient model for U-tube steam generators at natural circulation conditions has been one of the main tasks during the period before and also during this period.

Starting from the corresponding mass, energy, momentum and/or volume balance equations the theoretical model has been derived under the following assumptions: The heat exchanging zone of the steam generator has been taken to be represented by a number of identical U-tubes. The primary coolant flows on the inner (having a co-current and a counter-current branch), the secondary on the outer side of the tube. The coolant channels can axially be divided into max. 7, the tube wall radially into max. 3 nodes. The steam dome has been characterized by its parameters: system pressure, water volume and water level. The steam removal system takes into account a steam mass flow perturbation which can be caused by isolation-, safety-, bypass-, turbine-trip and/or turbine-control valves, but also by feedbacks from the steam turbine. It has been considered that a temperature perturbation at the entrance of the downcomer (being either caused by a change in system pressure and thus saturation temperature and in feed water temperature or mass flow) propagates through the downcomer thus entering the heat exchanging tubes only after a certain delay time. The resulting mass flow from natural circulation has been calculated by solving the momentum balance equations along the whole recirculation line.

Based on this theoretical model (in general consisting of a system of maximal 91 ordinary nonlinear differential equations and a number of state equations) the digital code UTSG (U-tube steam generator) could be established. The required thermodynamic properties of water and steam will be obtained by using appropriate approximation equations. To calculate heat transfer coefficients an own generally applicable digital code TUMHTC for nuclear steam generators has been developed and inserted into the code UTSG. The slip behaviour will be determined from a formula approximating the Marchaterre-Hoglund correlation, the one-phase flow friction by using in a similar way the Moody correlation, the two-phase flow multiplication friction factor by the corresponding Martinelli-Nelson correlation. The resulting system of ordinary differential equations had been solved by means of the subroutine DIFSYS, an integration procedure based on the Bulirsch-Stoer method.

c) Core power distribution control:

The program CORECON controlling the power distribution in the core, coupled to the core simulator QUABOX, has been used to assess the safety of the control system under disturbed conditions. The disturbance considered was a stuck rod indicent. This specific distrubance was chosen since such an incident occured in the HALDEN reactor, when the power distribution control was in operation.

The simulation runs with stuck rods show that an unstable control behaviour may arise. Thus, additional means are necessary to detect such a failure and to provide the control system with the relevant information, leading to a reduced but safe operational mode of the control.

Other disturbances have not been investigated, since an improved control method has been announced at the Halden Reactor Project, which will be available in spring 1977.

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d) BWR nuclear power plant simulator (digital code LIMBO):

To test modern plant control methods the real power plant has to be simulated by a corresponding theoretical model which is small enough to cause not a too heavy computational effort but still sufficiently accurate. Thus, for simulating a BWR nuclear power plant, a lumped parameter nonlinear transient model and its computer code LIMBO ("lumped parameter simulation model for BWRs") has been developed and - by adding artificial noise - enlarged to the code NIMBO.

On account of computational reasons the optimal control algorithm has to be based on a very much simpler plant model, containing 5 ordinary differential equations. Thus linearizing and simplifying the above discussed BWR plant simulation model yields the code SIMBO, small enough to be applicable within the modern control methods.

e) Model building and parameter identification of a BWR nuclear power plant for use in optimal control:

A comprehensive study of two off-line identification procedures is given in /2/.

A model suitable for control purposes was derived by simplification of the equations used for the plant simulation (see 4.1 d).

For purposes of load following in a large operating range (30 % - 100 %) of the nominal power), such a simple linear model is no longer useful, since its validity is restricted to small deviations around the operating point. To adjust the model to the actual operating point during load following, on-line parameter identification of the model will be adopted. One specific identification algorithm for multivariable systems has been tested and has given satisfactory results. It will be included into the optimal control of a BWR power plant.

f) Optimal control of a BWR nuclear power plant:

The programs developed for optimal filtering and control have been coupled to the plant simulation LIMBO, resulting in a code named OMAR. For steady-state control as well as for load demands up to ± 15 %, the control provides smooth and fast operation of the plant /3; Compact for VDI/VDE-GMR-Fachtagung "Prozeßmodelle", April 1977/. The possibility of weighting state and input velocities provides easy means of achieving a desired control behaviour, e.g. slow variations of average fuel temperature and thermal power, without disturbing the desired new steady-state. Furthermore, hard constraints may be posed on states and inputs, thus contributing to keep the plant in its safe operating region.

4.2 Essential Results

a)
The combination of the codes ADYPMO and FRETI has been applied to calculate the frequency response and transient behaviour of the PWR nuclear power plant GKN, the codes ADYMSOR and FRETI to calculate the frequency response behaviour (and to compare it with measurements) and the transient behaviour of the compact, sodium cooled reactor power plant KNK-I.

b)
The digital code UTSG has been widely tested. A series of calculations (simulating perturbations on the primary and secondary side of the steam generator) has been undertaken and showed good results (see e.g. /1/).

c)
The simulation of a control system disturbance (stuck rod) has shown the necessity of investigating possible system failures and developing counteracting measures to ensure safe operation.

d)
Measurements within the start-up test period at the BWR nuclear power plant WÜRGASSEN (a test with a turbine trip without opening the bypass valve, and a test simulating a main coolant pump failure) have been recalculated with the code LIMBO and showed very good agreement.

e)
An on-line parameter identification procedure has been tested and found suitable for control purposes.

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f)
The optimal control was shown to give very satisfactory results when operating on a simulation of a BWR nuclear power plant. Control features include the weighting of state and input velocities, as well as hard constraints on states and inputs.

5. Next Steps

a)
No further work on frequency response models has been done.

b)
The development work on the nonlinear nodal natural-circulation U-tube steam generator model and on the corresponding digital code UTSG is finished. A publication of the theoretical model and the program description of UTSG is in preparation.

c)
An improved control method will be taken over from the Halden Reactor Project. After coupling this new control to the core simulator QUABOX, three types of problems will be investigated: operation under undisturbed conditions, failures of control system components and reactions of the control system to external disturbances.

d)
The main development work on LIMBO, NIMBO and SIMBO has been done. If necessary, smaller improvements on the codes have to be undertaken. A description of the theoretical background to these models is in preparation.

e)
Off-line identification will be improved by use of continuous Enlarged Kalman Filters.

f)
The on-line identification procedure will be included into the existing control to yield an adaptive control loop.

6. Relation with Other Projects

Co-operation with KWU Erlangen and the OECD Halden Project on behalf of optimal power distribution.

7. Reference Documents

/1/
A. Höld
Nichtlineare Simulationsmodelle niederer Ordnung im Rahmen eines digitalen Regelungskonzeptes für SWR-Kernenergieanlagen
Compacts of the Reaktortagung, Düsseldorf, März/April 1976

/2/
K. Volf
Parameter Identification of a BWR Nuclear Power Plant Model for Use in Optimal Control
MRR 155, February 1976

/3/
D. Beraha
Model Building Identification and Control of a BWR Nuclear Power Plant
IAEA/NPPCI Specialists' Meeting on Use of Computers for Protection Systems and Automatic Control, Neuherberg/München, May 1976
MRR 160, July 1976

/4/
Quarterly reports in the series IRS-Forschungsberichte.

8. Degree of Availability

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

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Classification: 14	
Title 1 (Original Language): System- und Zuverlässigkeitsanalyse an Leichtwasserreaktoren (ATT 085 A - I.1.8., Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (English): System and Reliability Analysis on Light Water Reactors	Project Leader: i.V. Dr. D. Wach Dr. Hörtner Dr. Kafka
Initiated (Date): / Completed (Date): Status: continuing / Last Updating (Date): December 1976	

1. General Aim

The detailed investigation of possible accident sequences, especially of their probability and the implementation of the results in a risk concept.

2. Particular Objectives

- Detailed investigation of typical 1200 MWe-PWR-plants
- Set-up of cause-consequence diagrams in order to find out the most important accident sequences
- Detailed analysis of the engineered safety features installed to cope with the associated accidents and estimation of their reliabilities
- Systematic investigation of important electronic equipments with the aim of finding the failure rates of the relevant failure modes.

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3. Experimental Facilities and Program

Not relevant.

4. Project Status

4.1 Progress to Date

For the nuclear power plants considered, the reliability investigation /1/ of the engineered safety features needed to control the design basis accident "Large LOCA" were finished. Very detailed sensitivity studies and discussions of the results were done, including common-mode failures The reliability analysis of the reactor control rod scram systems for different transients was also completed by a discussion of common-mode failures. A reliability analysis was carried out for the reactor coolant system safety and relief valves needed to cope with the ATWS "Loss of Normal Power".

The experiences gained by the investigations of the engineered safety features control system were summarized.

For different electronic modules investigations were performed in order to find out the various failure modes and to evaluate the associated failure rates.

The computer programs to calculate the reliability data for complex systems were further developed and are described in special reports /2, 10, 12/.

A report with critical comments on the Reactor Safety Study WASH-1400 was finished /6/.

4.2 Essential Results

The results of the studies for the engineered safety features were discussed in papers /5,7,12,17,2o,21/. From these results it can be seen that very detailed analyses of the systems are necessary. Only

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by this means all the significant contributions to the unavailability can be detected, e.g. contributions through common-mode failures caused by system intermeshings. Examinations of the entire system needed to cope with the considered accident are therefore required.

The results of the reliability investigations for electronic devices were described in /4,8,1o/. These reports contained the detailed tabulations of the failure mode and effect analyses performed. The intermediate and final results for the different failure modes were shown.

5. Next Steps

The project on System analysis of the engineered safety features of a German PWR plant has been finished. The experiences gained during this work will now be used for the risk study of a German PWR plant (Project RS 215 and RS 217).

6. Relation with Other Projects

Blowdown and refilling calculations for LOCAs as well as investigations concerning the dynamic behaviour of plant in the case of transient events are done within other sections of the LRA. These investigations are taken as the basis for the reliability analysis.

7. Reference Documents

/1/
H. Hörtner, E. Dressler, E. Nieckau, H. Spindler
Kernkraftwerk Biblis, Block A. Zuverlässigkeitsuntersuchung der für die Beherrschung des Auslegungsstörfalls "Bruch einer kalten Hauptkühlmittelleitung" erforderlichen Sicherheitssysteme, Teil I. Vertraulicher Bericht
MRR-V-9, November 1975

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/2/
E. Dressler, H. Lurz
SAFTL und CRESS. Beschreibung zweier Programmsysteme zur Berechnung der Zuverlässigkeit von komplexen Systemen. Programmbeschreibung
MRR-P-21, Dezember 1975

/3/
K. Singh-Wadwa
Perspectives on the Risks of Nuclear Power Production - A Literature Review. Internal Report
MRR-I-62, Januar 1976

/4/
S. Goßner, S. Steindl
Experimentelle Ausfalleffektanalyse an dem Binärverteiler ZL 412 (S&F)
Vertraulicher Bericht
MRR-V-13, Februar 1976

/5/
H. Hörtner, E. Dressler, E. Nieckau, H. Spindler
Kernkraftwerk Biblis, Block A. Zuverlässigkeitsuntersuchung der für die Beherrschung des Auslegungsstörfalls "Bruch einer kalten Hauptkühlmittelleitung" erforderlichen Sicherheitssysteme, Teil II. Vertraulicher Bericht
MRR-V-14, April 1976

/6/
H.P. Balfanz, P. Kafka
Kritischer Bericht zur Reaktorsicherheitsstudie (WASH 1400). Interner Bericht
MRR-I-65, IRS-I-87, April 1976

/7/
H. Hörtner, E. Nieckau
Kernkraftwerk Biblis, Block A. Zuverlässigkeitsuntersuchung der Systeme zur Druckentlastung des Reaktorkühlkreislaufs für den "Notstromfall".
Vertraulicher Bericht
MRR-V-15, Mai 1976

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/8/
S. Goßner, S. Steindl
Experimentelle Ausfalleffektanalyse an dem Rechengerät TZA 20 (H&B).
Vertraulicher Bericht
MRR-V-16, Mai 1976

/9/
W. Bastl
CSNI Task Force on Problems of Rare Events in the Reliability Analysis of Nuclear Power Plants
Ispra, Jund 1976

/10/
S. Goßner
Experimentelle Ausfalleffektanalyse an dem CMR-Meßumformer ETU 120 (H&B)
Vertraulicher Bericht
MRR-V-17, Juni 1976

/11/
R. Daugherty, L. Schlösser
CRESSEX - Beschreibung eines Zuverlässigkeitsrechenprogramms zur Ermittlung wichtiger Kenngrößen von komplexen Systemen. Programmbeschreibung
MRR-P-23, September 1976

/12/
W. Ullrich, W. Frisch
Untersuchungen von Betriebsstörungen bei Versagen der Reaktorschnellabschaltung (ATWS) und anderer ausgewählter Sicherheitseinrichtungen
MRR 163, IRS-W-22, September 1976

/13/
E. Dressler
Theoretische Grundlagen zum Programmsystem SAFTL und CRESS zur Berechnung der Zuverlässigkeit von Systemen
MRR 164, September 1976

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/14/
J. v. Linden
Kernkraftwerk Biblis, Block A. Untersuchung der Kühlmittelverluste im Nuklearen Zwischenkühlkreis für den Auslegungsstörfall "Bruch einer Hauptkühlmittelleitung". Vertraulicher Bericht
MRR-V-19, Oktober 1976

/15/
P. Kafka
Reactor Safety Systems
Lecture at IAEA Course, Karlsruhe, 11.11.1976

/16/
E. Nieckau
Schlußfolgerungen aus Zuverlässigkeitsanalysen von Reaktorschutzsystemen
MRR 166, November 1976

/17/
W. Dietlmeier
Kernkraftwerk Biblis, Block A. Fehlerbaumanalyse der Notspeise-Pumpen-Turbine. Vertraulicher Bericht
MRR-V-20, November 1976

/18/
H. Hörtner
Stand von Störfall- und Zuverlässigkeitsanalysen für Kernkraftwerke
KTG-Fachseminar "Störfallverhalten von Leistungsreaktoren", Jülich, 23. - 26.11.76.
MRR-I-77, Interner Bericht, Dezember 1976

/19/
P. Kafka, E. Münch
Risikokonzept für kerntechnische Anlagen, eine Einführung
KTG-Fachseminar "Störfallverhalten von Leistungsreaktoren", Jülich, 23. - 26.11.76
MRR-I-80, Interner Bericht, Dezember 1976

/20/
H. Hörtner, E. Nieckau, H. Spindler
Kernkraftwerk Biblis, Block A. Ergebnisse der Zuverlässigkeitsuntersuchung für den Auslegungsstörfall "Bruch einer kalten Hauptkühlmittelleitung"
MRR 168, Dezember 1976

/21/
S. Goßner, J. v. Linden
Kernkraftwerk Biblis. Zuverlässigkeitsuntersuchung des Gesamtsystems zur Reaktorschnellabschaltung. Vertraulicher Bericht
MRR-V-21, Dezember 1976

/22/
Quaterly Reports in the Series IRS-Forschungsberichte

8. Degree of Availability

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

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

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Classification: 14	
Title 1 (Original Language): Vergleich von Rechenprogrammen zur Zuverlässigkeits-analyse von Kernkraftwerken RS 172 - I.1.8., Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: IRS, Köln
Title 2 (English): Comparison of Calculation Programs for the Reliability of Nuclear Plants	Project Leader: W. Otto
Initiated (Date): 25.8.1975 Status: finished Completed (Date): 15.2.1976 Last Updating (Date): December 1976	

1. General Aim

The aim of the project is the comparison of the efficiency of existing codes which are used to evaluate complicated fault trees. The comparison and the assessment of the methodical status reached by the programs, shall serve as basis of decision for future research in fault tree analysis within the research project "Risk and Reliability" which is sponsored by the Federal Ministry of Research and Technology in the framework of the Reactor Safety Research Program.

2. Particular Objectives

Analytical and simulative programs of various institutions are to be checked, in calculating several fault trees, which were worked in advance by three institutions (1. Institut für Reaktorsicherheit - IRS, 2. Laboratorium für Reaktorregelung und Anlagensicherung - LRA, 3. Gesellschaft für Kernforschung - GfK) which themselves took part in the calculations.

3.1 Experimental Facilities

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3.2 Research Program

The fault trees having been set as the task considered e.g. repair work,

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inspection and switching processes by multiphase calculations. A short description of the fault trees is contained on table 1.

4. Project Status

4.1 Progress to Date

Following institutions and project leaders took part in the project in calculating the fault trees completely or partially:

Ingenieurbüro Buck, Lemförde	(Dr. Buck)
Gesellschaft für Kernforschung, Karlsruhe - GfK -	(Dr. Caldarola)
Industrieanlagen-Betriebsgesellschaft, Ottobrunn - IABG -	(Dr. Keller)
Interatom, Bensberg	(Dr. Rosenhauer)
Institut für Reaktorsicherheit, Köln - IRS -	(Dr. Heuser)
Euratom, Ispra	(J. Amesz)
Laboratorium für Reaktorregelung und Anlagensicherung, Garching - LRA -	(Dr. Kafka)
Messerschmitt, Bölkow u. Blohm, München - MBB -	(Fiedler)
Technische Universität Berlin, Institut für Kerntechnik	(Prof. Memmert)
TÜV Rheinland, Köln	(K.R. Hartung)

The results received by the various institutions and the programs used were judged by IRS /1/.

4.2 Essential Results

Various computations performed for the given tasks of fault tree analysis (Fault Trees 1 - 5) have shown that for the purpose of system reliability analysis in the Federal Republic of Germany several effective computer programs just were available for application. Referring to the present status of system analysis methods (component-specific system analysis without handling partial failure) further promotion of research within this field is not recommended.

A general research program for further methodological development (e.g. description of time-dependent interrelationships, physical parameters and short-term effects) should only be proposed until first experience and results, derived from the work on the German risk study, are available. Besides this, however, limited promotion of some special research projects, related to well-defined subtasks, (e.g. transient analysis) is considered worthwile.

5. Next steps

The project is finished.

6. Relation with Other Projects

RS 106 Calculation of reliability data for complex systems using the success path method

RS 134 Safety and reliability analysis of nuclear plants

7. Reference Documents

/1/ Final Report RS 172 Comparison of Calculation Programs for the Reliability of Nuclear Power Plants (German)

Quarterly reports in the series IRS-Forschungsberichte.

Report period	July - Sept. 1975	IRS - F - 27	(German)
	Oct. - Dec. 1975	IRS - F - 28	(German)
	Jan. - March 1976	IRS - F - 30	(German)
	A 75	IRS - F - 29	(English)

8. Degree of Availability

Documents are available through GRS, D-5000 Köln 1, Federal Republic of Germany.

Table 1: Description of Fault Trees

FT	General Problem	Concrete Example	Asked Values
1	Calculation of reliability and availability taking into account inspection and repair	System of components with constant failure rates, regularily inspected, repair times small as compared to duration of time for detection of failures	Reliability and availability of system, mean elapse of time until detection of system failure
2	Testing of simulation routines (changes between service states)	Failure behaviour of an electrical energy supply for a nuclear plant	System reliability and availability, contributions of subsystems to the failure behaviour of the total system
3	Differentiation between standby and operating state	Emergency power supply	Reliability and availability of the emergency power distribution
4A 4B	Reliability analysis of a safety system	Emergency energic system	4A: Mean unreliability of the safety system 4B: Failure Probability in the after cooling phase most important failure combinations
5	Simulation of an investigation from the Rasmussen report	Evaluation of a High-Pressure Injection System of a PWR	Minimal cut sets, unreliability of the system

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Classification: 14	
Title 1 (Original Language): Risiko- und Zuverlässigkeitsanalyse von Kernkraftwerken (PNS 4530 - I.1.8., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (english): Risk and Reliability Analysis of Nuclear Power Plants	Project Leader: L. Caldarola
nitiated (Date): 1.1.1974 Status: continuing	Completed (Date): Last Updating (Date): December 1975

1. General Aim

General aim of the project is the reliability analysis of systems with respect to nuclear power plants.

2. Particular Objectives

The objective of the program is to develop an analytical computer program as well as a Monte-Carlo-program for fault tree evaluation and to develop methods for the automatic construction of fault trees.

3. Experimental Facilities and Research Program

The project is a mathematical and analytical one. Therefore no experimental facilities are needed. Each program step depends on the results of the step before. (See Essential Results and Next Steps).

4. Project Status

4.1/2 Progress to Date and Essential Results

A computer program has been written and tested. Main features of the program are the following

- Four different types (classes) of components can be handled
  - a) Irreparable components
  - b) Reparable components, with failures which are immediately detected (revealed faults)
  - c) Reparable components, with failures which are detected upon inspection (unrevealed faults)

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d) Components characterized by a constant unavailability

- Capability to identify all minimal cut sets in order of importance
- Capability to analyse systems characterized by two phases on following the other in time

- Compatibility test allows one to find out if the two fault trees are logically compatible.

- The following quantities can be calculated as functions of time
  a) System point unavailability
  b) System average unavailability (unavailability averaged over the time)
  c) System failure intensity
  d) System average failure intensity (failure intensity averaged over the time)
  e) System integral of failure intensity (in the present from of the program this quantity is used as system unreliability)

- The system average failure rate (1/MTTF) can also be calculated with all components intact at the initial state /2/.

The computer program needs 480 K in CPU. This allows to analyse fault trees either with a maximum of 256 elements and 200 points on each time axis or with a maximum of 2000 elements and no calculation on the time axis (average and maximum values only).

A new and more sophisticated theory /1-2/ to calculate the unreliability of complex repairable systems has been developed. The method is based on a set of integral equations each one referring to a specific minimal cut set of the system. Each integral equation links the unavailability of a minimal cut set to its failure probability density distribution and to the probability that the minimal cut set is down at time "t" under the conditions that it was down at time "t'" ($t' \leq t$).

Three test problems of the "BMFT Leistungsprüfung" were solved with the Karlsruhe computer program. The results are shown in /3/.

5. Next Steps

- A fifth class of components will be included in the computer program. This will include components which are inspected and repaired when they are demanded to operate (unrevealed faults).

- The theory under section 4 will be incorporated in the computer program.
- The feature to handle systems characterized by many operating states will be built in the program.
- The feature to handle correlated faults (among various components) will be also built in the program.

6. Relation with Other Projects

No direct relation but cooperation.

7. Reference Documents

/1/ L. Caldarola
"A method for the calculation of the cumulative failure probability distribution of complex repairable systems"
(being published in "Nuclear Engineering and Design")

/2/ L. Caldarola,
"Calculation of the mean time to failure of a redundant repairable system"
Bericht Nr. IRE/1/4530/17/75

/3/ L.Caldarola and A. Wickenhäuser,
"BMFT-Leistungsprüfung - Vergleich von Rechenprogrammen zur Zuverlässigkeitsanalyse von Kernkraftwerken, RS 172 - Abschlußbericht" (in English)
Bericht Nr. IRE/1/4530/19/75
PNS 59/75

8. Degree of Availability

/2/, /3/, Internal Reports

1599

Classification: 14	
Title 1 (Original Language): Rechenprogramm für Zuverlässigkeit und Verfügbarkeit von Gesamtkernkraftwerken mit Hilfe der Zustandsanalyse (RS 189-I.1.8, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: INTERATOM
Title 2 (English): Computer Program for Reliability and Availability of Nuclear Power Plants using the Method of State Analysis	**Project Leader:** Dr. H. Zeibig
Initiated (Date): 01.01.76 **Status:** Continuing **Completed (Date):** 31.12.78 **Last Updating (Date):** December 1976	

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

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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

After receiving the contract in June 1976 (Date initiated 01.01.76) work was started in September 1976 because of priority for other projects.
The first step was the review of documents and codes concerning the state analysis available at INTERATOM.

(3.1) A first model with not more than twelve groups of states was constructed in order to have an example at hand for the development of codes. This preliminary model includes the case which is relevant for technical applications, that states exist for which the Markov property is not valid.

(3.2) The code MARKOV available at INTERATOM is not appropriate to handle state models in which the Markov property is not valid. The rough model of 3.1 showed, however, that a realistic description of technical systems has to include such Non-markovian transitions. Principal solutions of the problem are well known. Thus the development of a computer code was started which is able to handle general state models. 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

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7 Next Steps

(3.1) The elaboration of the preliminary state model will be continued by detailed analysis and creation of substates.

(3.2) The further development and testing of the code for general state models will be continued.

8 Relation with Other Projects

9 References

Quarterly reports in the series IRS- and GRS-Forschungsberichte
July - Sept. 76 IRS - F - 33; Okt. - Dez. 76 GRS - F - 35

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10 Degree of Availability of the Reports

1603

Classification: 14	
Title 1 (Original Language): Zuverlässigkeitsbeurteilung für den Sicherheitseinschluß (SE) am Beispiel des Druckwasserreaktors (RS 201 - I.1.8, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Tech.Univ.Munich
Title 2 (English): Reliability Assessment of the Secondary Containment of a PWR	Project Leader: Prof.H. Kupfer
Initiated (Date): June 1, 1976 Status: Continuing Completed (Date): May 31, 1978 Last Updating (Date): Dec. 1976	

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

1604

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.

3.4 Experimental facilities to be developed during the course of the research will simulate impact load conditions which may have to be expected to act on the full scale structure. Test series are to be carried out using plane and reinforced concrete samples.

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. Progress to Date

(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 leack 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 all systems, the reactor protection system, etc. are checked during the failure event sequence with respect to their functioning or non functioning. It is therefore intended to determine for each possible failure sequence the resulting survival probability. For calculating the pressure values with the Program ZOCO the following input data have been evaluated:

- the internal structures which are located in the containment as different numbers of pressure rooms, for determining the influence of these partitions on the maximum pressure, the occurrence time of the maximum pressure and the computational time
- the flow from the primary system into the containment within a short interval during different failure sequences using the code BRUCH-D-05.

3.4 The results gained at the conference in Boston (Second International Conference on the Mechanical Behaviour of Materials) have been classified and evaluated.

In the course of the computation of the steelhull a sensitivity analysis has been performed. This calculation shows the influence of the different material properties and the influence of some internal loading conditions on the probability of failure of the containment on the basis of a simple analysis. Furthermore data of acceptance tests of special containment steels have been statistically evaluated.

3.5 For calculating the load effects of the steel hull with the finite element program SAP, the whole containment was divided up into 250 thin shell elements. Two of the largest perforations, the material- and the staff lock have been taken into account. In order to simulate numerically the impact loading process - which is to be determined as described above under item 3.4 - shell model consisting axi-symmetric elements has been performed. For this purpose the load effects as obtained from the dynamic analysis of the global system are used as boundary conditions.

Since for the time being no reliable values and information concerning the type of distributions and their parameters for loads and strength are available, only estimations of the order of magnitude of the failure probability of the steel hull given a LOCA has been calculated so far. This was done using normal distributions.

6. Results

(With reference to item 3.: Research Program)

3.1 A complete numerical analysis of the failure event sequence as described above has not been performed as yet. However for the purpose of performing preliminary stress calculations a maximum value of 4.7 bar has been estimated. As a first estimate, the failure of the emergency cooling system has been determined to be 5.10^{-4}/usage. The treatment of the problem in a more differentiated manner is carried out at the present.

3.4 The evaluation of the results reported at the conference in Boston indicates in the case of high rates of loading on unreinforced concrete a decrease of the coefficient of variation of the ultimate strength and shows an increase of the meanvalue of the ultimate strength.The sensitivity analysis shows the strong influence of the statistic dispersion of the strength of the material on the probability of failure. Under certain circumstances the safety intervall, chosen at conventional calculation, has not the only decisive importance.

3.5 For the linear elastic solution a stress concentration factor at the boundaries of the perforations of 1.7 has been determined as a preliminary information.

7. Next Steps

(With reference to item 3.: Research Program)

3.1 The time dependent pressure- and temperature values are calculated using ZOCO. The frequencies of their occurrence are calculated using failure event sequence analysis.

3.4 The material data of containmentsteels will be adapted to probability density functions in a physically justifiable way. Furhermore material data of fracture mechanics will be recorded as far as they influence the probability of failure of the containment.

3.5 The calculation of the load effects for the given internal pressure- and temperature values will be continued using linear elastic and non linear analysis respectively. Parallel to that, the pipewhipping problem is modeled by a finite element code. For this purpose the force-time relation will be determined using an approximate energy calculation. Furthermore,applying the time history method the dynamic analysis of the steel hull under earthquake loading will be performed.

8. Relation with Other Projects

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9. References

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10. Degree of Availability of the Reports

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Classification: 14	
Title 1 (Original Language): Risikostudie zur Sicherheitsbeurteilung von Kernkraftwerken mit Druckwasserreaktoren für einen Standort in der Bundesrepublik Deutschland (RS 215 u. 217 - I.1.8, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GRS
Title 2 (English): An Assessment of Risk for Nuclear Power Plants PWR on German Siting Conditions	**Project Leader:** Prof. Birkhofer Dr. Heuser
Initiated (Date): 1.6.1976 **Status:** continuing **Completed (Date):** 30.9.1978 **Last Updating (Date):** December 1976	

1. General Aim

The principal objective of the study is an overall risk analysis for a nuclear power plant 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 id 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 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 during Phase A, or respectively on those accident pathes having been found as main contributors to overall risk.

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3. Research Program

The study will be performed by the Gesellschaft für Reaktorsicherheit (GRS) as the main contractor in cooperation with other coworking organizations and institutions, such as the Gesellschaft für Kernforschung (GfK), the Gesellschaft für Strahlenforschung (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
3.11 Dispersion and Consequences
3.12 Risk Assessment

4. Experimental Facilities, Computer Codes

Reliability analysis calculations required for system analysis will be performed with the GRS reliability computer programs CRESSEX /2/ and FESIVAR. Calculations that are required for description of meltdown processes and fission product release from core and containment will be performed with the codes BOIL and CORRAL used in WASH 1400.

Especially with respect to non-comparable siting conditions and a higher population density within the Federal Republic of Germany there is a need for developing an own consequence model apart from this one applied in WASH 1400.

5. Progress to Date

After having started in June 1976 work has been initiated nearly on all subtasks of the research program. In the following some comments are given to various topics.

To 3.3 Loss of Coolant Accidents

First of all the spectrum of different sizes of LOCAs and adjoint requirements for emergency cooling and decay heat removal have been specified according to the requirements of licensing procedure. The corresponding event trees describing various accident sequences have been developed. First calculations of adjoint fault trees for the emergency cooling and decay heat removal systems have been performed.

To 3.4 Transients

A systematic approach for the analysis of transients potentially leading to a core meltdown has been started. Furthermore a detailed event and fault tree analysis starting with the event "Loss of Offsite Electric Power" is underway.

To 3.7 Containment
3.8 Core Meltdown

Referring to the results of the event tree analysis obtained for the large LOCA calculations for core meltdown have been performed by means of the US-program BOIL and for comparison of different modelling using the German program system BILANZ. Subsequently first calculations of long-termed containment pressure response have been done.

To 3.11 Dispersion and Consequences

The consequence model foreseen for the study will be develpped by the GfK in cooperation with GSF. After having outlined conception and main features of the model, special work has been done to establish two versions of a meteorological dispersion model (short distance and long distance model) for computing dispersion of radioactivity in terms of concentration in the air and on the ground as a function of time and distance from the reactor.

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6. Results

7. Next Steps

In addition to the comments given in section 5, brief remarks on some work having been started now or being planned for the next future, will be made in what follows.

To 3.3 Loss of Coolant Accidents

For a more refined development of accident sequence analysis special attention is given to an assessment of uncertainties associated with calculational techniques for emergency cooling.

To 3.5 Component Failure Behavior and Failure Data

A first proof of failure data which are summarized in WASH 1400, App. III, and further completion of this list by data from literature and data that have been derived from own field experience in a conventional plant, /3/ is taken up in cooperation of the GRS with the Technical University of Berlin.

Furtheron investigations on the failure behavior of pressurized components (reactor pressure vessel, steam generator et al.) start with an evaluation of failure statistics available for conventional pressure vessels and steam drums.

To 3.6 Operating Experience

In order to check accident and system analysis of the study an evaluation of operating experience of German nuclear plants especially with respect to PWRs, is planned.

To 3.11 Dispersion and Consequences

In context of the consequence model an evacuation model will be developed by the Institut für Unfallforschung of the Technischer

Überwachungsverein Rheinland. For Phase A of the study influence of evacuation will be handled in a similar way as has been considered in WASH 1400.

8. Relations with other Projects

Besides the methods of probabilistic analysis that have been outlined and applied in WASH 1400 in general, the investigations that are required for the study are performed on the basis of results which are derived from the German Research Program on Reactor Safety which is sponsored by the Federal Ministry of Research and Technology. Hereto special reference is given to the results that have been derived so far for the analysis of core melt accidents.

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/ R. Dougherty, L. Schlösser
CRESSEX-Beschreibung eines Zuverlässigkeitsrechenprogramms zur Ermittlung wichtiger Kenngrößen von komplexen Systemen, Programmbeschreibung,
MRR-P-23, September 1976

/3/ P. Hömke, H. Krause
Der Modellfall IRS-RWE zur Ermittlung von Zuverlässigkeitskenngrößen im praktischen Betrieb,
IRS-W-16, November 1975

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Classification: 14	
Title 1 (Original Language): Unsicherheiten der Ausfallraten von Komponenten und der Eintrittswahrscheinlichkeit störfallauslösender Ereignisse RS 228, Jahresbericht A 1976	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: IKT TU Berlin
Title 2 (English): Uncertainty of the Failure Rates of Components and the Probability of Occurrences of Failure Causing Events	**Project Leader:** Prof. Dr.Memmert
Initiated (Date): Aug. 1, 1976 **Status:** continuing **Completed (Date):** July 31, 1978 **Last Updating (Date):** Dec. 31, 1976	

1. General Aim

Determination of the Uncertainty of the Failure Rates of Components and the Probability of Occurrences 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 is necessary to compile the generally 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 the 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

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4. Experimental Facilities, Computer Codes

ad 3.2 Development of a computer code to calculate the necessary statistic parameters of different distributions for the data which are available

5. Progress to date

ad 3.1 Compilation and examination of already known literature containing reliability data. This has essentially been finished. The addition of more literature to the collection is possible during the whole period of investigation.

ad 3.3 Characteristic values of the logarithmic normal distribution of failure rates quoted in WASH 1400 /1/ were calculated and compared with the statements made there.

6. Results

ad 3.1 Up to now there is no information available about reliability data of components in nuclear power plants from the Federal Republic of Germany. The only systematic data collection was carried out for a conventional 300 MWe power plant /3/. However, these data should be considered in any case, because they represent at the moment the only relevant German data material. The further procedure will be based on the data quoted in WASH 1400 /1/ to which values from /2/ and /3/ will be added.

ad 3.3 The calculated mean values and distributions largely agree with those given in WASH 1400 /1/. Differing results still require detailed analysis.

7. Next Steps

ad 3.1 Additional data collections should be considered

ad 3.2 Determination of mean values and distributions according to certain aspects for a list of defined components.

ad 3.3 Analysis of the deviation of the mean values and distributions from the data in WASH 1400 /1/.

8. Relation with other Projects

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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

/2/ Nuclear Plant Reliability Data System,
1975 Annual Reports of System and Component Reliability, Southwest Research Institute, San Antonio, Texas 78 284

/3/ Zusammenstellung von Ausfallraten im Kraftwerk Neurath-RWE,
Interne Mitteilung des IRS vom 15. Oktober 1976

10. Degree of Availability of the Reports

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Classification 14

Title 1 Vurdering af systemers pålidelighed	Country Denmark Sponsor DAEC Risø ORGANIZATION DAEC Risø
Title 2 Evaluation of the Reliability of Systems	Project leader Hans Erik Kongsø
Initiated 1970 Completed Status: In progress Last updating Currently	Scientists: Hans Erik Kongsø

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 The computer code, RELY 4, has been developed. The code is written in ALGOL, it is of the Monte Carlo type and both direct simulation and simulation with variance reduction can be applied. The code calculates reliability and availability of systems both with and without repair.

A cumputer code, MARKOVA, has been developed. The code is written in FORTRAN, it is an analytic program, and it can be used for solving Markov equations, both the differential equations and the linear equations at equilibrium.

A computer program, REDIS, has been developed. The code is written in FORTRAN, and it is of the Monte Carlo type, based upon direct simulation (no variance reduction). The program calculates a series of reliability characteristics for a given system, and it has been designed with particular emphasis on flexibility and for analysis of details in component- and system performance.

The program can be used for any system in principle, and it is designed for analysis of standby systems, taking into account the various kinds of errors, that can arise during operation as well as

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standby conditions; in addition the program can take into consideration, that in standby systems certain groups of components will have common operation and standby periods. Dependencies between faults and faults with spread on the data can also be considered, and routine testing of selected groups of components can be taken into account.

4.2. Essential results The REDIS and MARKOVA codes have been used for calculation of the unavailability of the feedpump systems in two conventional power plants.

5. Next steps

Future work will include development of computer codes for very large systems and attempts to apply the Monte Carlo technique for cause-consequence analysis.

6. Relation with other projects The projects has close relations to the Risø-project concerning the cause-consequence method (classification 6).

7. Reference documents RELY 4. A Monte Carlo Computer Program for System Reliability Analysis. Risø-M-1500, H.E. Kongsø. June 6, 1972.

REDIS, A Computer Program for System Reliability Analysis by direct Simulation. IAEA-SM-195/17. H.E. Kongsø. April 1975.

REDIS, A Computer Program for System Reliability Analysis by direct Simulation. Program Description and Manual. Risø-M-1781. Hans Erik Kongsø and Robert Korre Larsen, June 1975.

8. Availability The project information is freely available apart from cases, where commercial interests may be violated.

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.

8. Availability The project information is freely available apart from cases, where commercial interests may be violated.

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Classification 14

Title 1 Probabilistisk brudmekanik	COUNTRY Denmark SPONSOR DAEC Risø ORGANIZATION DAEC Risø
Title 2 Probabilistic Fracture Mechanics	**Project leader** Per Becher
Initiated 1970 **Completed** **Status:** In progress **Last updating** Currently	**Scientists** Per Becher

1. General aim To develop methods for evaluation of the reliability of pressure vessels for the purpose of safety analysis.

2. Particular objectives Development of computer codes, based upon probabilistic methods and fracture mechanics, for evaluation of the reliability of pressure vessels.

3. Experimental facilities

4. Project status

4.1 Progress to date

Two computer codes was developed, PFM 690 and PEP 706. Both are written in FORTRAN in versions for a Burroughs B 6700 and a IBM 370/160.

PFM 690 is a monte carlo program, which calculates the crack distribution as a function of time, on the basis of an initial crack, crack growth characteristics and stress transients. PEP 706 uses monte carlo with importance sampling and calculates the probability of failure of a pressure vessel, based upon the distribution functions for cracks, stresses, yield strength and charpy-V.

4.2 Essential results

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5. Next steps So far the models have been based upon linear elastic fracture mechanics, but attempts will be made to incorporate elastic-plastic fracture mechanics into the models.

In coorperation with other research institutions and power reactor vendors analysis of actual reactor pressure vessels will be carried out, based upon more pertinent data from practical experience. The specific objectivs being assessment of the influence of the neutron embrittlement and development of a new crack growth model.

6. Relation with other projects The project has relations to the Risø projects, dealing with stress analysis of primary steel components and with evaluation of the reliability of systems.

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. (Presented as paper no. M6/4 at 2. Smirt Conf., Berlin 10 - 14th September 1973).

8. Availability The project information is freely available apart from cases, where commercial interests may be violated.

123-11-10/4105-30

Classification : 14

1625

Titre	Pays :
Banque de données de fiabilité.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais)	Organisme exécuteur : CEA/DSN-SETSSR
Reliability data bank.	Responsable : MME.A.CARNINO (DSN-SETSSR)
Date de démarrage : 01/01/73 Date prévue d'achèvement 31/12/81 Etat actuel : en cours Dernière mise à jour : 27/01/77	Scientifiques :

Objectif général :

L'action a comme objectif général de fournir des données de fiabilité qui représentent la base indispensable de toutes les analyses de sûreté par les méthodes probabilistes. Il s'agit en permanence de collecter des données sur les matériels des centrales où cela est possible et de rechercher les données jugées "crédibles" dans la littérature ou dans des enquêtes.

Objectifs particuliers :

Une action de collecte de pannes et fichier informatisé correspondant marche depuis 1973 sur le réacteur Phénix. Compte tenu du temps cumulé de fonctionnement, des données intéressantes pour la filière pourront être extraites par un travail d'étroite collaboration avec EDF-Production Thermique. Il a été possible de tirer des données d'exploitation de centrales nucléaires françaises (Saint-Laurent 1 + 2), ainsi que d'extraire de l'expérience de centrales conventionnelles, certaines enquêtes sur des matériels électriques.

123-11-20/4105-30

Classification : 14

1627

Titre	Pays :
Données de fiabilité tirées de l'expérience de fonctionnement d'un réacteur nucléaire.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Reliability data from the operating experience of a nuclear reactor.	Organisme exécuteur : CEA/DSN - SETSSR
	Responsable : MME A. CARNINO (DSN - SETSSR)
Date de démarrage : 01/01/73 — Date prévue d'achèvement : 01/03/77 Etat actuel : en cours terminaison — Dernière mise à jour : 02/11/76	Scientifiques : MME. C. VILLEROUX, A. BLIN

Objectif général :

Détermination de données opérationnelles de fiabilité à partir de l'expérience de fonctionnement du réacteur de Saint-Laurent-des-Eaux.

Objectifs particuliers :

Calculer, par les informations contenues dans le fichier de Saint-Laurent, les taux de défaillances et de réparations des matériels. Confronter ces résultats avec les résultats internationaux.

Etat de l'étude :

1) Avancement à ce jour :

Analyse des informations de Saint-Laurent en 1973 et 1974.
Calcul de données concernant les vannes et pompes de Saint-Laurent en 1975 et 1976.

2) Résultats essentiels :

Rapports sur les taux de défaillance de calculateurs, de thermocouples, les turbosoufflantes, les relais, des pompes et des vannes. Erreur humaine.

Prochaines étapes :

Comparaison des résultats avec des organismes étrangers (NRC, SRS, UKAEA).

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Documents de référence :

"Résultats de fiabilité du fichier d'incidents de la centrale de Saint-Laurent des Eaux". A. BLIN- Rapport SETS 27, octobre 1973.
"Données de fiabilité des vannes de la centrale de Saint-Laurent des Eaux", C. VILLEROUX - Rapport SETS 40, novembre 1975.

133-11-10/4105-20

Classification : 14

1629

Titre	Pays :
Développement du programme PATREC pour le calcul de la fiabilité de systèmes complexes.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Development of the PATREC code for calculation of complex systems reliability.	**Organisme exécuteur :** CEA/DSN - SETSSR
	Responsable : MME A. CARNINO (DSN - SETSSR)
Date de démarrage : 01/04/72 Date prévue d'achèvement : 31/12/76 Etat actuel : en cours terminaison Dernière mise à jour : 02/11/76	**Scientifiques :** A. BLIN A. PAVRET

Objectif général :

Calcul par arbre de défaillances 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.

Objectifs particuliers :

- Améliorations successives du programme à partir de la 1ère version de 1972.
- Optimisation de la version PATREC-MC permettant de calculer la sensibilité du résultat aux variations des paramètres d'entrée.
- Optimisation de la version PATREC-DE.

Etat de l'étude :

1) Avancement à ce jour :

Améliorations successives apportées à la 1ère version de PATREC, datant de 1972.

2) Résultats essentiels :

Les améliorations obtenues portent sur : Le traitement des dépendances, les lois de probabilités, l'optimisation du temps de calcul, l'utilisation des techniques de MONTE-CARLO pour le calcul de la dispersion du résultat selon les dispersions de probabilités élémentaires, la recherche des chemins critiques, l'introduction de diagnostics d'erreurs.

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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-DE.

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.

133-11-11/4112-20

Classification : 14
1631

Titre	Pays :
Utilisation des techniques de MONTE CARLO pour les calculs de fiabilité.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) The use of MONTE CARLO techniques for reliability calculations.	Organisme exécuteur : CEA/DRE-SERMA
	Responsable : MME A.CARNINO (SETSSR) MME JM.LANORE (SERMA)
Date de démarrage : 01/01/77 Date prévue d'achèvement : 31/12/77 Etat actuel : en cours Dernière mise à jour : 01/02/77	Scientifiques : B.DUCHEMIN

Objectif général :

Amélioration du code PATREC par les techniques de MONTE CARLO.

Objectifs particuliers :

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.Un biaisage sera recherché. Diverses applications seront réalisées.
Une étude MONTE CARLO sera étendue à la résolution analytique d'un graphe de MARKOV à transitions non constantes.

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.

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Prochaines étapes :

Réagencement et reprogrammation de certaines parties de PATREC pour y inclure une vérification et une évaluation de modes communs de défaillances sur un arbre de défaillances. Résolution de graphes de MARKOV à transitions non constantes par MONTE CARLO.

Documents de référence :

"PATREC-MC - Programme de calcul de l'incertitude de la probabilité de défaillance d'un système complexe par la méthode de MONTE-CARLO", H.KALLI, JM.LANORE, Rapport SERMA N° 263.
"Calcul de la disponibilité d'un système complexe par la méthode de MONTE-CARLO", H.KALLI, JM.LANORE - Rapport SERMA T.281.
"Méthode de biaisage appliquée au programme de fiabilité PATREC-MC" H.KALLI, JM.LANORE, Rapport SERMA S.285.

133-11-20/4105-20

Classification 14

1633

Titre	Pays :
Adaptation de programmes de calcul de fiabilité.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Reliability calculation programs adaptation.	Organisme exécuteur : CEA/DSN -SETSSR
	Responsable : MME.A.CARNINO (SETSSR)
Date de démarrage : 01/01/75 Date prévue d'achèvement : 01/06/77 Etat actuel : en cours Dernière mise à jour : 02/11/76	Scientifiques : D.LACOTTE

Objectif général :

Adaptation de codes de calcul étrangers pour le calcul de la fiabilité et de la disponibilité de systèmes complexes par arbres de défaillances : PREP et KITT, code US ; CADI, code EURATOM (ISPRA)

Objectifs particuliers :

Optimisation et améliorations de ces codes sur ordinateurs de Saclay.
Recherche des limites d'utilisation des codes.

Etat de l'étude :

1) Avancement à ce jour :

Version originale de PREP et KITT rendue opérationnelle à Saclay.
Version originale de CADI rendue opérationnelle à Saclay.

2) Résultats essentiels :

PREP-TRI et KITT 7 : Versions améliorées de PREP, KITT et CADI en production à la CISI.

Prochaines étapes :

Réunion en un seul programme de PREP-TRI et de KITT 7 et augmentation des performances de ces codes. Adaptation des codes précédents à un petit ordinateur (WANG) au moyen du FORTRAN de base.

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<u>**Documents de référence :**</u>

"PREP et KITT : Note d'utilisation", D.LACOTTE - Rapport SETS 35, Juillet 1975.

"Les équations de la Kinetic Tree Theory - méthode de calcul par ordinateur de la fiabilité des systèmes", D.LACOTTE - Rapport SETSSR 46, décembre 1975.

"PREP-TRI : un programme de calcul pour la recherche des coupes minimales des arbres de défaillance - Notice d'utilisation", D.LACOTTE, Rapport SETSSR 49, juin 1976.

"KITT 7 : un programme de calcul de la fiabilité de grands systèmes complexes", D.LACOTTE, Rapport SETSSR 50, août 1976.

133-11-30/4105-20

Classification : 14

1635

Titre	Pays :
Codes de calcul de l'indisponibilité de systèmes STANDBY testes.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Tested STANDBY systems unavailability calculation codes.	Organisme exécuteur : CEA/DSN -SETSSR
	Responsable : MME A.CARNINO (SETSSR)
Date de démarrage : 01/12/75 Date prévue d'achèvement : 01/09/77 Etat actuel : en cours Dernière mise à jour : 02/11/76	Scientifiques : JP.SIGNORET

Objectif général :

Calcul de la disponibilité instantanée et moyenne d'un système en attente testé à intervalles réguliers.
Recherche de l'intervalle optimal entre tests.

Objectifs particuliers :

Elaboration d'une formule d'indisponibilité utilisable par PATREC.
Calcul de la disponibilité d'un système à 1 composant.
Calcul de la disponibilité d'un système complexe.

Etat de l'étude :

1) Avancement à ce jour :

Formulation théorique.

2) Résultats essentiels :

Versions provisoires des codes INDI-1 et INDIGO.

Prochaines étapes :

Généralisation de la formulation (Rapport).
Programmation de codes de calculs.

Titre	Pays : FRANCE
Fiabilité des systèmes de sécurité des P.W.R.	Organisme directeur : CEA
Titre (anglais) Reliability of safeguard systems of P.W.R.	Organisme exécuteur : CEA/DSN-SETSSR
	Responsable : MME A. CARNINO
Date de démarrage :01/04/73 Date prévue d'achèvement :31/12/81 t actuel : en cours Dernière mise à jour : 02/11/76	Scientifiques : A. BLIN JP. SIGNORET R. QUENEE

Objectif général :

Etudier la fiabilité/disponibilité de tous les systèmes de sauvegarde et des systèmes vitaux des centrales P.W.R.

Objectifs particuliers :

Répondre aux besoins de l'analyse de sûreté de Fessenheim et Bugey.
Etudier le comportement de certains systèmes dans le cas d'accidents.

Etat de l'étude :

1) Avancement à ce jour :

Un certain nombre de systèmes de sauvegarde et vitaux des centrales PWR ont été étudiés. (Fessenheim et Bugey).

2) Résultats essentiels :

Les résultats sont consignés dans les rapports cités en référence.

Prochaines étapes :

Analyse d'autres systèmes des P.W.R. et du comportement à long terme après accident de certains systèmes (E.A.S., R.C.V., R.P.R.)

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Documents de référence :

"Centrale de Fessenheim - Analyse qualitative partielle de la fiabilité du circuit d'alimentation de secours des générateurs de vapeur à l'aide de la méthode des arbres de défaillances", JP.SIGNORET, B.GACHOT, P.VUILLEMIN - Rapport DSN 48, septembre 1974.

"Analyse qualitative et quantitative de la fiabilité du circuit hydraulique d'injection de sécurité basse pression (ISBP)", JP.SIGNORET - Rapport SETSSR 33.

"Analyse qualitative et quantitative de la fiabilité du circuit hydraulique d'injection de sécurité haute pression (ISHP)", JP.SIGNORET - Rapport SETSSR 32.

"Fiabilité du circuit RRI de Fessenheim", A.BLIN - Rapport SETSSR 48, juin 1976.

142-11-10/4105-20

Classification : 14

1639

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 : MME A. CARNINO(SETSSR)
Date de démarrage : 01/09/76 Date prévue d'achèvement : 31/12/81 Etat actuel : démarrage Dernière mise à jour : 02/11/76	Scientifiques : JP. SIGNORET J. DUBAU

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.

Prochaines étapes :

Détermination des initiateurs.

Documents de référence :

"Séquences accidentelles importantes pour la sûreté d'une centrale à eau", A. CARNINO, B. GACHOT, JP. SIGNORET - Rapport DSN 62, Avril 1975.

1641

PROJECT TITLE :	CLASSIFICATION
Fault analysis of the conventional island in a LWR nuclear power plant.	14
SPONSORING COUNTRY :	ORGANISATION : FRANCO TOSI S.p.A.
DATE INITIATED : April 15, 1975 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 VIII 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 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-equlibrium 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

1645

Title 1 (Original language)	Classification 14
Title 2 (English) Interactive Synthesis of Fault Trees for Safety Assessment of Nuclear Power Plants	Country ITALY Sponsor CNR-EURATOM Organisation CESNEF-Politecnico Milano
Date initiated 1976 Date completed 1978 Last updating April 1977	Project Leader 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 **Sponsor** FIAT-T.T.G. **Organisation** Nuclear Energy Division
Date initiated June 1976 **Date completed** 1980 **Last updating** April 1977	**Project Leader** 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)

1648

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.

1649

Title 1 (Original language)	Classification 14
Title 2 (English) Fault-Tree Sensitivity Analysis for Reliability Calculations in Nuclear Power Plants	Country ITALY Sponsor IAEA Organisation CESNEF-Politecnico Milano
Date initiated 1975 Date completed 1977 Last updating Sept. 1976	Project Leader 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).

1651

TITLE 1 (original language) Studi di affidabilità	Classification 14
TITLE 2 (english) Reliability studies	Country: ITALY Sponsor: CNEN Organisation: CNEN
Date initiated October 1974 Date completed in progress Last updating June 1976	Project Leader 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.

1653

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 Date completed In progress Last updating April I977	Project Leader G. Tomassetti

General aim

Studies on reliability data banks through experimental and metodo logical 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

1655

TITLE 1 (original language)	Classification
Fault tree analysis for nuclear power plants	$\boxed{6}$-14
TITLE 2 (english)	Country: ITALY Sponsor: CESNEF, Politecnico di Milano Organisation: " "
Date initiated 1973 Date completed 1975 Last updating	Project Leader S. GARRIBBA

1657

TITLE 1 (original language) Fault analysis of the conventional island in a LWR nuclear power plant.	Classification $\boxed{6}$ - 14
TITLE 2 (english)	Country: ITALY Sponsor: Franco Tosi S.p.A. Organisation: " " "
Date initiated April 15, 1975 Date completed April 15, 1976 Last updating June 1976	Project Leader V. Bedogni

1659

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

1661

PROJECT TITLE : Statistical analysis of random signals for safety problems	CLASSIFICATION $\boxed{10}$ - 14 - 8
SPONSORING COUNTRY : Italy	ORGANISATION : CNEN
DATE INITIATED : 1966 DATE COMPLETED : in progress	PROJECT LEADER : A. Federico

1665

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

1.665

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

1667

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 Sponsor CNEN Organisation CNEN
Date initiated 1/1/1976 Date completed in progress Last updating June 1977	Project Leader A. Serra

1669)

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 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.

31.719

PROJECT TITLE : Status Report on Data Systems	LWR 14
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : JRC - Ispra
DATE INITIATED : 1976 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.

1672.

3. <u>Project status</u>

The following systems so far have been analyzed as for input, processing, output : IRS (Germany), ENEL (Italy), EDF (France), SRS (England), NPRDS (USA).

4. <u>Next steps</u>

Investigation on abnormal occurrencies as a supplement to the Reliability Data Status Report. Compatibility study on components/systems and failure characteristics.

5. <u>Reference documents</u>

JRC Ispra Safety Programme Annual Progress Reports 1976.

1673

PROJECT TITLE : Validation Experiments of a Reliability Analysis	LWR 14
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : JRC Ispra
DATE INITIATED : 1976 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 - Δ 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 (Δ W) in primary pressure variation (Δp) taking into account also the pressure gradient Δp.
  - 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.

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 (± 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.

PROJECT TITLE : Probabilistic Accidental Transients Analysis	LWR 14
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : JRC Ispra
DATE INITIATED : 1974 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].

31.719

1676

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.

1677

NETHERLANDS ENERGY RESEARCH FOUNDATION (ECN)		CLASSIFICATION: 14
TITLE: Faalweg en faalkansvoorspellingen van reaktorsystemen		COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs ORGANIZATION: ECN
TITLE: (ENGLISH LANGUAGE): Failure mode and failure rate prediction of reactor systems		PROJECTLEADER: K. Terpstra
INITIATED: June 1974	LAST UPDATING: March 1977	SCIENTISTS: -
STATUS: Progressing	COMPLETED: Summer 1978	

1. 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.

2. 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.

3. Experimental facilities: None

4. Project status:

Updating theoretical knowledge and evaluation and updating of existing computer codes

5. Next steps

The next step is to create a balanced code to do the following:
a. Constructing the logical fault tree
b. To determine the MCS
c. To calculate the availability of components and system
d. To rank components and MCS according to their importance with respect to the stop event in a certain measure.

After creating the code, a test run has to be made of a reactor safety device.

6. Relations with other projects: -

7. Reference documents: -

8. Degree of availability

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

9. Budget: -

10. Personnel: 0.9 manyear

Classification 14	
Title 1 Faalkans-analyse met behulp van gebeurte-nissen- en foutenbomen	Country The Netherlands Organization KEMA
Title 2 Failure analysis by application of event- and fault trees	Projectleader R.W. van Otterloo

1. 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.

2. 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.

3. Experimental facilities

Not applicable.

4. Project status

Methods and computer codes have been compared.

5. Next steps

Not applicable.

6. Relation with 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.

1680

7. Reference documents

See 6.

Several applications of this method are written in the Dutch and the English language.

8. Degree of availability

Through the organization KEMA.

15. INTERRELATION BETWEEN REACTOR PLANT AND OPERATING PERSONNEL

1681.

Classification 15

Title 1	
Operatør studier	COUNTRY Denmark SPONSOR Research Establishment, Risø ORGANIZATION Research Establishment, Risø
Title 2 Study of the Process Operator	Project leader: L.P. Goodstein
Initiated: Approx 1966 - in its present form 1973 Completed: Status: progressing Last updating: 1976	Scientists: L.P. Goodstein 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.

J. Rasmussen 1973
The Role of the Man-Machine Interface in Systems Reliability
November 1973, Risø Report R-10-73
Reprinted from NATO Conference on Generic Techniques in Systems Reliability Assessment. Liverpool, July 1973

J. Rasmussen 1974
The Human Data Processor as a System Component
Bits and Pieces of a Model, June 1974, Risø Report R-8-74

J. Rasmussen 1976
Outlines of a Hybrid Model of the Process Plant Operator
Risø Note N-7-76
Reprinted from NATO Conference on Monitoring Behaviour & Supervisory Control, Berchtesgaden, March 1976.

1683

J. Rasmussen & J.R. Taylor
Notes on Human Factors Problems in Process Plant Reliability & Safety Prediction - Risø-M-1894 1976.

8. Available from the library at Risø.

124-11-10/4600/50

Titre	Pays :
Paramètres humains de la sûreté.	FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Human factors in nuclear safety.	Organisme exécuteur : CEA/DSN - SETSSR
	Responsable : MME A. CARNINO (DSN - SETSSR)
Date de démarrage : 01/01/76 Date prévue d'achèvement : 31/12/81 Etat actuel : en cours Dernière mise à jour : 23/12/76	Scientifiques :

Objectif général :

Lorsqu'on étudie la causalité élémentaire des accidents ou incidents nucléaires on constate que les composants d'origine humaine sont majoritaires, il était donc capital de s'efforcer d'étudier scientifiquement l'organisation humaine, 3ème composante d'une installation nucléaire.

Objectifs particuliers :

3 objectifs ont été déterminés :

1) Analyse des organisations humaines autour des installations nucléaires;
2) Etudes ergonomiques particulières influant directement sur la sûreté nucléaire (interface homme/pupitre de commande, entretien en milieu radioactif) ;
3) Essai de mesure de fiabilité humaine dans l'industrie nucléaire.

Etat de l'étude :

1) Avancement à ce jour :

Deux études d'organisation ont été lancées en 1976 : près d'un réacteur ; près d'une installation de chimie radioactive.

Prochaines étapes :

Deux autres installations vont être prochainement étudiées.

Relation avec d'autres études :

Ces études sont en relation :
Avec les études entreprises sur la part nucléaire au département de protection;
Avec les travaux de fiabilité entrepris par le groupe fiabilité au DSN.

1687

Classification: 15.1		
Title 1 (Original Language): Ermittlung und Analyse menschlicher Funktionen beim Betrieb von Kernkraftwerken (SR 100 - II.5.2., Jahresbericht A 75)		COUNTRY: BRD SPONSOR: BMI ORGANIZATION: TÜV Rheinland, Köln
Title 2 (english): Identification and Analysis of Functions of the Human Operator in the Operation of Nuclear Power Plants		Project Leader: Prof. Dr. Kuhlmann
Initiated (Date): April 1973	Completed (Date): 1975	
Status: finished	Last Updating (Date): December 1975	

1. General aim

Operation experiences in nuclear power plants indicate that the human operator considerably contributes to the system output on normal as well as on faulty operating conditions. Thus an effective power plant design and development has to take into systematic account the possibilities and limitations of the human element. In view of this the study aims at identifying and analyzing the functions of operating and maintenance personnel.

2. Particular objectives

The objective of the project is to find out to what extent the above requirements are met in existing plants, i.e. to identify what the human operators are required to do and how they achieve it. It is expected that these analyses will result in basic Human Factors recommendations for the design of safe and effective operating and maintenance procedures.

3. Program

3.1 Experimental facilities

None.

1688

3.2 Research program

(1) Search, compile and annotate the literature of both technical and Human Factors origin in order to give a survey of the present state of the art.

(2) Analyze functions of the plant personnel in terms of tasks and responsibilities assigned to them by operating procedure manuals, work regulations, etc.

(3) Analyze incident reports to better take into account random events that cannot be observed directly.

(4) Administer interviews to operating and management personnel in order to obtain informations on the functions as well as on their subjective evaluation.

(5) Observe directly personnel carrying out routine and, if possible, non-routine work. This is considered a major source of information for identifying and analyzing tasks.

4. Project status

4.1 Progress to date

Functions of operating and maintenance personnel in three nuclear power plants were investigated by consideration of both nonfunctional and dynamic factors. The latter were established by observations of tasks carried out, by interviews administered to personnel, and by analyzing reported operations of all kinds. The former ones included recording and evaluating equipment, facilities, job aids, regulations, etc., as well as interviews. Besides, pertinent literature references and incident reports were compiled and evaluated. The data collection activities have been terminated.

4.2 Essential results

Results were obtained on the role of static characteristics such as the design of displays, controls, and communication facilities, structural and technical design, job aids, and personnel and work organization. Further results concerning the performance of tasks include the use of information transmission equipment and of manuals, the analysis of individual activities and the structure of tasks, the impact of characteristics of work and personnel organization, and of particular working modalities such as the physical environment.

5. Next steps

Organize and report the results of the study.

6. Relation with other projects

The study is related to projects RS 70 (completed September, 1973) and SR 36 (in progress).

7. Reference documents

Quarterly reports (in German) in the series "IRS-Forschungsberichte":

Report period covering Jan.-Mar. 1975
Report period covering Apr.-Jun. 1975
Report period covering Jul.-Sep. 1975

8. Degree of availability

The "IRS-Forschungsberichte" can be obtained from the organization issuing these research reports.

Classification: 15.2	
Title 1 (Original Language): Entwicklung und Aufbau eines Ausbildungssystems im Medienverbund zur Intensivierung der Schulung und Ertüchtigung von Betriebspersonal von Kernkraftwerken (RS 152 - II.5.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Frankfurt
Title 2 (English): Development of a Training System for the Staff of Reactor Plants	Project Leader: H. Martin
Initiated (Date): 1.1. 75 — Completed (Date): 31. 3. 77 Status: Continuing — Last Updating (Date): 31. 12. 76	

1. General Aim and 2. Particular Objectives

Information systems have been investigated which were suitable for training the staff of operating reactors. Criteria for the choice of an optimal schooling program were to be developed, considering several information carriers and media. An example of a training program had to be produced and tested.

3. Research Program

Work was concentrated on the following program:

a) Investigation of the applicability of various information systems.
b) Development of a suitable combination of several media.
c) Production criteria for schooling programs.
d) Build-up of training programs for the staff.
e) Production of special training examples.
f) Documentation.

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4. <u>Experimental Facilities</u>

No experimental facilities necessary.

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

Comparative tests with diapositive-projectors with higher power were performed. Tests were started with super-8 and 16 mm films. The quality of the pictures was compared.

Some production examples were created for group schooling purposes. The time sequences were optimized.

An information program for the staff of power plants was developed.

The special training (detail work on systems and functions of a special power station) was considered during the start-up phase on the base of one day per week.

6. <u>Results</u>

The diapositive projectors were not well suited for special purposes. The tests with super-8 and 16 mm films showed that the super-8-film needed higher effort for illumination and had a lower quality of the pictures compared with the 16 mm film.

A trick-technique for the presentation of components by successive build-up of single parts has shown good results. This methods allowed the arrangement of components in the right series.

The time criteria for single szenes and sequences were checked. Real-szeneries should not be longer than 5 min. Theoretical presentations and trick sequences should follow, but not longer than 50 - 60 sec.

The training program will consist of 4 - 5 phases. The basic course has to procure fundamental knowledge. The second phase gives basic informations of the special power station type and the systems. The third phase gives details of special systems and functions. The fourth phase offers simulator-training of dynamic processes, malfunctions and normal operation. The fifth phase is the so-called "on the job"-training when the staff gets the needed knowledge during active work on the station.

7. Next Steps

Corrective discussions with utilities about details of the suggested training program will be completed.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

1695

Classification 15.2

Title 1	Country / Sponsor / Organization
Specificatie van eisen voor bedienend personeel in kern(energie) centrales.	**Country** Nederland **Sponsor** Ministry of Social Affairs **Organization** Reactor Centrum Nederland

Title 2	Projectleader
Specification of requirements for operating personnel at nuclear (power)stations.	F.M. de Meulemeester

Initiated August 1974 Completed July 1975 (intended)

Status Progressing Last updating n.a.

1. General aim

To advise the authorities on the training requirements for operating personnel at nuclear power stations.

2. Particular objectives

The programme will consist of:

- an inventory and evaluation of requirements imposed on the operating personnel bij the Dutch utilities and the requirements prescribed by the authorities in other coutries,
- preparing a report to the Dutch authorities on the basis of the results obtained from the forementionned study.

1696

3. Experimental facilities	)	
4. Project status	)	
5. Next steps	)	not applicable
6. Relation with other projects	)	
7. Reference documents	)	

8. Degree of availibility

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

16. ENVIRONMENTAL PROTECTION

1697

Classification: 16.2	
Title 1 (Original Language): Stereobildübertragungssystem (RS 113 - I.2.3., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Forschungslab.Klein-wächter, Lörrach
Title 2 (english): Stereo-Television System	Project Leader: Prof. Dr. Klein-wächter
Initiated (Date): .pril 1974 — Completed (Date): June 1975 Status: finished — Last Updating (Date): June 1975	

General Aim

The aim of this investigation was constructing a three dimensional picture transmission system for observation tasks in radioactive contaminated surroundings. Television equipment can give spatial view to man, attribute of remotely controlled saving and repairing operations - especially in telemanipulator applications.

Particular Objectives

The objective of the project's first step was the conception of an elemantary stereo-tv-system for demonstrating that you can effect a spatial impression of transmitted movable pictures in man's mind. The principle is to transmit the signals of two cameras that are arranged in man's eye distance beside each other. These pictures are reproduced in two different colours on a colour monitor. The pictures are separated by coloured spectacles. Subsequentely, stage 2 and later on stage 3 provided the progressive improvement of a fadeless colour stereo-tv-system.

Project Status

Progress to Date / Essential Results

Several stereo-tv-systems have been realized. After having completed all works on the complementary colour system, the elements for a chopper system were selected. The most suitable material is ferroelectrical ceramic because of its short rise and fall times. These elements are still in the experimental status and some probes arrived with Klera at the end of the project time, so that not all works could be completed in time. All electronical units were prepared.

1698

To reach the aim explained above - especially in colour - a twoway-system with colour-tv-equipment was planned and realized. The camera head consisting of two colour-tv-cameras was optimized, all degrees of freedom are remotely controlled. These degrees are focus, zoom, convergence - pitch and azimuth-axis of the camera head. The aperture is controlled automatically.

The complementary system of stage one as well as the colour-tv-system of stage two and three give a very good spatial impression to the viewer if the cameras are optimally adjusted.

Next steps

The chopper system should be completed. A prolongation of the project was proposed at the end of stage 3.

Relation with other Projects

The development is done for the KTH of the GfK-Karlsruhe and coordinated with RS 21 (Synchronous Telemanipulator System).

Reference Documents

Quarterly reports in the IRS-Forschungsberichte (German).

Report period	April	1974	- June	1974	V 74/2
	July	1974	- Sept.	1974	V 74/3
	Oct.	1974	- Dec.	1974	V 74/4
	Jan.	1975	- March	1975	V 75/1
	April	1975	- June	1975	V 75/2
Annual report (Engl.)	April	1974	- Dec.	1974	IRS-F-24

Degree of Availability

Request necessary.

1699

Classification: 16.2	
Title 1 (Original Language): Ferngesteuerte Arbeitsgeräte und mobile Systeme zur Schadenserfassung (PNS 4422 - I.2.3., Jahresbericht A 75)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** GfK, Karlsruhe
Title 2 (english): Remotely Controlled Working Gear and Mobile Systems for Damage Assessment	**Project Leader:** G.W. Köhler
Initiated (Date): 1971 **Status:** finished	**Completed (Date):** December 1975 **Last Updating (Date):** December 1975

General aim

For the damage assessment after emergency cases in nuclear facilities it can be necessary to dispose remotely controlled mobile systems with manipuators.
The development and improvement of those systems is the aim of this project.

Progress to date and essential Results

The chassis of the lightweight MF3 manipulator vehicle for the Nuclear Emergency Brigade was completely assembled and subjected to works trial runs and acceptance tests.
All funktions of the MF3 chassis with its variable geometry chassis were verified in accordance with specifications.
Two electric MasterSlave EMSM II manipulators and the action control system for the whole MF3 system including command transmission, transmission of information and power supply were subjects of a tendering procedure after the end of the project design phase and following completion of the design. Fixed price bids have now been submitted by industries on the EMSM II manipulators and the action control system.
The possibilities of employing "MF3" have been studied in depth and covered in an internal report.

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The Expert Committee on "Equipment for Emergencies and for Removing the Consequences of Incidents" of the Federal Ministry of Research and Technology has not been able to recommend the application for funding the development projects of electric MasterSlave manipulators, "EMSM II" and of the action control system for "MF3". This is due to the responsibilities for the Nuclear Emergency Brigade, which presently are in need of clarification.

The additions for test rigs to the electric EMSM I MasterSlave prototype manipulator have been finished.

The "EMSM" manipulator has been tested in detail. In the light of the experience accumulated, the load carrying capability and the cooling system were improved upon and the elasticity of the system was reduced.

Next Steps

The "EMSM II" project and the action control system will again be submitted to the responsible body of experts as soon as the responsibility for the Nuclear Emergency Brigade has bee clarified.

Reference Documents

"Manipulator vehicle system MF2 and its possibilities of application", Kerntechnik (1975) No. 12 (german and english)

Report KFK 1859 (1973) p. 215 (german)
Report KFK 1908 (1973) p. 235 (german)
Report KFK 2050 (1974) p. 268 (german)
Report KFK 2130 (1974) p. 336 (german)
Report KFK 2195 (1975) p. 446 (german)
Report KFK 2262 (1975) p. (german)

Semiannual reports in the series IRS-Forschungsberichte
Report period Jan.-June 1974 IRS-F-21 (german)
July-Dec. 1974 IRS-F-23 (german)
Jan.-June 1975 IRS-F-26 (german)

Degree of Availability

The distribution of the KFK-reports is restricted.

17. NUCLEAR ACCIDENT RECOVERY AND DECOMMISSIONING

1701

Classification: 17.1		
Title 1 (Original Language): Entwicklung von Dekontaminationsverfahren (PNS 4411 - I.2.4, Jahresbericht A76)		COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (English): Development of Decontamination Methods		Project Leader: Dr. Dippel Dr. Hentschel DI Kunze
Initiated (Date): Jan 1973 Status: Continuing	Completed (Date): Dec 1977 Last Updating (Date): December 1976	

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

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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 Development of cleansing pastes and application of molten salts for the decontamination of metal surfaces

Cleansing pastes have been prepared by mixing combinations of HF, HNO_3 and HCl with a HF resistent, highly dispersed baryte type material. The effectiveness of these pastes has been proved with samples taken from power stations.

$KHPO_4$ and sodium peroxide powder have been applied as decontaminants by coating metal surfaces to be decontaminated. Both melts are very effective, as could been shown by decontaminating power station samples.

The development of technical equipment for the application of pastes and molten salt to surfaces in thin layers is finished and the components in question are ordered.

5.2 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.

5.3 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. Further experiments are on the way. Out of others HNO_3, HNO_3/HF, alcaline KMO_4 solution, citric acid and oxalic acid are tested.

6. Results

6.1 Development of cleansing pastes and application of molten salts for decontamination of metal surfaces

Baryte material based pastes show in all experiments and application tests the better effectiveness than pastes on polyethelene-titanium-oxide-mixtures. Pastes without hydrochloric acid, have a similar effectiveness than those containing hydrochloric acid, but the decontamination time is longer, about twice as much. Because of the absence of the polyethelene powder the thermal stability of the paste is higher and difficulties in the transportation of the waste waters caused by this powder are eliminated.

It could be demonstrated, that $KHPO_4$ as decontamination agent is very efficient in decontaminating primary circuit samples from PWRs. Activities of 0.5 to 1.5 $\mu Ci/cm^2$ prior to decontamination are removed up to between 1% and 5% residual activity.

Regarding the technical equipment for coating contaminated surfaces it has been demonstrated that pastes can be applied in thin layers. Only current type technical components are necessary. Compatibility problems are solved by selection of special surface protection coatings for all those parts which come in contact with the chemical agents.

6.2 Optimization of a Liquid Cleanser

The above mentioned mixtures show a definitely better thermal stability in comparison to currently used liquid decontamination

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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.

6.3 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 m citric acid/0.3 m oxalic acid mixture are the most effective.

7. Next Steps

7.1 Development of cleansing pastes and application of molten salts for decontamination of metal surfaces

After a technical demonstration of the coating equipment these developments are finished.

7.2 Optimization of a Liquid Cleanser

In case a composition of the cleanser can be found the effectiveness of which is comparable with that of Papan-Dekopan for the isotopes selected, the range of application of the cleanser will be extended to glass and plastic surfaces.

7.3 Optimization of the Decontamination of Components

The experiments mentioned are continued aimed at optimal concentration with respect to decontamination effectiveness. In this the use of other application techniques is included.

8. Relation with Other Projects

There is a cooperation with KAH company in the development of decontamination agent application equipment.

9. References

Th. Dippel, D. Hentschel, S. Kunze
Decontamination and Decontamination Wastes
Kerntechnik, 12, 256-531, 1976

10. Degree of Availability of the Reports

The report is available at the authors.

Classification: 17.3	
Title 1 (Original Language): Entwicklung von Methoden und Verfahren zur Stillegung und Endbeseitigung nuklearer Anlagen (PNS 4421 - I.2,4, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** **ORGANIZATION:**
Title 2 (English): Development of Methods and Techniques for the Decommissioning and Ultimate Disposal of Nuclear Facilities	**Project Leader:** G.W. Köhler
Initiated (Date): Jan. 1976 **Status:** Continuing **Completed (Date):** 1978 **Last Updating (Date):** December 31, 1976	

1. General Aims

Development of methods and techniques for the decommissioning and ultimate disposal of nuclear facilities.

2. Individual Tasks

2.1 Studies on the conditioning of large reactor components.

2.2 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

Conditioning:

3.1 Means of dismounting large reactor components from the reactor building.

3.2 Supplementing and completing drawings for conditioning facilities.

Possible sites for a conditioning facility on the reactor premises.

3.3 Investigation of thermal separation methods for application in nuclear facilities. Selection of mechanical separation tools.

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General Analysis:

3.4 Dismounting, crushing - steel/concrete
    Conditioning of large components
    Auxiliary systems

3.5 Means of reactor component transportation from nuclear power station

3.6 Waste conditioning

3.7 Setup of time schedule for decommissioning

3.8 Special arrangement of reactor components

3.9 New reactor lines

4. Test Facilities, Computer Programs

5. Work Performed

Conditioning:

The investigations relating to 3.1 through 3.3 were performed and terminated within the period of reporting.

General Analysis:

The studies relating to 3.4 and 3.5 were performed and terminated.

Points 3.6 through 3.9 were treated although not terminated within the period of reporting.

6. Results

Conditioning Facility:

Ad 3.1 Dismounting large reactor components is feasible technically, even after an extended period of operation.

Except for the pressure vessel of new boiling water reactors, the components can be removed as integral units.

Dismounting unshielded components emitting a high dose rate is very expensive since it has to be done by remote handling and remote inspection.

Ad 3.2 A conditioning facility can be installed on the power station site of new light water reactors. Existing siding tracks can generally be used for transports from and to the facility.

Ad 3.3 Considering the sizes of components and the material thickness, thermal separation techniques should be preferably used in dismantling, above all plasma fusion cutting and autogeneous flame cutting (also combined).

For component dismounting and fragmentation of segments mechanical tools and machine tools, respectively, such as saws and hammer shears, are suitable means.

General Analysis:

Ad 3.4 Dismounting of components in the reactor building is feasible technically, but only reasonable in case of decommissioning.

If a component has to be replaced, external dismantling, e.g. in a conditioning facility, should be preferred so that the installation of the replaced components is not delayed.

Auxiliary systems such as a protective tent, hood, filter, etc. are necessary in both cases in order to avoid propagation of contamination.

Ad 3.5 Due to the high dose rate to be expected after longterm operation (20 - 30 years) the primary components must be shielded during transport. This increases the weight and sizes so that only medium-sized components (e.G. primary pump) can be transported as integral units.

7. Plans for Future Work

Completion of points 3.6 through 3.9.

8. Relations to Other Tasks

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9. Literature

G.W. Köhler, J. Weppner, M. Salaske, W. Hennhöfer, Studienentwürfe für eine Anlage zur Konditionierung schwerer aktiver Reaktorkomponenten, Trenn- und Zerlegemethoden sowie ferngesteuerte Arbeitsgeräte.
July 1976 (not published)

J. Wepnner, G.W. Köhler,
Die Konditionierung von Reaktor-Großkomponenten
- Ein Überblick -
December 1976 (not published)

10. Availability of Reports

1711

Classification: 17.3	
Title 1 (Original Language): Quantitative Mengenstromanalyse für radioaktive Abfälle (RS 220 - II.6.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: NIS - Hanau
Title 2 (English): Quantitative Analysis of Radioactive Waste	Project Leader: A. Gasch
Initiated (Date): 1. 10. 1976 Status: Continuing Completed (Date): 30. 6. 1977 Last Updating (Date): 31.12.1976	

1. General Aim

The radioactive waste from dismantling of nuclear power plants causes several problems. In order to solve these problems, it is necessary to analyse the radioactive waste. The results will allow to specify the requirements for handling, transport or burrial of the waste.

2. Particular Objectives

Within the framework of an overall project in the view of conditioning heavy reactor-components from the dismantling of nuclear power plants, this study should deliver the necessary data about the quantity and quality of the waste. A longterm summary analysis will be carried out for the BRD. The waste shall be classified considering different points of views. Waste quantities as function of time are to be determined.

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3. **Research Program**

3.1 Establishment of a system of classifying criterias for the waste from dismantling nuclear power plants

3.2 Determination of quantity of bulky activated waste

3.3 Interpretation of operating experience in nuclear power plants

3.4 Determination of quantity of bulky contaminated waste

3.5 Decontamination of bulky waste from 3.4

3.6 Secondary waste from the dismantling of nuclear power plants

3.7 Rad waste quantity after LOCA

4. **Experimental Facilities, Computer Codes**

re 3.2 The NIS Computer Code AKAT II calculates the activation of reactor components resulting from thermal, epithermal, and fast neutron flux. Radioisotopes produced by various reactions are added by the machine.

Only the following input data are necessary: the activatable reactor components with specific neutron flux, the materials of reactor components with their composition and their exposition time. The nuclidespecific data of the activatable materials and produced radioisotopes are stored in a library (file).

5. **Progress to Date**

re 3.1 A system of classifying criterias has been created. Several classifying systems had been tested with respect to applicability for classification of dismantling waste. The classification of dismantling rad waste follows the recommendations of the IAEA.

re 3.2 The base for calculation of activities has been formed

re 3.4 Computations of the contaminated bulky waste quantities have been started

6. Results

re 3.1 The dismantling waste must be handled, transported and burried. Conforming with the resulting necessity the waste will be classified corresponding to their physical conditions. Categories within this classification follow specific activities which is opposite to the recommendations of the IAEA. Furthermore chemical aggression and toxity are considered.

re 3.2 Definit results are not available

re 3.4 Definit results are not available

7. Next Steps

re 3.2 The work will be finished by April

re 3.3 The interpretation of operating experiences in nuclear power plants is planned for first quarter of 1977

re 3.4 The work will be finished by April

re 3.5 The work will start later on

re 3.6 The work will start later on

re 3.7 The work will start later on

8. Relation to other Projects

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9. References

IAEA "Standardization of Radioactive Waste Categories"
Technical Reports Series No. 101, 1970

BMI-Entwurf vom 16. 11. 1976
"Verordnung über den Schutz von Schäden durch ionisierende Strahlen" (Strahlenschutzverordnung)

Staatsanzeiger für das Land Hessen
"Bedingungen über die Lagerung von schwachaktiven Abfällen im Salzbergwerk Asse"
Nr. 18, Seite 202
H. Ramdohr, Kerntechnik 11. Jahrgang 1969, Nr. 5

10. Availability of Reports

18. FUEL CYCLE

1715

Classification: 18	
Title 1 (Original Language): Kritikalitäts-Studien (ATT 085 A - I.4.1 , Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMI ORGANIZATION: LRA, Garching
Title 2 (English): Criticality Studies	Project Leader: Prof. Dr. Birkhofer, W. Thomas
Initiated (Date):	Completed (Date):
Status: continuing	Last Updating (Date): December 1976

1. General Aim

Problems of criticality safety for fissile material outside reactors are investigated and solved. Safety criteria, critical and safe parameters will be established for handling of fissile material in enrichment procedures, fuel manufacturing, reprocessing, storage and transportation. Risks and consequences of criticality accidents for the environment and personnel will be evaluated.

2. Particular Objectives

Special problems of neutron interaction and isolation will be treated. The influence of concrete structures of hot cells on neutron reflection will be considered. Problems of homogeneous and heterogeneous neutron poisoning are investigated especially for reprocessing facilities. Assessments of radiological consequences of criticality accidents or other severe accidents in plutonium and uranium fabrication facilities are included.

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3. Experimental Facilities and Program

No experimental work.

4. Project Status

4.1 Progress to Date

During last year criticality data for homogeneous mixtures of ^{233}U and thorium including homogeneous poisons have been computed. Further calculations have been performed for plutoniumtributyl phosphate mixtures and plutonium-polyethylene systems. The separation and reflection effects of various concrete reflectors with and without nuclear absorbers have been investigated in a series of calculations by Monte-Carlo technique. For all cases slab geometry has been assumed. Similar calculations for storage assemblies of plutonium nitrate cylinders in concrete vaults are under way. A study for use of boron or hafnium plates as nuclear poison has been done for an extraction column in the first cycle of an oxide fuel reprocessing plant. Overview calculations have been performed for the storage of spent PWR fuel bundles in poisoned casks.

In the field of risk analysis of the nuclear fuel cycle a theoretical analysis to predict the power and time dependence of criticality accidents has been issued. Further implementations in the computer code RADCA for calculation of the consequences of criticality accidents have been incorporated.

4.2 Essential Results

A new supplement has been issued to the Criticality Handbook.

5. Next Steps

Completion of our Criticality Handbook. A safety study is under way for plutonium recycle in LWR, especially for fuel manufacturing of uranium elements and U-Pu-mixed oxide fuel.

6. Relation with Other Projects

Relations exist with design studies for a new reprocessing plant performed at INR, Karlsruhe. Similar work is under way in the UK, France and USA, as documented in:

- Handbook of Criticality Data, J.H. Chalmers et al., AHSB 1965/7
- Guide de Criticité, CEA, CEA-R-3114, 1967
- Criticality Handbook Vol. I-III, R.D. Carter et al., ARH 600, 1968/72.

7. Reference Documents

/1/
W. Thomas
Plattenvergiftung als Kritikalitätskontrolle im ersten Extraktionszyklus einer Wiederaufarbeitungsanlage
MRR 158, Mai 1976

/2/
W. Heinicke, W. Thomas
Kritikalitätskontrolle durch heterogene Vergiftung bei der Lagerung von BE-Bündeln in Wasserbecken
atw 8, August 1976, S. 411-412

/3/
W. Heinicke, Ch. Müller, W. Thomas, R. Warnemünde, W. Weber
Handbuch zur Kritikalität, 7. Teillieferung
LRA Garching, November 1976

8. Degree of Availability

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

182-11-10 Classification : 18

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Titre	Pays : FRANCE
Traitement des effluents réacteurs.	Organisme directeur : CEA/DSN
Titre (anglais)	Organisme exécuteur : CEA-DSN-SETSSR
Processing effluents in nuclear reactors.	Responsable : J.GUIRLET
Date de démarrage : 1/1/76 Date prévue d'achèvement : 31/12/81 Etat actuel : en cours Dernière mise à jour : 1/4/77	Scientifiques : J.ROUX

Objectif général :

Etude générale des traitements et installations concernant les effluents liquides gazeux et le conditionnement des déchets solides du point de vue :
- de la nature des procédés mis en oeuvre
- des modes de fonctionnement
- des résultats attendus

Objectifs particuliers :

Etude particulière du fonctionnement des installations mises en place dans le réacteur en service, en cours de mise en exploitation ou en projet dans les domaines précités : nature des installations et résultats d'exploitation, particulièrement du point de vue des performances des installations de traitement des effluents liquides et du conditionnement des déchets.

Etat de l'étude :

1) Avancement à ce jour :

Actuellement analyse des informations contenues dans les documents d'analyse de sûreté des réacteurs en cours de démarrage et en projet.

Prochaines étapes :

Enseignement tiré des installations de traitement des réacteurs de Tihange et de Fessenheim.

Relation avec d'autres études :

Application aux analyses de sûreté
Rejets des installations.

.../...

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<u>Documents de référence</u> :

Rapports de sûreté
Notices d'exploitation.

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TITLE 1 (original language) Valutazione di proprietà chimiche e fisiche di rifiuti radioattivi solidificati	Classification 18
TITLE 2 (english) Evaluation of chemical and physical properties of active solidified wastes.	Country: ITALY Sponsor: CNEN Organisation: CNEN
Date initiated 1975 Date completed 1977 Last updating June 1976	Project Leader F. GERA

Description : The purpose of the project is to control the main chemical and physical properties of active solidified wastes to reduce possible 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 Sponsor: CNEN Organisation: (*)
Date initiated 1974 (present phase) Date completed In progress Last updating June 1976	Project Leader 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) Ricerche e sviluppo su un contenitore di trasporto.	Classification : 18
TITLE 2 (english) Research and development of a fuel cask.	Country: ITALY Sponsor: C.A.M.E.N. Organisation: C.A.M.E.N. - UNIVERSITY OF PISA
Date initiated : 1975 Date completed ; In progress Last updating June 1976	Project Leader : G. SARNO (CAMEN) 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) Sicurezza del trasporto del materiale radioattivo.	Classification 18
Title 2 (English) Safety Problems in Design of Packaging and Transportation of Radioactive Materials.	Country ITALY Sponsor CNEN-CNR Organisation University of Pisa
Date initiated 1968 Date completed 1980 Last updating May 1977	Project Leader 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.

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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) Ricerche e sviluppo su un contenitore di trasporto.	Classification 18
Title 2 (English) Research and development of a fuel cask.	Country ITALY Sponsor: C.A.M.E.N. Organisation C.A.M.E.N. - UNIVERSITY OF PISA
Date initiated : 1975 Date completed: In progress Last updating : April 1977	Project Leader G. SARNO (CAMEN) 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)	Country ITALY
Evaluation of chemical and physical properties of radioactive solidified wastes	Sponsor CNEN Organisation CNEN
Date initiated 1975 Date completed 1978 Last updating 1977	Project Leader 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.

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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 **Organisation** (°)
Date initiated I974 (present phase) **Date completed** In progress **Last updating** April I977	**Project Leader** 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)

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Classification 18	
Title 1 Ontwikkeling van een computercode voor het schrijven van een éénduidige en optimale herladingsprocedure	Country The Netherlands Organization KEMA
Title 2 Development of a computer code which writes an unambiguous and optimum reload procedure	Projectleader K.P. Termaat

1. 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 element movements can be included.

2. 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.

3. Experimental facilities

Not applicable.

4. Project status

The programme is in the development status.

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5. Next steps

Complete development and test with previous handwritten reload procedures.

6. Relation with other projects

Physics reload scheme, fuel inspection programme, fuel test programme.

7. Reference documents

To be made.

8. Degree of availability

Through the organization KEMA.

19 ECONOMICS OF SAFETY

20. OTHER TOPICS

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Classification: 20	
Title 1 (Original Language): HDR-Sicherheitsprogramm (RS 0123 A + B - III, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GfK, Karlsruhe
Title 2 (English): HDR Safety Program	Project Leader: Müller-Dietsche
Initiated (Date): April 1, 1974 Status: continuing Completed (Date): 1981 Last Updating (Date): December 1976	

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

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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 100 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 fully 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 Testings

Non-destructive testings 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 generation 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, thermoshock, 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 last few highly radioactively contaminated components were removed from the plant and most of the work associated with the modification for experimental operation was finished. The test loop with an electric boiler to simulate conditions close to the operating conditions (310°C, 110 bar) was installed. In addition, a central measured data acquisition system for 450 fast and 150 slow measuring stations has been ordered.

Specific Project 1000: Non-destructive Tests

To assess the actual state of the components under study in an accurate way, a first fully volumetric automatic ultrasonic test of the RPV was carried out with a special type of manipulator built for this purpose. Areas specially selected of the RPV pipe connections, top lid, welds, defect areas, transitions and parts of the piping system were additionally tested manually (manual ultrasonic test, manual tandem test, radiography).
In a pressure test (143 bar, 50° C) sound emission measurement was used on the RPV and the piping system. It furnished various noise readings whose evaluation, however, indicates that leakages and other defect noises had been detected and that there had not been any changes in the RPV.

Specific Project 2000: RPV and Piping Studies

The first materials data and the structural conditions were determined in material studies and metallographic examinations of drill cores taken from the pressure vessel and of specimens of the piping system. It turned out that the strength and ductility properties of the materials in the base material and the welds satisfied the requirements as specified and did not show any weakening defects.
An experimental stress analysis carried out on the pressure vessel in the "cold" pressure test (143 bar, 50° C) revealed peak stresses and stress distributions on the RPV and certain regions of the piping system. A first comparison between measurement and calculation of the stresses determined on a large RPV-nozzle showed good agreement.

Specific Project 3000: Blowdown Experiments

Preliminary calculations performed as a basis for the experiments were carried out on the RPV with respect to the following criteria:

- Development of pressure relief waves under various boundary conditions.
- Load acting upon the internals as a function of the length and diameter of the pipe connections.
- Influence of the initial pressure and temperature distributions upon the load forces.

For the containment, the factors calculated were

- the differential pressure loads acting upon the structures,
- the pressure and temperature development to be expected.

Fabrication of the test components, especially of the RPV internals and the valves, was continued. The extensive measuring gear is being purchased.

Specific Project 4000: Earthquake Studies

The experiments carried out with low excitation and various types of excitation, i.e.,

- shaker, snapback, explosion

were evaluated. A first comparison with vibration calculations showed differences between the measurement and calculation, both in the eigenfrequencies determined and in the amplitudes.

Extensive calculations of the vibration behavior under the impact of low excitation compared with measurements have been initiated by various institutions.

Specific Project 5000: Leak Rate Studies

In the period April/May/June 1976 leak rate measurements were carried out on the cold plant. Especially the influence of the sequence of pressure steps, the action of the air trapped in the concrete and the critical swelling pressure were investigated.

4.2 Main Results

The first automatic ultrasonic test of the RPV, the additional manual ultrasonic tests and the material studies indicated that the pressure vessel and the piping system of the HDR did not show any sizable defects or cracks before the start of the blowdown experiments. A de-

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tailed stress analysis at 146 bar, 50° C revealed no strains; in the elastic region the measurement and the finite element calculation showed good agreement.

Here are some of the results of the leak rate studies:

- Checking the leak rate at 0.5 bar is sufficient for the recurrent tests. Extrapolation of the leak rate from the rest pressure to the design pressure is possible. This, e.g., does not apply for a test pressure of 0.03 bar or 1.5 bar.
- The leak rate values measured during the pressure surge phase are on the conservative side.
- A period of assessment of 4 - 6 hours is not sufficient to allow the in- and outgassing effects in the concrete to be neglected. Also after the end of the transient time there may well be in- and outgassing effects accompanying major changes in the absolute pressure.

5. Next Steps

The test loop will be tried out. Afterwards the whole measuring setup will be installed for the first blowdowns, which are planned for mid-1977. They will include measurements of pressure, temperature, strain, displacement, density, mass flow, water entrainment, water separation, and heat transfer.

Important points of further investigations include

- experimental stress analysis (Spec. Proj. 2000) in the "hot" pressure test (310° C, 110 bar) for RPV and pipings.
- Leak rate measurements on the containment in the "hot" state (SP 5000).
- First Blowdown experiments with steam isolation valves of NW 350 dia. (SP 3000).
- Theoretical studies of the vibration behavior of the plant under the impact of low excitations compared with measurements.

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 50

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 Spectroxcopy 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).

8. Degree of Availability

Unrestricted distribution.

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Table 1 Main Plant Data

Safety Containment:	Dimensions	
Inside height	m	60
Inside diameter	m	20
Test pressure	atm.g.	7,12
Design pressure: Overpressure	atm.g.	5,60
Underpressure	mm water	350
Design temperature	°C	155
Wall thickness of cylindrical part	mm	30
Material specification of steel cladding	–	FB 50 S fine grained steel
Free containment volume	m^3	10,853

Pressure Vessel:	Dimensions	RDB	HDU	SDU	UK
Number	–	1	1	1	2
Inside diameter	mm	2960	1765	1978/1508	700
Overall height	mm	12 700	14 110	10 920	4365
Wall thickness (average)	mm	112	45	61/46	23
Design pressure	atm.g.	110	110	110	110
Design temperature	°C	360	550	400	320
Material	–	23 NiMoCr 36	BHW 38	BHW 38	BHW 39 S
Flow: primary	t/h	–	170	170	170
secondary	t/h	–	130	130	130

Water Circuits:	Dimensions	Primary circuit between RDB and HDU	Primary circuit between HDU and SDU	Primary circuit between SDU and RDB	Secondary circuit up to reducer station	Recirculation loop
Design pressure	atm.g.	110	110	112	110	112
Design temperature	°C	550	400	320	400	320
Material	–	No. 4961	No. 4550	No. 4550	No. 7335	No. 4550
Inside diameter	mm	250 – 300	250	200	300 – 350	350 – 450

Notes:

Reactor pressure vessel = RDB
Superheated steam converter = HDU
Saturated steam converter = SDU
Subcooler = UK

Classification: 20	
Title 1 (Original Language): Auswirkungen von Kühltürmen großer Kernkraftwerke auf ihre Umgebung (PNS 4152 - III, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** **ORGANIZATION:** PNS/GfK
Title 2 (Engli... Environmental Impact of Cooling Towers of Large Nuclear Power Plants	**Project Leader:** K. Nester
Initiated (Date): 1.1.1974 **Status:** Continuing **Completed (Date):** 31.12.1976 **Last Updating (Date):** December 1976	

1. General Aim

The aim is pursued under this task to study the impact of emissions from large cooling towers on the environment.

2. Particular Objectives

The rise and the dispersion of heat, humidity and droplet emission from cooling towers as a function of ambient conditions should be calculated.

3. Research Program

3.1 Development of a three-dimensional computer program for the calculation of the distributions of temperature, specific humidity, vertical velocity, cloud and rain droplets in the plume of cooling towers.

3.2 Verification of the computer program.

3.3 Application of the computer program to a case study.

4. Experimental Facilities, Computer Codes

Experiments have been performed by DFVLR (compare 7)

Computer Codes (compare 3.1)

5. Progress to Date

With the help of the WALKÜRE program allowing to calculate the rise and diffusion of plumes from cooling towers a number of computations were performed for complex meteorological environmental conditions. The model used in the determination of the

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dispersion coefficient with induced turbulence was improved once more. It is now based on the turbulent kinetic energy and rate of dissipation. This required two further partial differential equations to be solved so that their number has been increased to seven. Moreover, the overlapping of two cooling tower plumes was calculated for the first time.

6. Results

The results of computations performed for complex meteorological conditions have shown that the horizontal distributions of the relevant parameters normal to the direction of transport clearly distinguish from each other in the individual layers. The influence exerted by inversions gets particularly noticeable. Such structures cannot be treated by one-dimentsional models. In some extreme cases the geometry of the visible plume completely differs from that which would be obtained with one-dimensional models. In case of overlapping of two cooling tower plumes the distributions of humidity, temperature and vertical velocity indicate the plumes from the individual cooling towers, even at distances of some kilometers from the sources, although the visible plumes can combine after 200 m already. Use of the new model for the dispersion coefficients also allows a more realistic calculation of their distribution. Overlapping of the plumes of several sources is now possible without any restrictions.

7. Next Steps

The program will be terminated on December 31, 1976 with the completion of the WALKÜRE computer program. Since the results of measurement campaigns relating to the cooling tower plumes of the Neurath and Meppen power stations were not available before mid-December 1976, detailed "calibration" was not possible. The verification (3.2) as well as the application (3.3) of the WALKÜRE program used to calculate overlappings of cooling tower plumes will be carried out in 1977 within the framework of the Upper Rhine Valley Project initiated by the Federal Office of Environmental Protection (Bundesumweltamt).

8. Relation with Other Projects

The Project will be continued in the Upper Rhine Valley Project (compare 7).

9. References

Report KFK 2262 (1976) p 152 (German)
Report KFK 2266 (1976) p 115 (German)

10. Degree of Availability of the Reports

Unrestricted distribution.

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Classification: 20	
Title 1 (Original Language): Untersuchung des Wärmeaustauschs Fluß/Atmosphäre am Beispiel des Rheins unterhalb des Kernkraftwerks Philippsburg als Beitrag zu Abkühlungsmodellen (PNS 4151 - III, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** **ORGANIZATION:** GfK
Title 2 (English): Investigation of the Heat Exchange River/Atmosphere at the Rhine River Downstream of the Philippsburg Nuclear Power Station. A Contribution to Heat Loss Models	**Project Leader:** Dr.W.Schikarski DP.H. Sauter
Initiated (Date): January 1972 **Status:** Finished **Completed (Date):** December 1976 **Last Updating (Date):** December 1976	

1. General Aim

Measurements of heat exchange over the river surface and of meteorological and hydrological parameters. Theoretical modelling.

2. Particular Objectives

Direct measurement of evaporation and convection by means of eddy correlation method. Tests of available heat exchange models, calculations on total heat exchange, statistics, classification.

3. Research Program

See above

4. Experimental Facilities, Computer Codes

Measuring station in the Rhine River. Evaluation codes.

5. Progress to Date

Station has been in the run mode with few interrupts due to repair. Two longer periods of breakdown occured in August/September for 6 weeks due to lightning damage and in December for 4 weeks due to an unindentified top event. First evaluations and special survey calculations were performed.

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6. Results

A final report (KFK-2374) summarizes the activities and results of the program during the leadership of PNS.

7. Next Steps

In order to meet the original plan of measuring over nearly naturally temperatured water and over thermal stressed water resulting from the Philippsburg Nuclear Power Station's cooling maintenance, the programm will continue for one year under the institute's own control. Further evaluations are in progress.

8. Relation with Other Projects

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9. References

KFK-2374 "Die Meßstation Rheinhausen zur Untersuchung des Wärmetransports aus Fließgewässern", Januar 1977; G. Hoffmann, H. Sauter, W. Schikarski

10. Degree of Availabilities of the Reports

Unrestricted distribution.

Classification: 20	
Title 1 (Original Language): Programmentwicklung zur mehrdimensionalen Kontinuums-mechanik (RS 139 - III, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** INTERATOM Berg.Gladbach 1
Title 2 (English): Code development in multi-dimensional continuum mechanics	**Project Leader:** Banasch
Initiated (Date): 1.9.74 **Completed (Date):** open **Status:** Continuing **Last Updating (Date):** 31.12.1976	

1. General aim

The aim of the project is the development of methods in the field of multi-dimensional continuum mechanics in order to describe

- shock and pressure wave propagation together with one and two-phase large scale fluid flow in a general geometry (compressible as well as non-compressible viscous hydrodynamics in an Eulerian coordinate system)
- and coupled with that the elastic-plastic deformation, penetration and failure of complex structures (in a Lagrangian coordinate system).

2. Particular objectives

The experimental verification of all important aspects of the methods developed is a particular objective of the project:

- evaluation of available tests
- participation in the vessel explosion programme being performed at Euratom/Ispra (pre- and post-shot theoretical analysis)

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3. Research programme
   - literature survey of available methods for solving problems in continuum mechanics
   - development of procedures for the hydrodynamics calculation in discrete Eulerian meshes
   - description of the solid structures in a Lagrangian system (finite elements, thin-shell theory, multi-axial stress and strain conditions, general materials laws)
   - coupling of the Eulerian and Lagrangian procedures for solving the integral hydrodynamic and structural dynamics problem
   - development of a theory describing the high pressure gas bubble through an Eulerian coordinate system
   - inclusion of energy dissipation due to friction (e.g. flow through perforated plates) into the description of hydrodynamics
   - verification of the theoretical methods by analysis of appropriate tests.

4. Experimental Facilities Computer Codes

Several parts of the main Code KOELSCH (KOntinuumsmechanik mit Euler-Lagrangekoordinaten und SCHalentheorie) have been written and tested using the records of several vessel explosion experiments.
The Lagrangian INTERATOM code ARES is used for the participation in the Code Validation Programme at JRC Ispra.

5. Progress to Date

Since the participants on the project during the year 1976 have mainly been working on other projects, the progress is unsatisfactory.

An improved method for the calculation of an expanding explosion bubble was tested using the records of an explosion experiment (Belgonucleair Experiment E 114). The results were good.
The twodimensional computational hydrodynamics were extended so that problems with free surfaces and water hammer could be treated.

Shell theory and the one-dimensional version of the Euler-Lagrange-hydrodynamics were coupled, so that shock waves in tubes with elastic and plastic deformation could be calculated.
The work for the coupling of shell theory and twodimensional Euler Lagrange hydrodynamics was started.
The participation in the Code-Validation (COVA)-Programme in JRC Ispra was initiated.
Several explosion experiments (IT4-6) with low density charges were successfully performed in thick-walled water-filled vessels and the results compared with the experiment IT1-3, which used high explosives.

A new coating of the low-density charge was used for the repetition of the experiment IT5.
The pressure impulses were 25-3o % higher than in the previous shots. It is suspected that the charge didnot burn completely in the original experiments IT5 and IT6 due to moisture entering into the charge. The experiments IT5 and IT6 will probably be repeated once more with the new coating of the charge . The first experiment with a deformable shell (IT7) was executed but the charge probably didnot burn correctly . The experiment was therefore repeated on October, 1976. Much work has been spent on the calibration of the high pressure transducers. A comparison with the transducers from the collaborating British establishments showed systematic differences. They are being further investigated. An analysis showed that the frequences of the data recording apparates have an influence on the high pressure peaks. These peaks should therefore not be considered when a detailed comparison of calculations and experiments is made.
An improved version of ARES (Interatom version, may 1976) was converted from CDC-Cyber to IBM 37o and installed in Ispra. The experiments IT1-3 and IT5 were calculated with this code.

6. Results

The main program will combine a set of largely independent program modules. These modules are being written and tested.

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Programs for one- and twodimensional Euler-Lagrangian hydrodynamics and three-dimensional Eulerian hydrodynamics have been written and are being tested.
Further, a program for the description of the expansion of a high pressure bubble is under development. Polar coordinates in combination with the Eulerian grid are used here. Theoretical work for the development of a program, that calculates the motion of structures under influence of external and internal stresses, has nearly been completed. This program will use shell theory in combination with finite difference methods.
The cooperation with the code validation programme in JRC-Ispra has worked out well and the work is progressing steadily.

7. Next Steps

The work for the coupling of twodimensional Euler-Lagrange hydrodynamics with shell theory will be resumed. Work on the complete coupling of the subcodes into the main code KOELSCH will be initiated. The cooperation with the COVA-programme in Ispra will be continued. The conversion of the Overlay version of the Code ARES will be completed.

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Classification: 20	
Title 1 (Original Language): Störfallanalyse von Natrium-Wasser-Reaktionen im Dampferzeuger unter Berücksichtigung der Bildung von Zwei-Phasen - Zwei-Komponenten-Gemischen (RS 140-III, Jahresbericht A 76)	**COUNTRY:** BRD **SPONSOR:** BMFT **ORGANIZATION:** INTERATOM
Title 2 (English): Accident Analysis of Sodium-Water Reactions in Steam Generators and the Associated Generation of Two-Phase, Two-Component Mixtures	**Project Leader:** H. Banasch
Initiated (Date): 1.9.1974 **Completed (Date):** 30.9.1977 **Status:** Continuing **Last Updating (Date):** 31.12.1976	

1. General Aim

The object of this work is to define improvements in the methods of accident analysis of Sodium Water reactions caused by water tube failures in steam generators of a sodium cooled fast breeder reactor.

2. Particular Objectives

A detailed examination of the experiments carried out by Interatom on Sodium Water reactions has shown that the removal of sodium results in a hydrogen saturation containing very small sodium droplets.

Furthermore, pressure losses will increase as a result of the removal of sodium and these losses will depend to a great extent upon the degree of acceleration,(they increase with the acceleration).

As a consequence, the residual sodium droplets have the effect of cooling the gas because of the rapid diffusion of hydrogen into the sodium droplets resulting in a considerable variation of behaviour when compared with the computer model by Salmon.

The specific task, therefore, is to develop a computer program which will describe the phenomena of an accident which can occur in a steam generator system.

3. Research Program

Evaluation of existing research material based upon existing programs and subroutines currently in development.

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4. Experimental Facilities - Computer Codes

Computer program PARA III enables the calculation of certain parameters by the comparison of experimental and calculated results.

From this it has been demonstrated that calculations carried out with the computer code POOL (version GT) were too strongly influenced by the ideas formulated by Salmon.

For example, hydrogen was considered to be isothermal and this was the assumption on which Salmon based his proposals and conclusions. Iteration problems were experienced in the program POOL, version GT when the gas had reached a velocity which was close to the critical velocity. It could be that three solutions were found instead of one.

Consultations with INR Karlsruhe provided some useful information. Only one code is known - the code KACHINA of Los Alamos - in which the Two Component - Two Phase flow can be calculated dynamically. In it changes in temperature, resulting from changes in pressure and in the kinetic energy of the gas etc. are included in the iteration.This may have a considerable effect on the procedure.

In addition, experiments made relating to small leaks have shown that oscillations having very large amplitudes were found in the computer program but not in the experiment.

Further studies have shown that this problem can only be solved by using the TREGONNING hypothesis. According to this hypothesis there is an immediate reaction of the water entering the reaction zone. On the other hand a water-steam mixture would show a slower reaction.

The kinetic gas theory stipulates that the molecules collide with the droplets of sodium. By penetrating into the reaction zone, part of the water may evaporate and will be influenced by the pressure of the hydrogen to produce a dynamic effect.

5. Progress of Work to date

6. Results

Important improvements have been made in the subroutines of PARA III during the period under review and a further subroutine has been developed which takes into account the influence of acceleration on wall film thickness.

In addition, another subroutine is being written which calculates the physical data if another parameter value should be selected for the iteration.

Experiments have shown that this iteration cannot be carried out as formerly (isothermal calculations). Earlier pressure losses were changed and then checked against the differences between the assumed and the calculated pressure losses to confirm whether the losses were small enough. This procedure must be changed to the average speed of a zone.

7. Further Steps

The respective subroutines have yet to be developed and tested. Then the convergence must be checked.

8. Relation to other Projects

Certain subroutines are to be brought into such a form that they can be used as subroutines in other computer programs dealing with accident analysis.

9. References

- Salmon, M.A. McDonald, I.S., Effects of tube leaks in sodium heated steam generators, NAA - SR- 8140, Atomic International, Canoga Park, 15th April 1963.

- Dumm, K., Dr. Mausbeck, H., Schnitker, W., Status of sodium - water-reaction test work at INTERATOM, ANL - 7520, part I, p. 374 - 383

- Schnitker W., The peculiarities of the reaction of sodium and water in vessels, Atomkernenergie, Bd 24, 1974/1975, p. 225 - 232.

- Amsden, A., Harlow., F., KACHINA : AN EULERIAN COMPUTER PROGRAM FOR MULTIFIELD FLUID FLOWS, Los Alamos, LA-5680

- Tregonning, K., Mathematical modelling techniques for large scale sodium-water reactions in heat exchangers, BNES, Liquid Alkali Metals Conference, Nottingham, 4 - 6th April 1973, Proceedings, p. 115 - 122.

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10. Degree of Availability of the Reports

The reference documents are available at:

Zentralstelle für Atomkernenergie-Dokumentation
(ZAED)
Kernforschungszentrum

7514 Eggenstein - Leopoldshafen

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Classification: 20	
Title 1 (Original Language): Entwicklung fernbedienter Ultraschallprüftechnik für Schnellbrutreaktoren (RS 143-III, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: INTERATOM
Title 2 (English): Development of a remote controlled ultrasonic testing device for fast breeder reactors	Project Leader: H. Banasch
Initiated (Date): 01.09.1974 Completed (Date): 31.12.1976 Status: Completed Last Updating (Date): December 1976	

1. General Aim

The object of this project is to develop methods of ultrasonic testing welds in austenitic steels and of the associated remotely controlled mechanism for moving the test probes to inspect the reactor tank externally.

This development will be carried out with special reference to the environmental conditions which are to be expected during inservice inspection of the reactor tank of a sodium cooled Fast Breeder Nuclear Power Plant. Thus the test system will operate in an environment where:

- the temperature is 250° C at the reactor tank wall and surrounding atmosphere,
- the radiation level is ca. 10^4 rad/h,
- the atmosphere is Nitrogen
- the reactor is loaded.

It is required to develop and optimise

- ultrasonic (US) test probes (transduceres), to detect defects in

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welds up to 60 mm thick, both longitudinally and transversally,
- a remotely controlled manipulator to carry the US test probes,
- system for the acquisition of the US signal data and handling of the data,
- a medium for coupling the US transducer to the test object.

2. Work performed and results attained

2.1 Development of ultrasonic probes for austenitic stainless steel welds

The fundamentals of the transmitting-detecting techniques for inspecting welds in austenitic steel were optimised. In this respect, at the beginning of the year under review, a series of prototype transducers was built with various ranges of penetration of the reactor tank wall.

This transducer series was subsequently extensively tested to detect artificial and natural defects of welded steel plate. Good results were achieved [1]. Since the plane of orientation of some defects diverges strongly from the side of the weld, it was necessary to design and build transducers with small insonation angle.

This technique also was used for testing transversal defects. Here a series of transducers,was built having an increasingly larger distance between probe and weld. It was possible with these transducers to detect transversal defects in the region near to the transducer (3 mm dia flatbottom holes). Testing of the surface away from the transducer is still not satisfactory. Tests of curved components (for instance pipes) were performed with good results.

Extensive experiments were also performed under high temperature. A test facility with heating furnace and manipulator was set up for measurements at temperatures up to 250°C. By means of this measurements of the velocity of sound as a function of temperature were made with higher accuracy using the correlation method. Coupling medium experiments were made before these measurements. A specific Polyphenyl Ether and two other coupling media satisfy the requirement for a low evaporation rate. Measurements of sound attenuation as a function of temperature for these particular substances showed satisfactory results.

In contrast with the transmitter-receiver transducer technique the use of focused beams brings an improvement of signal to noise ratio.This could be verified by a comparison between the SEL-3 transducer and a newly developed focused angle beam transducer for longitudinal waves used on the same test plate. However, due to the narrow sound beam the probability of detecting a defect is significantly lower than with the transmitter-receiver transducers. It will therefore be useful to carry on with these experiments with the object of optimisation in so far as,for instance,the transmitter-receiver transducers could be used for defect detection and the focused transducers for defect measurements.

Further work program:

The next steps to be performed within the frame of the following project, RS 244, will be as follows:

- examination of ultrasonic properties of specimen and transducer materials,
- further- and new-development of ultrasonic test methods to increase the defect detection probability,
- conception and manufacture of ultrasonic transducers for testing in the laboratory and at the sodium test facility.

2.2 Development and construction of the transport mechanism for the test system

The transport mechanism has to perform the following tasks:

- transfer and positioning of the ultrasonic test system,
- garantee the necessary pressing force and mechanical support of the ultrasonic transducers for the correct coupling,
- among others, visual control of the test using a TV camera.

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The transport mechanism will be brought to the test position via the inspection ducts. The whole test area at the perimeter of the reactor tank is divided into 12 sections. In this way the transport mechanism has to scan a lateral area of approx. 1 m in each direction. An important requirement for the ultrasonic inservice test will be exactly reproducible positioning of the moving mechanism. The development of the control for the manipulator will, however, not be a part of this project.

Because of the different geometrical conditions in the gap between reactor tank and guard vessel at least three carriages are required. In the time period under review work was mainly concentrated on the carriage which is designed for testing the cylindrical portion of the reactor tank. This design covers the horizontal transport mechanism and support for the test system.

Further work program:

Continuation of this project will be the following project RS 254, necessitating the following tasks:

- continuing with the design of test system for the cylindrical portion of the tank followed by the preparation of specifications for manufacture,
- finalisation of the design of the support for the test system followed by the preparation of specifications for manufacture,
- design and development of ancillary equipment for testing the portion of the tank above the s-shaped curve of the guidrails (in the vicinity of flanges),
- development and construction of a test system to inspect welds as joints between pipes and nozzles,
- development and construction of a curved support for the test system to facilitate inspection of welds at nozzle-piping joints,
- construction and development of ancillary equipment to inspect the bowl shaped bottom of the reactor tank,
- design of a system for feeding and removing the coupling medium, including development, construction and preparation of specifications for manufacture.

2.3 Setting up of a sodium test facility for the whole test system

The remotely controled manipulator will be guided to the test spot by means of a chain rail system. The chain rail system allows vertical movement. This system which has already been developed for another project was set up in a test facility at INTERATOM and after modefying the rails the first tests could be started. A sodium container with weld defects which will be used on this test facility has been ordered.

Further work program:
The next steps in this project will be performed within the frame of the following project RS 254. The assembly of the test container and its connection to the sodium circuit of the test facility is scheduled for the middle of 1977. Manufacturing of the first components for the remotely controled transport mechanism is scheduled for the middle of 1977. The start up of the test facility including control, data acquisition and processing is scheduled for the end of 1978.

2.4 Experiments for the selection of the coupling media and for testing of a removal system for these coupling media from the wall of the reactor tank.

Experiments on coupling media (to perform in the projects RS 244,250, 254):
The scope of this task was to test the suitability of potential coupling media. As a result of the experiments two coupling media looked promising. These substances, which among other things look attractive owing to their low evaporation-rate must, however, be checked for their sound propagation properties and for their stability against radiation.

Development and test of a removal system for the coupling medium from the wall of the reactor tank (to perform in the projects RS 244, 250,254):
The task of this installation is to remove, as completely as possible, the coupling medium after terminating the measurements at the wall

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of the reactor and to minimise loss of the coupling medium. A prototype support for the test system was manufactured from the design developed and was tested under realistic conditions. Promising results have been attained through these experiments.

References

[1] Ultrasonic testing of the SNR 300-testwelding with natural defects, Bundesanstalt für Materialprüfung, Berlin, 1976, author: Kuhlow, Römer, Matthis.

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Classification: 20.	
Title 1 (Original Language): Code Validation (RS 162 - III., Jahresbericht A 75)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: IRS, Köln
Title 2 (english): Code Validation - Design Basis Accident Modelling Theory	Project Leader: Dr. Scharfe
Initiated (Date): March 1975 Completed (Date): December 1975 Status: finished Last Updating (Date): December 1975	

1. General Aim

To understand the effects of overprompt-critical power excursions of fast reactors chemical charges are fired in well-instrumental tank models. At the same time it is tried to verify these experimental data with computer codes. Thus the reliability of the codes used for hypothetical accidents should be augmented.

This report concerns a participation at this program over a period of one year.

2. Particular Objectives

The topics of this program are the

- tank models
- the instrumentation
- charge
- analysis of tests by computer codes

3.1. Experimental Facilities

The models to be used are of loop (SNR) and pool (CFR) reactor proportions. The modelsystem is sufficient large to enable the components to be well made. The models consist of a base plate and a top plate held together with tie-bars and with the base firmly fixed to the large mass (~47 tons) of the bottom of a new bunker.

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The general aim of the instrumentation to be used in these experiments is to place it at a standardized set of measuring locations that can be used in all experiments performed for validation. Strain gauges as well as pressure transducers are mounted at the walls and the latters in the fluid too.

To check the accuracy of all transducers, they are calibrated statically and dynamically before and after every shot. Permanent elongations can be measured additional by putting a rectangular grid over the whole of the outer cylinder area.

The experiments except the first four ones will be carried out using a low density high explosive charge (LD HE) in order that the stress levels reacted in the various components will be comparable with the stress levels achieved by a UO_2 vapour explosion or UO_2 - Na interaction. The properties of the LD charge which is developed in the UK represent a compromise between the demand of a start pressure as low as possible (o,5 -1,0 kbar) and the necessity of reproducibility and the independence of confinement.

3.2. Research Program

The safety assessment of fast reactor design has emphasized the need for validated codes. Therefore it is decided to start the series of experiments with very simple rigid tanks without internals filled with water. For every test a preshot calculation is made and if necessary a postshot calculation. If all relevant details of the experiment are also represented by the calculation the next more complicated test will be performed. A time step of about six weeks for each experiment is planned. At the end of the program a complex model of a reactor tank with flexible inner and outer walls, curved bottom, neutron shields, diagrid and dipplate shall be fired and calculated.

To reach this goal a parallel development and improvement of the computer codes is necessary.

4. Project Status

4.1. Progress to Date

At the begin of the reference period the first three (of 22) experiments were already performed. These tests should be identical and were made to enable a cross check concerning the instrumentation with two other laboratories in the UK where similiar experiments are carried out. A further test with a different height of the water level should give informations about the influence of this parameter on the roof impact. By reason of the big failure rate of pressure transducers (up to 50 %) at high pressures, new transducers had to be ordered from the UK. Additional an improved type was developed and manufactured in Ispra. The new pressure gauges arrived in december, so that no further test could be performed. Analysing the results of the first three tests, they showed an excellent agreement with each other. Im comparision with the results of the calculation the agreement was good, except some few points where the discrepancies could be explained by the numerical treatment in the calculation or the insufficient measurement. Nevertheless even this simple test showed the necessity of code validation experiments.

At the present, the following codes are available at the JRC Ispra and provided for validation: REXCO - H Release 2, SURBOUM Version VD 7 and VD 8, ARES 3 and ASTARTE. The CDC-Version of ARES 3 was adapted to the IBM-machine. In early 1975 it was decided to introduce compressibility into the incompressible eulerian SURBOUM-code. This work is almost finished i.e. simple experiments with rigid walls can be treated. The possibility of the graphical output of the results is given at all available programs.

4.2. Essential Results

The first experiments with a high explosive high density charge in a rigid tank are performed. It is proved that the experimental equipment, the instrumentation and the data recording system work satisfactorily. An experimental and a theoretical test report is produced. All relevant and available european computer codes are established at the JRC Ispra and are running on the IBM-machine. The limited code development of a compressible version of SURBOUM is encouraging. The LD charge and the according facilities as well as new transducers are arrived from the UK so that the next tests are prepared.

5. Next Steps

not relevant for this contract

6. Relation with other projects

none

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7. Reference Documents

Quarterly reports in the series IRS-Forschungsberichte

Report period	April	1975	-	June 1975		
" "	July	1975	-	Sept. 1975	IRS - F - 27
" "	Oct.	1975	-	Dec. 1975	IRS - F - 28

8. Degree of Availability

No restrictions