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7. CONTAINMENT AND ASSOCIATED SYSTEMS

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.10. - 31.12.1978	7	RS 343

Vorhaben/Project Title	Land/Country: FRG
Abstellung zur USNRC zur Kooperation auf dem Containmentgebiet	Fördernde Institution/Sponsor: BMFT
Assignment to the USNRC to coordinate the containment work	Auftragnehmer/Contractor: Gesellschaft für Reaktorsicherheit (GRS) mbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.9.1978	31.8.1979	Dr. G. Mansfeld
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	31.12.1978	

1. General Aim

Cooperation within the containment field in accordance with the agreement between the USAEC and the BMFT as of the 6th of March 1974 regarding technical exchange and cooperation within Reactor Research and Development.

2. Particular Objectives

Comprehensive experiments and code developments in the containment field were carried out or will be done in the USA and in the FRG. Cooperation to date has been mainly by short visits and by the exchange of information. In particular the transmitted information is not updated or is incomplete. Therefore the particular objectives of the project are:

- To determine in detail all the theoretical and experimental work within the containment field on a bilateral basis,
- Collaboration in maintaining and developing computer codes,
- Collaboration in specifying experiments,
- Centralization of all the questions in the containment field coming from the FRG.

3. Research Program

3.1 Participation in expert meetings such as:

- Containment Code Review Group in the USA,
- Expert committee "Blowdown in Containment" in the FRG,
- Containment ad-hoc group of each country,

- Common Review Group.

A comprehensive survey of the theoretical and experimental work of each country will be gained by participation in the different meetings. Furthermore there will be the opportunity for input to proposals for containment related projects of interest to both countries.

3.2 Trips to the research laboratories and related institutions:
The close contact with the research laboratories and related institutions in the FRG and in the USA should increase the knowledge of containment problems of both sides and accelerate the information exchange. On request of the BMFT or other German institutions, which are interested in the containment field, a briefing will be provided.

3.3 Collaboration in maintaining and developing computer codes:
The close contact with the Analysis Development Branch of the USNRC and with other research institutions in the USA enables the author to look for worthwhile references for code development, if necessary or if such development can be done. In addition, it should be possible to obtain a qualified selection of the numerous computer codes.

3.4 Collaboration in maintaining and developing computer codes:
Due to the discussion with experts in the USA and in the FRG it should be possible to include additional information into the specification of containment experiments in both countries.

3.5 Collection point for questions from the FRG:
Questions from the FRG concerning the containment field should be centralized to the manager of the project. In this case it would be guaranteed that the problems will be solved as quickly as possible. Furthermore, other questions from the GRS can be handled, when it is necessary and possible.

4. Experimental Facilities, Computer Codes

The following experiments, sponsored by the BMFT, will be included in the project at this time:

- BATTELLE-Frankfurt: The phenomena occurring after the break of a coolant pipe (water or steam) of a water-cooled reactor were investigated in a multi-compartment model containment. Different room configurations were simulated.

- HDR: Within the framework of the blowdown investigations at the HDR, containment experiments will be conducted. Pressure and temperature distributions will be measured, and because there is a special interest in the phenomena, the water carryover compartments and the heat transfer to the wall and to the components in the compartments will also be measured.
- GKSS: The GKSS project covers the investigations of different phenomena occurring during a LOCA in a DAS (vent clearing, pool swell, low air content flows). The experiments of the second series - specified in extensive cooperation with the USNRC - are specialized to multivent investigations.

Furthermore, the following experiments, sponsored by the USNRC, will also be included into the project at this time:

- MIT: This program at the Massachussetts Institute of Technology studies the small scale modeling of flows which could be encountered in pressure suppression containments following a LOCA.
- UCLA: This program at the University of California at Los Angeles is an experimental study of the hydrodynamics of air and steam in water pools to provide basic experimental data for use in numerical model confirmation.
- LLL 1/5 Scale: This program at the Lawrence Livermore Laboratory was initiated to obtain experimental data for scale-model confirmation and licensing of the MARK I pressure-suppression containment design and for code assessment.

The following well-known computer codes should be reviewed as to their verification:

FRG-Codes: COFLOW, CORAN, CONDRU, DADDY, DESDUE, DRASYS, KONDAS, KSWING IV.

USA-Codes: BEACON, CONTEMPT-4, CONTEMPT-LT, COMPARE, PELE-IC.

The main parts of these codes should be reviewed for the possible use of them or parts of them in superior codes. This should lead to a reduction in the number of codes and to a definite use of the codes in the licensing procedure.

5. Progress to Date

Ad 3.1 Participation in the

- 6th Water Reactor Safety Research Information (WRSRI) Meeting in Gaithersburg, MD, USA, in a
- ACRS Fluid Dynamic Subcommittee-Meeting on MARK II LOCA and SRV pool dynamic loads and load combinations, and in
- different NRC-internal meetings.

Ad 3.2 Trips to/and discussions with/at MIT, UCLA, HDR, GKSS, BATTELLE-Frankfurt, BMFT, GRS-FB, GRS.

6. Results

The author started his work in Washington, D.C., on the 6th of November, 1978. The status of the author with the NRC in the office of Nuclear Regulatory Research (NRC-RES) in the Division of Reactor Safety Research is that of a Visiting Scientist. There is a close cooperation with the Containment Analyst of the Analysis Development Branch.

The NRC is very interested in such cooperation and has expressed his interest on many occasions. Accordingly the author has had the opportunity during the first weeks of his visit to look at the NRC-RES containment activities at MIT, UCLA, LLL and at the Idaho National Engineering Laboratory (INEL).

Some important points of interest are:

- Comprehensive verification programs are planned for the computer code PELE-IC and BEACON/MOD2A.
- The NRC-RES tries to obtain experimental results as much as possible (domestic and foreign), to have a broad basis for its verification programs.
- The NRC is very interested in the German containment tests at GKSS, BATTELLE-Frankfurt, KWU and HDR.
- The first operating license for a BWR MARK II will be issued perhaps in the second half of 1979 (Zimmer plant, Cincinnati, Ohio).

- The NRC has just started the review of the operating license for the first BWR MARK III (Grand Gulf, Mississippi).

The visits served as an initial contact and as a first hand explanation of the project for cooperation with the USNRC in the containment field. Discussions with all people confirmed their great interest in such a cooperation. They had already exchanged a lot of information.
In the frame of this project the course was set for a Common Review Group Meeting with participants from USA and FRG in March 1979 in connection with the forthcoming CASP meeting in Germany.

7. Next Steps

Ad 3.1
- Participation in the forthcoming Containment Review Group Meeting of the USNRC in January 1979.
- Collaboration and participation in the forthcoming Common Review Group Meeting and CASP Meeting in March 1979.

AD 3.2 Contacts with EG&G (BEACON code) and LLL (PELE-IC code).

Ad 3.3 Collaboration in providing a verification matrix for the BEACON/MOD2A code and applications of the PELE-IC code.

8. Relation to Other Projects

RS 50, RS 50A, RS78D, RS 223, RS 246, RS 263/1, RS 263/6, RS 263/7, HDR-Project, GKSS Experimental Program.

9. References

/1/ G. Mansfeld: Abstellung zur USNRC zur Kooperation auf dem Containmentgebiet. GRS-A-255, Januar 1978.

10. Degree of Availability of the Reports

From the GRS-Forschungsbetreuung.

Classification 7-1

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.

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 coad 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 th[illegible]ation of the various components that influence the pressure history such as :

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

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

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

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

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

Project Status :

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

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

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

7. Reference documents :

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

Nuclear Engineering and Design 23 (1972)

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

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

8. Degree of availability : CONTACT TRACTIONEL - BRUSSELS

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	7.1.	RS 50

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

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
May 1971	June 30, 1979	Dr. T. F. Kanzleiter
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	DM 21.940.000.--

1. General Aim

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

2. Particular Objectives

Problems to be investigated experimentally:

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

3. Research Program

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

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

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

3.4. Additional LOCA experiments.

4. Experimental Facilities, Computer Codes

The experimental facilities consist essentially of

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

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

5. Progress to Date

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

Ad 3.2 A series of basis containment LOCA experiments (Nos. D1 to D15) with steam line breaks and simplified containment geometry has been performed in 1977. Evaluation has been completed in 1978.

Experiment No. D15 has been chosen as a Standard Problem for an international comparison with theoretical results.

6. Results

Ad 3.2 see results for Research Project RS 50 A

7. Next Steps

Ad 3.1 and 3.2 Preparation of additional reports

Ad 3.3 and 3.4 Specification and preparation of additional experiments.

8. Relation to Other Projects

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

9. References

(1-4) Quarterly Reports in the Series "GRS-Fortschrittsbericht. Bericht über die vom Bundesministerium für Forschung und Technologie geförderten Forschungsvorhaben auf dem Gebiet der Reaktorsicherheit" (in German)
- January to March 1978
- April to June 1978
- July to September 1978
- October to December 1978

(5) BF-RS 50-21-1
"Die Containment-Versuchsanlage (C- und D-Versuche)", October 1978

(6-8) BF-RS 50-24-6 through 8
"Abschlußbericht Fördervorhaben BMFT RS 50 SWR. Kennwort: Containmentversuche Battelle, Unterauftrag SWR-Versuche".
Vol. 1: "Theoretische Betreuung von SWR-Versuchen"
Vol. 2: "Wirksamkeit der Sicherheitsbehälter-Sprühkühlung"
Vol. 3: "Spezifikation der SWR-Versuche"
Kraftwerk Union Erlangen, September 1977

(9) BF-RS 50-30-D2
"Versuchsdokumentation D2", June 1978

(10) BF-RS 50-30-D4
"Versuchsdokumentation D4", June 1978

(11) BF-RS 50-30-D5
"Versuchsdokumentation D5", June 1978

(12) BF-RS 50-30-D10
"Quick Look Report Experiment D10", April 1978

(13) BF-RS 50-30-D11
"Quick Look Report Experiment D11", January 1978

(14) BF-RS 50-30-D12
"Versuchsdokumentation D12", June 1978

(15) BF-RS 50-30-D13
"Quick Look Report Experiment D13", April 1978

(16) BF-RS 50-30-D14
"Quick Look Report Experiment D14", April 1978

(17) BF-RS 50-30-D15
"Quick Look Report Experiment D15", March 1978

(18) BF-RS 50-30-D15-2
"Quick Look Report Experiment D15 (English Version)", November 1978

(19) BF-RS 50-31-6
"Kondensatflächen im Containment bei den Versuchen D1 bis D15", July 1978

(20) BF-RS 50-32-D10
"Ergänzende Versuchsdokumentation D10", April 1978

(21) BF-RS 50-32-D11
"Ergänzende Versuchsdokumentation D11", February 1978

(22) BF-RS 50-32-D13
"Ergänzende Versuchsdokumentation D13" June 1978

(23) BF-RS 50-32-D14
"Ergänzende Versuchsdokumentation D14",
June 1978

(24) BF-RS 50-32-D15
"Ergänzende Versuchsdokumentation D15",
April 1978

(25) BF-RS 50-42-6(-2)
"Beurteilung der Ergebnisse des Versuchs D10",
December 1977/March 1978

(26) BF-RS 50-62-6
"Ergebnisse und Auswertung von Blowdown-Versuchen in einem mehrfach unterteilten Modellcontainment (D-Versuchsprogramm)", December 1978

(27) BF-RS 50 A-1, Vol. 1 and 2
"Abschlußbericht RS 50 A. Begleitende Arbeiten zu den D-Versuchen des Forschungsvorhabens RS 50 (Modellcontainment)", September 1978

(28) GRS-A-64 (Auftragsbericht)
G. Baier, "Voraus- und Nachrechnungen des im Battelle-Institut, Frankfurt/Main durchgeführten Containment-Versuches D6 des Forschungsvorhabens RS 50, D-Reihe", November 1977

(29) GRS-A-71
A. Berning, G. Mansfeld, "Nachrechnungen zu den Hauptversuchen C6, C9, C13 und C15 des Forschungsvorhabens RS 50 Druckverteilung im Containment (Battelle-Modell-Containment)", December 1977

(30) GRS-A-72
G. Hellings, "Nachrechnungen zu den beim Battelle-Institut, Frankfurt/Main durchgeführten Versuchen D1 und D3 des RS 50 Forschungsvorhabens Druckverteilung im Containment", December 1977

(31) GRS-A-117
G. Hellings, "Nachrechnungen zu den beim Battelle-Institut, Frankfurt/Main durchgeführten Versuchen D6, D7 und D8 des RS 50 Forschungsvorhabens Druckverteilung im Containment", March 1978

(32) GRS-A-213 (Auftragsbericht)
M. Tiltmann, "Langzeitvergleichsrechnungen mit EDV-Programm CONDRU zu ausgewählten Versuchen der D-Serie des Battelle-Modell-Containments", September 1978

(33) GRS-A-234 (Auftragsbericht)
A. Berning, W. Winkler, "Deutsches Standard-Problem Nr. 1 (Containment-Standard-Problem) Experimentelle Daten des Versuchs D15 (Battelle-Modell-Containment)", November 1978

(34) GRS-A-250 (Auftragsbericht)
A. Berning, W. Winkler, "OECD-CSNI Containment Standard Problem No. 1. Experimental Data of Test D15 (Battelle-Model-Containment)", December 1978

(35) GKSS 78/I/24
H. Schwan, "Thermodynamische Transienten und ihre analytische Simulation am Beispiel des Forschungsvorhabens RS 50", Geesthacht 1978

(36) G. Mansfeld, "Zum instationären Druckaufbau in Volldrucksicherheitsbehältern wassergekühlter Kernreaktoren nach einem Kühlmittelverlustunfall" Thesis, Technical University Munich, July 1977

10. Degree of Availability of the Reports

Reports are available through GRS-FB.
Documents (5) to (25) and (27) to (35) can be made available only by special agreement.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	7.1	RS 050 A
Vorhaben/Project Title Begleitende theoretische Arbeiten zu den D-Versuchen des Forschungsvorhabens RS 50 Analysis of the D-Series Experiments of Research Project RS 50		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor BATTELLE-INSTITUT E.V. Frankfurt am Main
Arbeitsbeginn/Initiated May 2, 1977	Arbeitsende/Completed September 30, 1978	Leiter des Vorhabens/Project Leader Dr. T. F. Kanzleiter
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds DM 399.850,--

1. General Aim

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

2. Particular Objectives

- To substantiate the initial evaluation of experimental results (Quick Look Report).
- To assist in the detailed planning of following experiments.

3. Research Program

3.1. Calculation of the containment pressurization immediately after the performance of an experiment using the experimental mass flow data. Documentation of the calculational results in comparison with experimental data in Quick Look Reports.

3.2. Analysis of experimental results by

- parametric studies
- variation of program options
- calculation with separate calculating models (e.g. overflow model) using experimental data as boundary conditions.

4. Computer Codes

ZOCO 6 with modifications,
RELAP4 Mod 5

5. Progress to Date

Ad 3.1 and 3.2: Post-calculations of ten experiments have been completed and described in Quick Look Reports and in the Final Report /7/.

6. Results

Ad 3.1 and 3.2: The most significant results of the evaluation of the RS 50 D-series containment LOCA experiments (steam line breaks, simplified containment geometry) are:

- Using conventional parameters for calculation, the ZOCO 6 code yields an overestimation of the containment short term pressurization by 20 to 30 percent in case of steam line breaks. (Earlier RS 50 experiments showed that in case of water line breaks the overestimation can exceed a factor of 2).

- A good agreement between the calculated and the measured pressure built-up in the short-term period can be achieved by assuming extremely high heat transfer coefficients up to α_W = 15,000 W/m^{2}K in the break compartment and lower values for the following compartments.

- A comparison of similar experiments with different vent geometry showed good agreement with model calculations using the isentropic flow model and the following contraction coefficients:
  - nozzle-type vents (dia 600 mm): $\alpha_D = 1.0$
  - orifice-type vents (dia 750 mm): $\alpha_D = 0.7$
  - channel (dia 740 mm, length 2000 mm) : $\alpha_D = 0.9$

  (subscritical flow, pressure ratio $\leq$ 1.5)

- Comparable results for vent flow are obtained with the flow data of Schiller (original ZOCO 6 version) combined with a discharge coefficient of $\alpha_D = 0.65$ for orifice-type vents and with the flow data of Frössel /1/ for the channel.

- For vents located in a velocity field the upstream fluid velocity must be accounted for either directly in the code (eg. COFLOW or DDIFF2) or, in an auxiliary manner, by increasing the contraction coefficient.

7. Next Steps

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8. Relation to Other Projects

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

9. References

/1/ Frössel, "Strömung in glatten, geraden Rohren mit Über- und Unterschallgeschwindigkeit". Forschung 7 (2), March/April 1936

/2/ BF-RS 50-30-D10 "Quick Look Report Experiment D10", April 1978

/3/ BF-RS 50-30-D11 "Quick Look Report Experiment D11", January 1978

/4/ BF-RS 50-30-D13 "Quick Look Report Experiment D13", April 1978

/5/ BF-RS 50-30-D14 "Quick Look Report Experiment D14", April 1978

/6/ BF-RS 50-32-D15 "Ergänzende Versuchsdokumentation D15", April 1978

/7/ BF-RS 50A-1, Vol. 1 and 2. "Abschlußbericht RS 50 A. Begleitende theoretische Arbeiten zu den D-Versuchen des Forschungsvorhabens RS 50 (Modellcontainment)", September 1978

/8/ BF-RS 50-62-6 "Ergebnisse und Auswertung von Blowdown-Versuchen in einem mehrfach unterteilten Modellcontainment (D-Versuchsprogramm)", December 1978

10. Degree of Availability of the Reports

Reports are available through GRS-FB.
Documents /2/ to /7/ can be made available only by special agreement.

Berichtszeitraum/Period 1.1. - 31.12.1978	Klassifikation/Classification 7.1	Kennzeichen/Project Number RS 263/1
Vorhaben/Project Title Analytische Tätigkeiten der GRS im Rahmen des Reaktorsicherheitsforschungsprogramms des BMFT **Druckverteilung im Containment** Analytical Activities of the GRS in the Frame of the BMFT Research Program on Reactor Safety Pressure Distribution in the Containment		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH
Arbeitsbeginn/Initiated 1.1.1977	Arbeitsende/Completed 31.12.1979	Leiter des Vorhabens/Project Leader DI G. Hellings
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 31.12.1978	Bewilligte Mittel/Funds

1. General Aim

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

2. Particular Objectives

2.1 Check of the computer codes to get information on their range of application and capability.

2.2 Improvement and further development of the computer codes.

2.3 Examinations to determine parameters having a large influence.

3. Research Program

3.1 Work on the basic versions of the computer codes COFLOW and CONDRU (documentation of the codes, verification of the codes by recalculation of experiments).

3.2 Further development of the computer codes and quantification of conservatism.

3.3 Verification of further developments of the codes.

3.4 Preparatory work for further RS 50 experiments.

3.5 Work with external computer codes.

4. Experimental Facilities, Computer Codes

Recalculations of RS 50 experiments, series C and D, carried out in the Battelle model containment, with the containment codes COFLOW (calculations during the first seconds of the LOCA) and CONDRU (calculations over long periods).

5. Progress to Date

Ad 3.1 Work on documentation and verification of the basic versions of the codes:

- The description of CONDRU /1/ and COFLOW /2/ were finished.
- Most of RS 50 D-Series experiments were recalculated (D1, D3 /3/, D6, D7, D8 /4/ and D11 with COFLOW; D8, D11 /5/ with CONDRU).

Ad 3.2 Participation in standard problems to quantify model specific conservatism:

- The influence of the different initial and boundary conditions of the experiments D10 and D15 was analysed to see wether D15 can replace the experiment D10 which failed or whether new calculations must be made /6/.
- After it was decided to repeat the standard problem, the experiment D15 was calculated as a new standard problem with COFLOW /7/ and CONDRU /8/.
- For the interpretation of the results of the new standard problem D15, experimental results were prepared and plotted in a specified way /9/.

Ad 3.4 Activities concerning further experiments:

- The results of the RS 50 experiments performed up to now were examined.
- Information is being obtained from other research programs (HDR, ECOTRA, GKSS test facility PSS).

6. Results

Ad 2.1 Recalculation of RS 50 experiments show the capability of the existing codes:

- With standard parameters (which are also used in licensing calculations) the experimental pressure was reproduced in a conservative way
- A quantitative statement on the overestimation of the pressure cannot be made yet
- Good correspondance between experimental and theoretical results can be achieved by parameter variations.

Ad 2.3 Information on important parameters can be obtained by the comparison of experimental and theoretical results of RS 50 experiments:

- Parameters of the containment codes, which mainly influence the analytical results, are the water carry-over factor, the heat transfer coefficient, and the contraction coefficient.
- The values for the parameters gained in the RS 50 experiments by parameter variations (low water carry-over factors, high heat transfer coefficients near the point of rupture) need to be confirmed by further experiments
- Better interpretation of the RS 50 experiments can be obtained if the discrepancy between measured heat transfer coefficients (≈800 W/m^2·K) and those found by recalculations (up to 10^4 W/m^2·K) can be resolved.
- A check of the measurement techniques and their error bands is necessary for further reflection on the accuracy of theoretical predictions.

7. Next Steps

Ad 3.1 Recalculation of the remaining RS 50 experiments.

Ad 3.2 Analysis and interpretation of the results of all RS 50 experiments performed until now; collection of information from other research programs to confirm and complete the results obtained with RS 50 experiments.

8. Relation to other projects

HDR, ECOTRA, GKSS test facility (PSS).

9. References

/1/ M. Tiltmann, B. Hüttermann: Beschreibung des Rechenprogramms CONDRU-4. GRS-A-124, März 1978

/2/ G. Hellings, G. Mansfeld: COFLOW - Ein Rechenmodell zur Ermittlung des instationären Druckaufbaus in Volldrucksicherheitsbehältern wassergekühlter Kernreaktoren, Programmbeschreibung. GRS-A-254, Dezember 1978

/3/ G. Hellings: Nachrechnungen zu den beim Battelle-Institut, Frankfurt/M, durchgeführten Versuchen D1 und D3 des RS 50 Forschungsvorhabens "Druckverteilung im Containment". GRS-A-72, Dezember 1977

/4/ G. Hellings: Nachrechnungen zu den beim Battelle-Institut, Frankfurt/M, durchgeführten Versuchen D6, D7 und D8 des RS 50 Forschungsvorhabens "Druckverteilung im Containment". GRS-A-117, März 1978

/5/ M. Tiltmann: Langzeitvergleichsrechnungen mit EDV-Programm CONDRU zu ausgewählten Versuchen der D-Serie des Battelle-Modell-Containments. GRS-A-213, September 1978

/6/ G. Mansfeld, H. Winkler: Untersuchungen zu den Rand- und Anfangsbedingungen des Standard-Problem-Versuchs D10 und des Wiederholungsversuchs D15 (Battelle-Modell-Containment). GRS-A-111, Februar 1978

/7/ G. Mansfeld, A. Berning: Containment-Standard-Problem: Vorausgerechnete Ergebnisse des Wiederholungsversuchs vom 20.12.1977 mit dem Rechenprogramm COFLOW und Kurzbericht. GRS-A-197, September 1978

/8/ M. Tiltmann: Containment Standard Problem - Vorausgerechnete Ergebnisse mit dem Rechenprogramm CONDRU zu dem RS 50-CASP-Versuch. GRS-A-196, September 1978

/9/ A. Berning, W. Winkler: Deutsches Standardproblem Nr. 1 (Containment-Standard-Problem), Experimentelle Daten des Versuchs D15 (Battelle-Modell-Containment). GRS-A-234, Dezember 1978.

10. Degree of Availability of the Reports

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

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	7.1	RS 93 A
Vorhaben/Project Title Weiterführung der Untersuchungen über die Auswirkungen des Ausströmens von Dampf-Wasser-Gemischen aus Rohrleitungslecks Investigations on critical two-phase flow with regard to transient, scaling and subcooling effects		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 513, Erlangen
Arbeitsbeginn/Initiated 1. 2. 76	Arbeitsende/Completed 31. 3. 79	Leiter des Vorhabens/Project Leader W. Kastner
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 78	Bewilligte Mittel/Funds 456.400,-- DM

1. General Aim

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

2. Particular Objectives

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

3. Research Program

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

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

3.3 To perform experimental investigations under steady-state conditions to the influence of subcooling on critical two-phase flow in a new program, the test facility will be modified now.

4. Experimental Facilities

The experiment should be performed with a test facility consisting of a 8 m^3 pressure vessel (100 bar, 310 deg C) and in contrast to RS 93 of a straight discharge pipe

(150 mm diameter, 2750 mm length) with a quick opening and closing valve. The nozzle has a discharge diameter of 40 mm.

5./6. Progress to Date/Results

To 3.3 The planning and design work for the modification of the test facility is completed. The components with a long lead time were ordered. Modification work has started, for example the arrangement of the baffle plate in the 48 m^3 container was changed, such that a measurement of the jet force could be performed. Design work of the nozzle with 40 mm diameter was performed. A measurement location plan for a total of 32 measurement pick-ups was prepared. There are 17 measuring positions at the pressure vessel and the discharge line for the measurement of pressure, temperature and water level in the vessel, 12 measuring points on the baffle plate for determination of the pressure distribution and 3 jet force transducer. This measurement location plan was discussed with GRS experts. Procurement of the instrumentation has started.

The subsequent experimental program as well as the accompaning calculations were proposed. The test matrix was planned and discussed with GRS experts, too.

7. Next Steps

To 3.3 Performing of final modification work at the test facility. Installation of the nozzle and the instrumentations at the pressure vessel, discharge line, nozzle and baffle plate. Connection to the data aquisition system. Shake down tests.

8. Relation with Other Projects

Previous efforts were conducted in the framework of a proposal by RS 93.

9. Reference

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10. Degree of Availability ---

Classification: 7.1 7.2

Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: MACON. A Containment Multiroom Transient Analysis Code.	Organization: Risø National Laboratory
Initiated date: 1975 Completed date: Status: Discontinued 1978.	Scientists: V.S. Pejtersen, M.B. Andersen

1. General aim

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

2. Particular objectives

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

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

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

3. Experimental facilities and programme

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

4. Project status

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

5. Next system

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

6. Relation with other projects

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

7. Reference documents

No reports available yet.

8. Degree of availability

Classification: 7.1, 7.2

Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: CONTAC-II. A containment transient analysis code	Organization: Risø National Laboratory
Initiated date: January 1969 Completed date: Feb. 70 Status: In use	Scientists: Aksel Olsen 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.

5. Experimental facilities and programme

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

4. Project status

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

5. Next steps

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

6. Relation with other projects

7. Reference documents

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

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

8. Degree of availability

Available.

Classification: 7.1 7.2

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

1. General aim

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

2. Particular objectives

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

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

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

3. Experimental facilities and programme

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

4. Project status

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

5. Next steps

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

6. Relation with other projects

7. Reference documents

N. Kjær-Pedersen:

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

N. Kjær-Pedersen:

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

8. Degree of availability

145-1-08		7-1
TITRE Conséquences d'un LOCA sur l'enceinte de confinement. Etude de la condensation sur un mur d'un mélange air-vapeur dans des conditions transitoires. Programme ECOTRA.		Pays FRANCE Organisme Directeur CEA/DgCS - EDF/SEPTEN
TITLE (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
Date de démarrage 01/01/76	Etat actuel en cours	Scientifiques
Date d'achèvement 31/12/80	Dernière mise à jour 01/79	

1. OBJECTIF GENERAL

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

2. OBJECTIFS PARTICULIERS

Développement de modèles physiques pour l'interprétation des expériences.

3. INSTALLATIONS EXPERIMENTALES 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 permettra de déterminer le coefficient d'échange.

Les principaux paramètres d'expérience sont :

- la nature de la section d'essais
- l'orientation de la section d'essais
- la pression de vapeur
- la vitesse du mélange "vapeur + air"
- le titre en air.

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4 - ETAT DE L'ETUDE

1) Avancement à ce jour

Installation opérationnelle depuis Avril 1977.

Expériences en cours sur une section d'essai en acier inoxydable.

2) Résultats essentiels

Influence de la vitesse, de la pression et de la teneur en air sur les coefficients d'échanges de condensation. Etude de l'influence de l'orientation de la plaque.

Etablissement de corrélation.

5 - PROCHAINES ETAPES

Essais sur une section d'essai en béton.

Fin des essais sur plaque en acier inoxydable avec :

- influence de la température de conditionnement initial de la section.
- essai de dépressurisation
- influence peinture
- influence rugosité

Essai sur une plaque en silice simulant du béton : série d'essais semblables à ceux de la section inox.

145-1-09		7 - 1
TITRE CONSEQUENCES D'UN LOCA SUR L'ENCEINTE DE CONFINEMENT. ETUDE DE L'ECOULEMENT DU BROUILLARD ENTRE LES CASEMATES D'UNE ENCEINTE DE CONFINEMENT : PROGRAMME REBECA.		**Pays** FRANCE **Organisme Directeur** CEA/DgCS
TITLE (Anglais) Mist flow between subcompartments of a containment : REBECA project.		**Organisme exécuteur** C.E.A. DRE/STRE **Responsable**
Date de démarrage 01/07/76	**Etat actuel** En Cours	**Scientifiques**
Date d'achèvement 31/12/80	**Dernière mise à jour** 01/1979	

1. OBJECTIF GENERAL

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

2. OBJECTIFS PARTICULIERS

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

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

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

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

4. ETAT D'AVANCEMENT

La boucle est en construction; la fin de la réalisation est prévue pour le mois d'avril 1979.
Les essais du mélangeur "classique" à pulvérisation d'eau sous pression se poursuivent.
Les essais réalisés sur l'installation TUYERE ont montré la validité des mesures de pression en simple phase (eau ou air).
La première section d'essai est en cours de fabrication.

5. PROCHAINES ETAPES

Début des essais en juin 1979.

6. RELATIONS AVEC D'AUTRES ETUDES

- MARVIKEN CFT (étude de débits critiques) - n° 145-1-10.
- BATTELLE (Francfort) : suivi de programmes étrangers, dans le cadre de l'étude n° 145-1-02.
- MOBY-DICK et SUPER MOBY-DICK - n° 145-1-03.

7. DOCUMENTS DE REFERENCE

7.1 - "REBECA - Two-phase two-component flow"
D. MENESSIER, J.P. BRUNET, A. MATTEI
Rapport DRE/STRE 76/002

7.2 - "Analyse des méthodes de calcul de pression dans les casemates d'une enceinte de confinement"
D. MENESSIER, A. MATTEI
Note technique DRE/STRE/LET 76/027

	CLASSIFICATION: 7.1
TITLE (ORIGINAL LANGUAGE): Forze di getto da fluídi bifase e carichi di "impingement"	COUNTRY: Italy
	SPONSOR: AMN-CNEN
TITLE (ENGLISH LANGUAGE): Research on jet forces and impingement loads	ORGANISATION: AMN
	PROJECT LEADER: R. Centi

INITIATED: 1976	COMPLETED:	SCIENTISTS: E. Lumini, F. Lo Nigro, G. Gaspari
STATUS: In progress	LAST UPDATING: July 1979	

1) General Aim

Containment structural components and active components sizing (safety-related) must allow for jet forces and consequent impingement loads, arising from pipe-break accident of high energy lines. The aim of this work is to study such forces, to compute their values as a function of rupture types, fluid thermodynamic conditions, target displacement, target geometry.

2) Particular Objectives

The work is going on following three principal lines:
a) Analysis of jet characteristics evaluation methods (opening angle, radial velocity profile, radial density profile).
b) Drafting of computational methods in design guide to implement the actual state of art.
c) Analysis of components hit by the jet.
Item "a" is being developed in collaboration with CNEN, CSN Casaccia research centre.

3) Project Status

Experimental facilities are built to confirm or improve the theoretical data. A computational code has been implemented to evaluate jet forces and impingement loads on the basis of theoretical studies.
A new correlation has been set up for pressure axial profile, for a saturated water jet.

4) Relation to Other Projects or Codes

7.1. JEFOC 3D

45.897

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Research on jet forces and impingement loads	7.1

5) Reference Documents

- R. Centi, "Forze di getto nel BWR", SPR/SNU 7705, AMN S.p.A.
- R. Centi, "Valutazione delle caratteristiche del getto", NT-SNU 7708, AMN S.p.A.
- S. Ferrini, G. Garofalo, "Considerazioni sugli efflussi critici per la determinazione delle forze di getto", 78(S) VBD-2; G71A, CNEN.
- R. Centi, "Design guide against jet-forces and impingement loads", AMN.
- R. Centi, "Specifiche di esperienze per lo studio delle forze di getto", IP78-05, AMN.
- R. Centi, "Codice di calcolo JEFOC", NTRNU 78-07 Rev. 0, AMN.
- R. Centi, E. Lumini, "Codice di calcolo JEFOC 3D".
- M. Faravelli, "Impianto Fog: prestazioni e caratteristiche", IC 78-04 Rev. 0, AMN.
- R. Centi, "Forze di getto a seguito di rotture di linee percorse da acqua satura (specifica di prova)", IV 79-04, AMN.
- M. Cumo et al., "Ricerche su efflussi critici",79(3)VDB-1/G 71 A-2, CNEN.

6) Additional Information

Studies are performed in the frame of CNEN-AMN Research Agreement.

(R. Centi, AMN, V. D'Annunzio 113, I-16121 Genova).

ENERGIEONDERZOEK CENTRUM NEDERLAND		CLASSIFICATION: 7,1
TITLE: CHARME-IM Berekening van de impulsbelasting door uitstroming bij pijpbreuken		COUNTRY: THE NETHERLANDS
		SPONSOR: ECN ORGANIZATION: ECN
TITLE (ENGLISH LANGUAGE): CHARME-IM Blowdown jet impingement process		PROJECTLEADER: Speelman, J.E.
INITIATED : 1976	LAST UPDATING : May 1978	SCIENTISTS: Putten, A.P.W.M. van der Bogaard, J.P.A. van den Koning, H.
STATUS : progressing	COMPLETED :1980	

General aim

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

Particular objectives

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

Experimental Facilities and program: none

Project status

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

Next steps: Introduction of special geometries

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

Reference documents: -

Degree of availability: Upon mutual agreement at ECN-Petten

Budget:

Personnel:

ENERGIEONDERZOEK CENTRUM NEDERLAND		CLASSIFICATION : 7.1
TITLE : CHARME-DIS Berekening van het thermohydraulische proces in de uitstroomleiding van veiligheidskleppen		COUNTRY: THE 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 : May 1978	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 subroutine to calculate shockwaves has been developed.

Next steps

Calculators on special applicators will be performed.

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

Reference documents: -

Degree of availability: Not yet applicable

Budget: -

Personnel: -

TNO-IBBC	CLASSIFICATION : 3.2/3.3/
TITLE : Responsieberekeningen voor reactorgebouw	COUNTRY: THE NETHERLANDS
	SPONSOR : Ministry of Soci. Affairs; Ministry of Publi works (DIV); TNO-IBBC ORGANIZATION :
TITLE (ENGLISH LANGUAGE): Dynamic response of reactor structures (building and containment)	PROJECTLEADER : Kusters
INITIATED : June 1974 — LAST UPDATING : June 1979	SCIENTISTS : Kusters, de Groot, de Witte, Tolman,
STATUS : Progressing — COMPLETED : 1979	

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

	CLASSIFICATION: 7.1
TITLE (ORIGINAL LANGUAGE): INTERNAL MISSILE EFFECTS	COUNTRY: UK
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD CULCHETH
	PROJECT LEADER: D L HUNT
INITIATED: 1974 / COMPLETED: 1979	SCIENTISTS:
STATUS: / LAST UPDATING: May 1979	

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 STATE

The 'steam rocket' experiment has been built to test SRD calculational methods. The flight of the pipe is measured over [illegible]5 ft mainly by fast photography (2,000 frames/sec) against a grid, but acceleration over the first 6" is by a photo-metric 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.

5. REFERENCE DOCUMENTS

Report to be published.

	CLASSIFICATION: 7.1
TITLE (ORIGINAL LANGUAGE): PWR PLANT AND CORE STRUCTURAL DYNAMIC RESPONSE (GSD 3.2)	COUNTRY: UK
	SPONSOR: UKAEA
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/UKAEA
	PROJECT LEADER: RNH McMILLAN
INITIATED: DECEMBER 1978 / COMPLETED:	SCIENTISTS/ENGINEER: ML BROWN
STATUS: / LAST UPDATING: MAY 1979	

BACKGROUND

The ability to predict the dynamic structural response of PWR systems and components following a LOCA is an essential requirement for the evaluation of accident behaviour, since such responses may influence the reliability and effectiveness of post-accident engineered safeguards. Assurance is particularly required for the reactor shutdown capability, maintenance of a coolable core geometry, effectiveness of ECCS and containment integrity. By way of illustration, Section 3 of the USNRC Regulatory Guide lists the information to be provided in both the Preliminary and Final Safety Analysis Reports. This includes justification for protection of the Primary Circuit and all essential components/systems against the effects of blowdown jet and reaction forces and pipe whip and missile impact.

Other accident generating or aggravating events need to be considered, including propagation of SG tube failures, small pipe fractures, relief valve operation, missile generation, etc.

Evaluation of these complex phenomena required analysis of fluid dynamics to predict forcing functions coupled with predictions of the resultant structural response.

OBJECTIVES

1. To review potential accidents and identify areas where dynamic structural analysis is needed to follow their course.
2. Develop the code SARCASTIC for analysis of missile and impulsive loads on PWR components and structures. (Main development funded by CRSR).
3. To review available analytical methods, chart their limitations, and recommend their applications in PWR safety evaluation. Experimental validation or the need for it is an important part of this review. Major work areas are:
   - 3.1 Local structural response: a code review covering the local response of structures, both metal and concrete eg missile impact, or impulsive jet loading on pipes or walls.
   - 3.2 System structural response: a code review of simplified, 3-D, mass-spring models for assessing the response of complete systems to dynamic loading eg piping system or reactor internals subject to LOCA loads.
   - 3.3 Overall structural dynamic response: A code review covering the modelling of complex structures directly eg deforming pressure circuits or containment structural response.
   - 3.4 Fluid dynamics: A review of 2 phase flow codes for the estimation of transient forces in structures eg containment vault pressure or reactor internal loadings in a LOCA. Also the identification of any need for coupled fluid/structure analysis methods.

	CLASSIFICATION:
TITLE (ORIGINAL LANGUAGE):	COUNTRY:
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION:
	PROJECT LEADER:
INITIATED: / COMPLETED:	SCIENTISTS:
STATUS: / LAST UPDATING:	

FACILITIES

(a) Analytical Codes The following is a (non-exhaustive) list of Codes for Review:

Pipe whip:
Water Hammer: WHAM
Internal Missiles: SARCASTIC CADROS SAP IV/NONSAP EURDYN
RPV Vault pressure:

(b) Experimental

AEEW experimental programme will form an important part of the development and validation of SARCASTIC.

INTERNATIONAL COLLABORATION

An information exchange agreement with CEA/EDF has recently been concluded which will, inter alia, cover both hard and soft missile impacts upon concrete.

Collaborative work on missile impact with GRS (EMFT) in the FRG is in progress, involving experimental and theoretical work. The German experimental work is at a lerger scale.

REFERENCE DOCUMENTATION

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	7.2	RS 263/ 7
Vorhaben/Project Title Analytische Tätigkeiten der GRS im Rahmen des Reaktorsicherheitsforschungsprogramms des BMFT Dynamisches Verhalten von Fluid und Struktur in Druckabbausystemen Analytical Activities of the GRS in the Frame of the BMFT Research Program on Reactor Safety Dynamic Behaviour of Fluid and Structure in Pressure Suppression Systems		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Gesellschaft für Reaktorsicherheit (GRS) mbH
Arbeitsbeginn/Initiated 1.7.1977	Arbeitsende/Completed 31.12.1979	Leiter des Vorhabens/Project Leader Dr. W.Ch. Müller
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 31.12.1978	Bewilligte Mittel/Funds

1. General Aim

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

2. Particular Objectives

Calculation of the non-stationary behaviour of pressure suppression systems (PSS) with regard to real PSS structures. Simulation of condensation events and their impact on the PSS. Evaluation of the interaction between condensation events, fluid response, and structure response. Analysis of experiments performed to date (GKM, GKSS). Inquiry on the influence of special forms of the systems with regard to shape and size of the resulting loads on the PSS. Planned strategy:

a) Uncoupled treatment in three steps,
   - simulation of condensation,
   - simulation of the physica of the fluid,
   - structural analysis of the PSS.

b) Coupled treatment of fluid dynamics and structure dynamics, coupling of the programs developed and modified for the analysis of the separated systems.

c) Coupling of condensation events and fluid response (impact of the fluid on condensation) on the basis of the available programs.

3. Research Program

3.1 Testing the applicability of the codes DRASYS, DAPSY and

SAP to simulate condensation events and the response of the PSS coupled fluid structure system.

3.2 Analysis of chugging phenomena by evaluating measured date and videofilms. Specification of load functions for a separate treatment of condensation events and the response of the fluid structure system.

3.3 Analysis of the coupled behaviour of the fluid structure system by modifying computer codes or by developing suitable codes based on available codes like ROTSYM, SAP-IV, DAISY.

4. Test Facilities, Computer Codes

Computer codes: DRASYS, DAPSY, SAP, ROTSYM, DAISY, LINFLU.
New codes and modified versions:
ROTSYM-C (New release of ROTSYM): FEM-Code for the static, modal and dynamic analysis of axisymmetric shells in contact with a compressible or incompressible linear fluid. The code is supplied with interfaces necessary for coupling with a suitable FEM-Code for the analysis of a linear fluid.
LINFLU: FEM-Code for the analysis of the stationary pressure distribution in an axisymmetric linear fluid, modal analysis and calculation of the matrices and vectors necessary for a dynamic analysis or for the coupling with a structural analysis code.

5. Progress to Date

Ad 3.1 Testing the available codes led to the decision to modify the code ROTSYM and to couple it with the newly developed code LINFLU, which was especially designed for this purpose.

Ad 3.2 Great efforts were made to obtain detailed data and films on condensation events in test facilities of reasonable size. Since these data of the GKSS tests and GKM tests were only available as raw data, considerable work was spent in selecting tests and preparing the data. Part of the GKSS data is available on IBM compatible computer tapes ready for evaluation. Video and high speed films have been transformed for the use on GRS equipment and a frame by frame evaluation of two filmed chugging events have been carried out. A preliminary specification of load function was done on the basis of the data available.

Ad 3.3 A new FEM-Code LINFLU for the static, modal and dynamic analysis of a linear fluid has been programmed and tested. The special features of the code are a small core storage and ultra-short running times. By the use of a compacted Cholesky method reasonably small errors are obtained even when single precision is used. This results in a vast saving of computer storage:

The computer code ROTSYM has been modified, so that it can be coupled with any FEM-Code for the analysis of a linear fluid, e.g. LINFLU.

The methods and techniques of the coupling itself are taken from standard control theory. The new code has been tested with a small size sample problem. These tests have been performed successfully and showed differences between the unwetted and wetted structure, and the use of a compressible or incompressible fluid.

6. Results

Ad 3.1 The work planned under 3.1 has been successfully carried out, leading to the decision to develop a new linear fluid FEM Code and to couple it with a modified version of ROTSYM.

Ad 3.2 A large bulk of data on condensation events and films have been prepared for further use and have been analysed in the search for better understanding and prediction of chugging events. For the first time detailed information on single chugging events is available. The analysis has been carried out, using a time step of about 1/500 of a second. It shows that condensation events do not consist of the collapse of a huge spherical bubble, but of small size events at the outlet of the condensation pipe, due to highly turbulent motion.

All the work done on the data was documented in a report that provides all information necessary for further analysis in this field. A preliminary report on load functions provides an overview of the work done up to date and gives load functions based on theoretical and experimental values. These load functions seem to be a good approximation. Nevertheless, further work is considered necessary because only a small part of the experimental data has been analysed for this report.

Ad 3.3 After the successful development of LINFLU and the modification of ROTSYM, a first version of a coupled fluid structure code is available which has been checked by a small size test sample. Calculations show that the use of a bare structure or a structure wetted by a compressible or incompressible fluid ends up with different solutions. Results do not indicate which way will lead to the highest impact on the structure, but rather show a clear tendency to lower the natural frequency of the wetted structure.

7. Next steps

Further work will concentrate on two points:

- Analysis of measured data in order to find limits on the extent of condensation events by the use of statistical methods. In this way means will be provided to reduce the uncertainty in determining load functions for the coupled fluid structure system.
- Analysis of the response of the coupled fluid structure system in order to trace back the events that caused the measured pressure pulses.

8. Relations to other Projects

RS 263/6, GKSS test program.

9. References

/1/ Auswertung von Kondensationsversuchen I, GKSS-Versuch Nr. 16 bis 20 und 33. PSP-Bericht Nr. 7825/M

10. Degree of Availability of the Reports

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

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	7.2	RS 263/6
Vorhaben/Project Title Analytische Tätigkeiten der GRS im Rahmen des Reaktorsicherheitsforshungsprogramms des BMFT Druckabbausysteme und Kondensation Analytical Activities of the GRS in the Frame of the BMFT Research Program on Reactor Safety Pressure Suppression Systems and Condensation		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH
Arbeitsbeginn/Initiated 1.1.1977	Arbeitsende/Completed 31.12.1979	Leiter des Vorhabens/Project Leader DI P. Schally
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 31.12.1978	Bewilligte Mittel/Funds

1. General Aim

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

2. Particular Objectives

2.1 Further development and assessment of DRASYS.

2.2 Acquisition and comparison of available computer codes for pressure suppression systems and condensation. Evaluation of GKM and GKSS tests.

2.3 Modification of the computer model DADY.

3. Research Program

3.1 Development and assessment of DRASYS.
Nonstationary behavior of temperatures in containment structures.
Simulation of chugging events with the code SCHUG.
Assessment of the SCHUG code with GKM-II and GKSS tests.

3.2 Acquisition, implementation and debugging of the code KONDAS.
Calculation of test examples.
Evaluation of condensation experiments at GKM and GKSS.

3.3 Modification of the computer model DADY to simulate the phenomena of bubble rise and bubble breakthrough.

4. Test Facilities, Computer Codes

Test Facilities

- Marviken experimental program serie I and II (RS 33).
- GKSS - one pipe test program.
- GKM-II test program.

Computer Codes

- DRASYS - Simulation of the integral behavior of a pressure suppression system.
- SCHUG - Simulation of chugging events.
- KONDAS - Pressure pulse development during steam condensation.
- DADY/DYWA - Simulation of pool swell.
- DESDUE - Simulation of the pressure-relief system.

5. Progress to Date

Ad 3.1 Further development of the code DRASYS to simulate the motion of water in the wetwell water pool and the nonstationary distribution of temperatures in containment structures.
Assessment of DRASYS with the Marviken and GKSS Tests.
Development of the computer model SCHUG for the simulation of chugging events.
Recalculation of selected GKM- and GKSS tests.

Ad 3.2 Recalculations of GKM and Marviken tests with the computer code KONDAS. Evaluation of GKSS test results (coordinated with RS 263/7).

Ad 3.3 Modification of the code DADY (now named DYWA) for the simulation of pool swell. Evaluation and selection of tests for multi-dimensional pool swell. Development of a two-dimensional bubble rise model.
Review of one-dimensional bubble rise analyses.

6. Results

Ad 3.1 The equations of motion used in the computer code DRASYS simulate well the motion in the wetwell water pool.
The modul HECU which is now included in DRASYS simulates the nonstationary distribution of temperatures in containment structures.
A short description of DRASYS is completed /1/.
The recalculations of Marviken and GKSS tests show good

agreement between theoretical and experimental results (figure 1).

The model development and programming of SCHUG is completed. First recalculations of tests show good agreements (table 1).

Ad 3.2 The recalculations of GKM- and Marviken tests with the code KONDAS and the documentation of the results is completed.

Ad 3.3 The development of the program DYWA was completed /2/. Most of the applicable tests can not be used for an assessment of DYWA or other codes, because of an insufficient measurement of the water surface behavior. Suggestions for future tests in the GKSS-facility have been proposed.

The technique to solve the system of differential equations (LAX-technique) leads to large computer times and will not be used in the future.

The studies of the one-dimensional bubble rise process showed an influence of the relative accelleration on the bubble rise velocity.

7. Next Steps

Ad 3.1 Further development of DRASYS.

Recalculations of GKM and GKSS Tests with SCHUG. Implementation of SCHUG in DRASYS.

Description of the program DRASYS and documentation of the short term period recalculations.

Investigation of harmonic condensation oscillation synchronisation with DRASYS.

Ad 3.2 Evaluation of GKM- and GKSS-tests.

Ad 3.3 DYWA test calculations and recalculations of selected tests. Implementation of the pool swell model in DRASYS.

8. Relations to other Projects

RS 263/7, GKSS test program.

9. References

/1/ P. Schally: Kurzdokumentation und Entwicklungsstand der Basisversion des Rechenprogrammes DRASYS. GRS-A-244, Dezember 1978

/2/ B. Schwinges: Untersuchungen zum Wasseraufwurf.
GRS-A-161, Mai 1978

10. Degree of Availability of the Reports

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

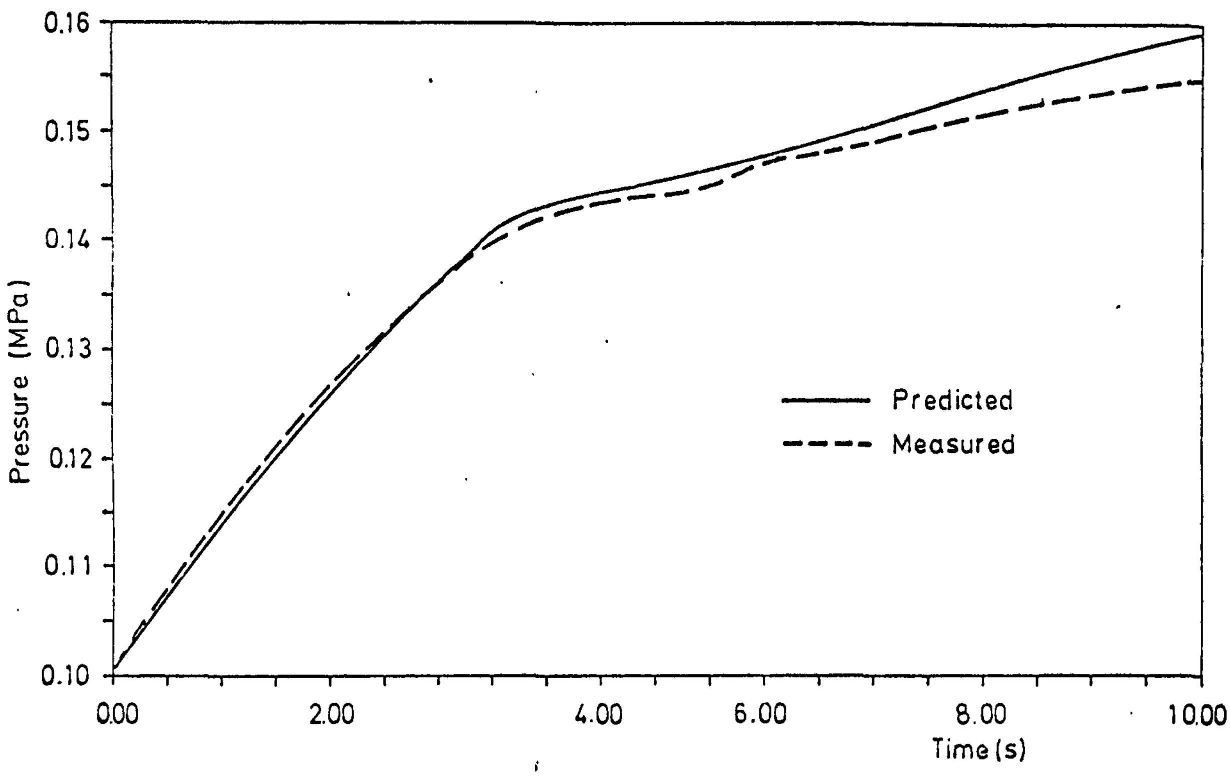

Fig. 1 Comparision of calculated and measured GKSS Test 21 drywell pressure

Table 1:

<u>Comparison of Analytical and Experimental Results of Steam Chugging</u>

Item	SCHUG Model	GKM Experiment 21
1) Chugging Period	1.97 s	1.9 s
Time bubble exists	0.48 s	0.4 s
Time water is in pipe	1.49 s	1.5 s
2) Maximum axial extent of a bubble below the pipe	0.87 m	1.2 m

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

Vorhaben/Project Title	
Kondensation V, Teil 1 Condensation V, Part 1	Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 521, Karlstein

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 4. 76	1. 12. 78	Dr. Simon
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Completed		

1. General Aim

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

2. Particular Objectives

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

3. Research Program

3.1 Condensation forces measurement at realistic back pressures

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

3.3 Visual observations of the phenomena

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

4. Test Facilities

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

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

5. Progress to Date

Evaluation of tests was continued. Special attention was paid to the statistical evaluation of the data obtained from tests 50 - 57 on the pressures in the water phase of the condensation chamber and on the stresses found in the supports of the condenser tubes.

For interpretation purposes the test results were converted to reactor conditions. In that way results obtained from model tests on multi-tube arrangements could be incorporated in the consideration of real conditions. Results gained from the KKB condensation tests were also considered.

In regard to the follow-on tests, pertaining to the adjustment to geometrical conditions for the BWR construction line 72, a concept is being investigated which will meet all requirements on boundary conditions, local conditions and demands of the TÜV-Bayern.

6. Results

The mechanical evaluation of the measurement data stored on magnetic tapes verified the results obtained from evaluation by hand of the influence of the expanded parameter range. The average value of the pressure loads experienced on the walls of the water purifier increases in the absence of back-pressure. A special effect of the by-passing of the pressure chamber is to cause a further increase of the pressure loads. The backfitting of the flexible wall led to a further increase and, with by-passing of the pressure chamber, a connection to the GKM I tests. It was noted that the influence of parameter on the measurement value is not independent of the magnitude of other parameters.

Parameters for the conversion of measurement results to reactor conditions were deduced from the test results. Both the comparison of the GKM II tests with previous tests in NPP Brunsbüttel, as well as the consideration of multi-tube model tests, verify the conservative nature of the computer model for the conversion of single tube measurement results to multi-tube arrangements.

The results obtained by the above test project complete the investigations on the pressure suppression system as requested by the Reactor Safety Commission (RSK) on Nov. 10, 1976.

7. Next Steps

Project completed.

8. Relation with Other Projects

RS 78 E.

9. References

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10. Degree of Availability

Berichtszeitraum/Period 1.10.78 - 31.12.78	Klassifikation/Classification 7.2	Kennzeichen/Project Number GKSS-231
Vorhaben/Project Title DAS-Basisprogramm, theoretische und experimentelle Untersuchungen an Ein- und Mehrrohr-Anordnungen für Druckabbausysteme (DAS) von Kernkraftanlagen PSS basis program, theoretical and experimental investigations on single and multiple vent configurations of pressure suppression systems (PSS) for nuclear power plants		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor GKSS, Geesthacht Institut für Anlagentechnik
Arbeitsbeginn/Initiated 1.10.77	Arbeitsende/Completed 31.12.83	Leiter des Vorhabens/Project Leader Seeliger/Aust
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General aim

The objective of basic experimental investigations is to clarify the pressure suppression process at single and multiple vent configurations. The emphasis of such tests are on phenomena like process of vent clearing, air transport mechanismen, process of pool surface raising and breakthrough, steam condensation by more or less air concentration (chugging) and their stress effect on the containment structure.

In parallel adequate computer codes analysizing the specific phenomena of pressure suppression have to be developed. Verification of these codes will be done by the resulting measured data.

2. Particular objectives

The significant characteristic data like vent-pipe diameter, vent submergence, pool area/vent and pool area/vent area are simulated in a test facility with 3 vents scaled relevant to the real plant. By this simulation single phenomena of the pressure suppression process have to be investigated varying objectively vent mass flow rate, temperature of wet well water and back - pressure. In the main the analytical work will be concerned with the development of a special module for the computer program CORAN describing the chugging-phenomenon and being able to calculate resulting lateral vent loads.

3. Research program

3.1 Experimental investigation in advance on the influence of vent area reduction.

3.2 Alteration of the test facility for the performance of multiple vent tests.

3.3 Main tests with 3-vents flowing simultaneously.

3.4 Main tests with single and two-vents arrangements.

3.5 Basic single vent experiments

3.6 Verification of the computer code CORAN

3.7 Further development of the computer code KONDAS as a model for CORAN

3.8 Fluid-wet well structure interactions caused by dynamic loads of the pressure suppression system.

4. Experimental Facilities, Computer Codes

The experimental work mentioned above by item 3 will be carried out in the pressure suppression facility of GKSS. This facility is going to be altered considerably as to component arrangement and instrumentation. Attendant analytical work is based on the computer code CORAN which has been developed by GKSS in context of designing the concept of a pressure suppression containment for a nuclear ship, and on the program KONDAS which has been developed for analysing condensing phenomena by the Research Center KFK.

5. Progress to Date

Ref. to 3.1 Three runs have been carried out to clarify the influence of vent area reduction.

Ref. to 3.2 Essential work on alterations of the test facility has been concentrated on instrumentation of the wet well during the period of this report. Special attention was called to the installation of high frequency temperature - compensated pressure transmitters, conductibility transducers, drag body for flow measurements and strain gages to determine lateral loads at the vent pipes.
As to alterations of components the installation of a blow down duct provided with a quick opening device for simulating a break should be mentioned. Manufacturing of this duct was requiring a considerable amount of producing and inspection work. Furthermore, the three vents were installed.

Ref. to 3.6 Theoretical work was done to examine the prediction ability of the computer code CORAN by comparing calculations. The module for analysing heat transfer conditions in the containment has been verified by results of the vent area reduction experiment. Furthermore, a study about water-entrainment has been carried out. Program description and instructions for application of the computer code CORAN has been completed.

Ref. to 3.7 Parts of the computer code KONDAS were implemented and tested in the GKSS-computer TR 440.

6. Results

Ref. to 3.1 The experiments have indicated that vent area reduction at the inlet of about 24 p.c. of the pipe diameter does not influence the reflection of pressure waves during chugging incidents and the translation of pressure waves, respectively.

Ref. to 3.2 The alterations of components and instrumentations have transformed the test facility working as a single chamber into an experimental arrangement which makes possible investigations on multiple vent configurations of pressure suppression systems relevant to reactor safety.

Ref. to 3.6 A very good verification of the computer code CORAN has been achieved by calculations for several similar experiments (RS 50 - MARVIKEN / PSS-experiment). Ability of true prediction, easy handling and being proper to quick calculations made the Licensing Board TÜV-Norddeutschland accept and use this code for nuclear licensing purposes.

7. Next Steps

Ref. to 3.2 Alterations of the test facility will be completed. Main activities are calibrating of transducers, removal of failures from transducer-lines, functional tests of instrumentation and data processing as well as tightness test for the whole system.

Ref. to 3.3 Main experiments will be started by 3 runs increasing degree of operation by steps up to conditions of the first main experiment.

Ref. to 3.6 Analytical work will be done using the computer code CORAN in order to predict the experiments in advance and the multiple vent experiments as well as to verify the code by the measured results.

Ref. to 3.7 Further work on the computer code KONDAS is aimed at reproducing the test run of the code in the GKSS-computer TR 440 and recalculating some of those experiments with a significant chugging-phase carried out for nuclear ship purposes as well as at verification of the computer code KONDAS by results of the multiple vent experiments.

Ref. to 3.8 Frequency analysis as to oscillation of the wet well containment structure during the chugging-phase are provided with emphasis on the following topics:

Determination and characterization of oscillation frequency

Investigation of possible interaction between oscillations of the wet well containment structure and the hammering caused by steam condensation.

8. Relation with other projects

RS 78 D
RS 78 E
PNS 4211

9. References

/1/ E. Aust, H.D. Fürst, H.-R. Niemann: Beschreibung des Druckabbau-Versuchsstandes der GKSS für Mehrrohrversuche, GKSS 73 03 AR B 58, Stand Dez. 78

/2/ H. Schwan: Analytische Simulation des Containment-Verhaltens beim Kühlmittelverlust-Störfall GKSS 78/E/36

/3/ E. Aust, H. Schwan, J. Vollbrandt: Beschreibung des Meßkonzeptes zur Erfassung von Druckabbau-Phänomenen, GKSS 73 03 AR B 59, Stand Dez. 78

/4/ J. Vollbrandt: Meßsystem für das DAS-Mehrrohrversuchsprogramm, GKSS 73 03 AR B 57, November 1978

/5/ E. Aust: Theoretische Untersuchung zum Freiblasvorgang der Kondensationsrohre im Druckabbausystem, GKSS 73 03 AR B 50, Stand November 1978

/6/ H. Schwan: Ein modulares Programm zur Analyse des Containmentverhaltens bei Kühlmittelverlust-Störfällen unter Berücksichtigung von Wärmesenken,
GKSS 78/E/44

/7/ H. Schwan: Containment-Code CORAN, Benutzerhandbuch,
GKSS-Bericht Nr. 73 03 AR P 02, Stand Dez. 78

/8/ E. Aust: Experimentelle Untersuchungen zum Druckabbausystem - Versuchsergebnisse des PSS-Großversuchsstandes der GKSS, Vortrag gehalten auf der Reaktortagung in Hannover, April 1978

/9/ H. Schwan: Nachrechnungen einiger Containment-Versuche des Forschungsvorhabens RS 50, Reihe D, mit dem Code CORAN, Vortrag gehalten auf der Reaktortagung in Hannover, April 1978

/10/ E. Aust: Druckabbauversuche auf dem PSS-Versuchsstand der GKSS - Basisprogramm über Mehrrohrversuche für Druckabbausysteme von Landanlagen,
GKSS 78/I/15

/11/ E. Aust: Erste experimentelle Ergebnisse zur Dampfkondensation am Kondensationsrohr mit Einschnürung,
Arbeitsunterlage 73 03 AR B 46, Stand Juni 78

/12/ E. Aust: Planung des Basisprogramms DAS für Ein- und Mehrrohr-Versuche für Druckabbausysteme von Landanlagen,
Arbeitsunterlage Nr. 73 03 AR B 31, Stand Juli 78

/13/ H.-R. Niemann: Berechnung der Abblasleitung DN 100 zur Verbindung des Dampfkessels mit der Druckkammer,
Arbeitsunterlage Nr. 73 03 AR N 02, Stand Juli 78

/14/ E. Aust, H.-D. Fürst: Spezifikation für die 3 Kondensationsrohre DN 600 am PSS-Versuchsstand,
Arbeitsunterlage Nr. 73 03 AR U 22, Stand Febr. 78

/15/ H. Schwan: HETRACO - Ein Programm zur Bestimmung transienter Wärmeübergangskoeffizienten aus gemessenen Temperaturen einer Struktur,
Arbeitsunterlage Nr. 73 03 AR P 01, Stand April 78

/16/ H. Schwan: Anforderungen an Meßtechnik und Dokumentation aus analytischer Sicht,
Arbeitsunterlage Nr. 73 03 AR B 41, Stand Mai 1978

/17/ H. Schwan: Auslegungsrechnung für das DAS-Mehrrohrprogramm,
Arbeitsunterlage Nr. 73 03 AR B 42, Stand Mai 1978

/18/ H. Schwan: Instationärer Wärmeübergang an Strukturen - Gegenüberstellung von Modellannahmen und ihr Einfluß auf den physikalisch relevanten Wärmeübergangskoeffizienten - Arbeitsunterlage Nr. 73 03 AR B 45, Stand Juni 1978

/19/ H. Schwan: Containment-Standard-Problem CASP No. 1, Wiederholungsrechnungen mit dem Code CORAN, Arbeitsunterlage Nr. 73 03 AR B 49, Stand August 1978

/20/ H. Schwan: Thermodynamische Transienten und ihre analytische Simulation am Beispiel des Forschungsvorhabens RS 50, GKSS 78/I/24

/21/ H. Niemann: Vorgesehener Einsatz der Ultraschall-Füllstandssonde im Behälter Wetwell des PSS-Versuchsstandes, Arbeitsunterlage Nr. 73 03 AR B 55, Stand November 78

/22/ H.-D. Fürst: Spezifikation einer 16 mm High-Speed-Kamera für Druckabbau-Untersuchungen am PSS-Versuchsstand, Arbeitsunterlage Nr. 73 03 AR Ü 26, Stand Dez. 1978

Classification: 7.1, 7.2

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

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

Classification: 7.1 7.2

Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: MACON. A Containment Multiroom Transient Analysis Code.	Organization: Risø National Laboratory
Initiated date: 1975 Completed date: Status: Discontinued 1978.	Scientists: V.S. Pejtersen M.B. Andersen

	CLASSIFICATION: 7.2
TITLE (ORIGINAL LANGUAGE): INSTABILITA' CONNESSE CON IL RILASCIO DEL VAPORE ATTRAVERSO LE VALVOLE DI SICUREZZA	COUNTRY: ITALY SPONSOR: CNEN-AMN
TITLE (ENGLISH LANGUAGE): INSTABILITY PHENOMENA RELATED TO STEAM RELIEF THROUGH S.R.V.	ORGANISATION: CNEN PROJECT LEADER: O.TAMPONE

INITIATED:	COMPLETED:	SCIENTISTS:
3/1976		D.PITIMADA O.TAMPONE M.PRESAGHI
STATUS: IN PROGRESS	LAST UPDATING: 6/1979	

1. GENERAL AIM
Experimental study of air ,water and steam discharge,with different discharge line conditions, through one single safety relief valve.
2. PARTICULAR OBJECTIVES
Determination of instabilities connected to air-water clearing,bubble dynamics and to steam flow pulsations.
Study of improving alternative solutions for the discharge system.Implementation of computer codes for the determination of chief parameters interesting the discharge.
3. EXPERIMENTAL FACILITIES AND PROGRAMME
S.P.EX. (Suppression Pool Experiences Loop)consisting of 1.5m^3 boiler (70kg/cm^2), 2" relief valve,35 m long,1.5"discharge pipe, different types of discharge devices, suppression pool.
Small suppression pool VASCA (1.5m^3),1/2"discharge pipe,interchangeable relief valves,compressed air tank.
4.PROJECT STATUS
The facilities are operating,results on water-air clearing bubble dynamics and steam condensation instabilities are available. Further results are in progress.
A computer code,concerning transient phenomena will be soon available.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
INSTABILITY PHENOMENA RELATED TO STEAM RELIEF THROUGH S.R.V.	7.2.

5. NEXT STEPS

Determination of water velocity, loads on pool wall, air and steam clearing parameters in different conditions inside the discharge line. Study of bubble formation and evolution, subsequent actuations of the S.R.V. Alternative solutions for steam discharge system.

6. RELATION TO OTHER PROJECTS AND CODES

1 BOLLA-REVACO-BOLLAX-WAGET-PRECC codes
2 AMN research on safety relief valves discharge phenomena, Pisa University research on pressure suppression, ecc.

7. REFERENCE DOCUMENTS

E.C.MASSA'-M.PRESAGHI-O.TAMPONE - Impianto sperimentale S.P.EX. Descrizione e manuale di operazione. CNEN78(7)VBE1/G71A-5

C.KROPP Contributo alla proposta sperimentale CNEN sullo spegnimento vapore. AMN SNU/77-6.

D.PITIMADA-M.PRESAGHI-O.TAMPONE - Rapporto relativo alla prima fase sperimentale sul circuito S.P.EX. - CNEN 78(2)VBE1/G71A-1

R.PARUTTO-D.PITIMADA - Proposta di utilizzazione del circuito CEV2 per la determinazione dei campi di velocità in getti sommersi. Nota Tecnica CNEN 78(10)VBE5/G71A-6

A.LUPI-O.TAMPONE - Proposta di sistema per il posizionamento di trasduttori di pressione nella vasca SPEX per il rilievo dei transitori di pressione nella fase di espulsione del getto d'acqua. Nota Tecnica CNEN 78(11)VBE6/G71A-7

M.PRESAGHI- Studio di rivelatori di livello e velocità dell'acqua all'interno del tubo di scarico del sistema di soppressione del vapore. Nota Tecnica CNEN 78(15)VBE8/G71A-9

D.PITIMADA-M.PRESAGHI-O.TAMPONE - Safety relief valve experimental studies. Second phase tests conducted on S.P.EX. (Suppression Pool Experiences) facility. CNEN 78(17)VBE-10/G71A-11

G.GIRARDI-D.PITIMADA- Codice di calcolo TUBO per la valutazione della trasmissione di potenza termica nella tubazione della S.R.V. D.L. nel transitorio conseguente la prima attuazione delle S.R.V. Nota Tecnica CNEN 78(16)VBE9/G71A-10.

8. DEGREE OF AVAILABILITY

Restricted distribution. Requests of information can be addressed to: O.TAMPONE, TERM/RIS Lab. Fluidodinamica e Vibrazioni, CSN Casaccia, CNEN C.P. 2400, I-00100 ROMA

9. ADDITIONAL INFORMATION

Studies are carried out in the frame of CNEN-AMN research agreement.

	CLASSIFICATION: 7.2
TITLE (ORIGINAL LANGUAGE): SOPRE 1 - Ricerca sul comportamento del sistema di contenimento a soppressione di pressione in caso di LOCA.	COUNTRY: ITALY
	SPONSOR: CNR-CNEN
TITLE (ENGLISH LANGUAGE): SOPRE 1 - Research on the behaviour of pressure suppression containment system after LOCA.	ORGANISATION: UNIVERSITY OF PISA
	PROJECT LEADER: Marino MAZZINI

INITIATED: (second phase) 1978	COMPLETED: (second phase) 1979	SCIENTISTS: R. BOVALINI, F. D'AURIA
STATUS: in progress	LAST UPDATING: January 1979	

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
   To investigate the pressure and temperature transient 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.
   Prepare new computer model able to forecast dynamic phenomena like pool swelling and associated pressure oscillations.

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 2nd and 3rd series of tests with blow-down nozzle diameter of 50 mm (10 runs) were carried out, with 55 kg of water in PIPER vessel and varying submergence of vent pipes (from 11 to 34 cm) and pool temperature (from 30 to 70 °C); initial water pressure inside the PIPER vessel was varied in each cycle from 30 to 70 kgp/cm^2 for safety reasons.
   The experimental results were compared with data from CONTEMPT-LT and SOPRE SD1 codes (the last is a studied in order to describe dynamic phenomena experimentally found).

5. Next steps
   Next program includes 7 runs with blow-down nozzle of 50 mm diameter, 55 or 70 kg of water inside the PIPER vessel and a starting pressure

45.897

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
SOPRE 1 - Research on the behaviour of pressure suppression containment system after LOCA.	7.2

from 20 to 85 kg_p/cm^2; the aim of these series of tests is the study of the influence of the number of the vent pipes.

6. Relation to other projects
Blow-down tests by PIPER Apparatus (the facility and instrumentation are the same) - Project Leader P. Vigni.
CNEN - Research on pressure suppression.
Analysis of thermal and hydraulic transients following a LOCA in LWR and associated containment systems. P. Leader N. Cerullo.

7. Reference documents

1- N. CERULLO et alii
"Experimental Investigation of the Behaviour of Pressure Suppression Containment Systems by the SOPRE 1 Facility"
Proceedings of the 4th SMIRT Conference, S. Francisco (USA), August 77.

2- N. CERULLO et alii
"Research on the Behaviour of Pressure Suppression Containment Systems Carried out at the University of Pisa: Comparison between Experimental Results and CONTEMPT-LT Calculations"
Istituto di Impianti Nucleari dell'Università di Pisa, RP 290(77).

3- R. BIGI et alii
"Research on the Behaviour of Pressure Suppression Containment Systems Carried out at the University of Pisa: Comparison between Experimental Results and SOPRE-SD1 Calculations"
Paper presented at the joint meeting ANS-ENS, Bruxelles, October 78.

4- M. MAZZINI et alii
"SOPRE 1: Ricerca sul sistema di contenimento a soppressione di pressione. Relazione sulla seconda serie di prove"
Istituto di Impianti Nucleari dell'Università di Pisa, RL 340(79).

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

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

	CLASSIFICATION: 7.2
TITLE (ORIGINAL LANGUAGE): Studi sui fenomeni connessi allo scarico delle valvole di sicurezza nei BWR	COUNTRY: Italy SPONSOR: AMN-CNEN
TITLE (ENGLISH LANGUAGE): Safety relief valves discharge phenomena	ORGANISATION: AMN PROJECT. LEADER: A. Venturini/C.Kropp

INITIATED:	COMPLETED:	SCIENTISTS:
January 1977		G. Giresini
STATUS: In progress	LAST UPDATING: July 1979	

1) General Aim

Analytical studies on experimental data.

2) Particular Objectives

Development of analytical models and computer codes implementation to improve the project tools and to suggest particular new solutions.

3) Experimental facilities and programme.

See project: Instability phenomena related to steam relief through SRV. The programme concerning new researches about some aspects is being studied at present.

4) Project Status

See project: Instability pheromena related to steam relief through SRV. Analytical models have been developed and compared with test data, showing a good agreement.

5) Next Steps

- Load combinations due to multiple valves actuation.
- Computer codes development.
- New solutions development.
- Analytical activities concerning new calculation methods.
- Design guides for SRV discharge lines.

6) Relation to other Projects and Codes

- Instability phenomena related to steam relief through SRV.
- REVACO code.
- PRECC code.
- WAGET code.
- BOLLAX code.

TITLE (ENGLISH LANGUAGE) :	CLASSIFICATION :
Safety relief valves discharge phenomena	7.2

7) Reference Documents

- "Le "Safety Relief Valves" nei BWR", 22/12/77. Doc. AMN IS 77-03.
- "Note sullo stato delle analisi dei risultati sperimentali sullo SPEx". AMN-Nota tecnica interna.
- "Alcune considerazioni sul transitorio di getti d'acqua sommersi". Doc. AMN-NT/RNU7801, 5/4/78.
- "Contributo alla proposta sperimentale CNEN sullo spegnimento vapore". Doc. AMN-SNU 77-6, 29/7/77.
- "Simulazione del transitorio di scarico vapore nell'impianto SPEx mediante il codice RELAP4/MOD5". Doc. AMN-NT/RNU 7806, 29/5/78.
- "Descrizione dei codici OSCI e BOLLAX per lo studio delle oscillazioni di bolle di gas sotto battente idraulico". NT-RNU 78-12.
- "Descrizione del codice PRECC (Pressure computer code) per la valutazione dei campi di pressione sulle pareti di una piscina in presenza di bolle d'aria". NT-RNU-78-13.
- "Definizione dei principali aspetti da indagare per lo sviluppo della soluzione impiantistica "clean steam". NT-RNU 78-08.
- "Safety relief valve discharge study. Pool dynamic analysis of SPEx. Test results (First phase)". IV-7802 (AMN).

8) Degree of Availability

Restricted distribution on some of documents above.

(A. Venturini, AMN-VPR/RTI, V. G. D'Annunzio 113, I-16121 Genova).

9) Additional Information

Studies are performed in the frame of CNEN-AMN Research Agreement.

	CLASSIFICATION: 1.1.1,1.1.2,1.1.4,1.2,7.2
TITLE (ORIGINAL LANGUAGE): Analisi dei transitori termici ed idraulici a seguito di LOCA nei reattori ad acqua leggera e nei sistemi di contenimento associati.	COUNTRY: ITALY SPONSOR: CNEN and CNR
TITLE (ENGLISH LANGUAGE): Analysis of thermal and hydraulic transients following a LOCA in L.W. reactors and in associated containment systems.	ORGANISATION: University of Pisa PROJECT LEADER: N. CERULLO
INITIATED: 1974 / COMPLETED: 1980 STATUS: in progress / LAST UPDATING: May 1979	SCIENTISTS: F. ORIOLO, G. MEI, G. CECCARELLI, G. BITETTI

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

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	7.3	RS 223

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

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 7. 1976	31. 5. 1978	H. Queiser
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	31. 12. 1978	

1. General Aim

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

2. Particular Objectives

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

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

3. Research Program

3.1 Theoretical Investigations

H_2-production

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

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

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

H_2-measurement
In connection with paragraph 3.2, suitable sampling and measurement methods will be selected, with special attention to post-LOCA conditions.

3.2 Experimental Investigations
Tests are planned in a suitable semi-technical lab-facility which is similar to the KWU off-gas system recombiner. The major components of this facility are: preheater, recombiner, cooler and the required measuring devices. The following investigations will be carried out:
Thermal capacity of the total system:
Evaluation of catalyst and recombiner material properties, both under normal and extreme temperature conditions of 50 to 600 °C consistent with possible failure modes.

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

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

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

4. Experimental Facilities

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

The arrangement consists of a proportioning apparatus for varying the hydrogen enrichment of the atmosphere. The air-H_2 mixture will be saturated with steam at 80 ^{O}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 poisoning Iodine will be determined by volumentric analysis. Hydrogen analysis will be performed by a continuously operating thermal conductivity measuring device.

5. Progress to Date

To 3.1 H_2-evolution

Determination of the exposed zinc surfaces in the power plant.

Preparation of a test program that will determine the zinc-water reaction under failure conditions and the specification of temperature behaviour.

Performance of several experiments in a KWU-owned experimental facility to determine zinc corrosion.

H_2-Distribution

Calculation of the local concentration differences considering diffusion and solubility transport.

H_2-distribution in the containments as a result of the action of a hydrogen separation system (WAS) (determination of the purification rate).

H_2-Measurement
Determination of poisonous material concentration as observed under conservative boundary conditions for noble gas and iodine decay properties.

To 3.2 With a bed height of 30 cm the temperature behaviour in the through-flow catalyst bed was examined under the following conditions:

- failure of the preheater
- failure of the complete heating jacket
- failure of a half section of the heating jacket.

Heating transfer properties with the catalyst bed flooded: Prior to test start the whole recombiner was flooded and after a certain length of time was emptied again. For the original test bed arrangement and with fully functioning contents, the local temperature distribution within the flooded catalyst bed and its heating rate up to the required temperature level was determined.

6. Results

To 3.1 H_2-evolution
Results of the experimental investigations of the zinc-corrosion.

surface:	medium	H_2-evolution
grid (galvanized) with approx. 80 μm Zn	deionized	0.95 ml/cm^2
St-37 (galvanized) with approx. 25 μm Zn	deionized	0.94 ml/cm^2

In order to obtain reproduceable results, all tests were performed at least two times.
The above-mentioned evolution rates indicate an average hydrogen volume concentration of 0.8 percent in the safety containment of the ISAR-type nuclear power plant.

H_2-Distribution
Diffusion and solubility transport
Under conservative H_2-evolution assumptions it appears that, because of the actual free cross-section, there are no unpermissible concentration variations within the pressure- or condensation-chamber. The solubility transport is responsible for the gas exchange between pressure and condensation chamber following completion of the "blow-down phase". Diffusion observations verify the assertion that subdividing the computer model into two large volumes (pressure chamber and condensation chamber) is adequate for evaluation of the H_2-distribution in the safety containment.

H_2-distribution using a hydrogen separation system.
Based on the above distribution mechanisms the concentration distribution during recombination operation were evaluated. With it the flow rate of the decontamination facility was modified such that with an average H_2-volume concentration of > 2 % any further increase of the H_2-concentration can be prevented at any time after an accident has occurred. The required flow rate of the hydrogen separation system is 80 m^3/hr.

H_2-Measurement
Possible contaminations in the suction line of the hydrogen separation system.
With respect to radioactivity the following substances will be found in the post-accident atmosphere. They are basically released from the reactor core.

isotope group:	theoretical conentration/kgm^{-3}
radioactive noble gases	$< 10^{-4}$
active and inactive noble gas follow-on isotopes	$10^{-5} - 10^{-4}$
iodine	5×10^{-5}

The range of noble gas products results from the different molecular weight of each compound which these elements are able to form.

It was suggested that tests should be performed with the above concentrations maintained for a maximum of 60 hrs, so that it is assumed that no more than the total amount reloaded is expected to attack the recombiner. The above values are based on a 10 % isotope release from the core inventory.

To 3.2 Preheater malfunction during test start-up
A rapid temperature increase to approx. 80 °C can be observed on all temperature measurement points when starting the compressor. Then temperatures in the upper bed area increase to 150 °C within 4.5 hours. The measurement points in the lower area show these temperatures approx. two hours later.

Preheater failure during test run
Within three hours temperatures drop from approx. 280 °C to 150 °C in the upper bed area. The lower region reaches these temperatures approx 1 hour later.

Overall failure of jacket heating
Jacket heating dropout results in a reduction of the temperature level from 280 °C to approx. 250 °C within approx. three hours. Different temperature layers in the catalyst bed are not observed.

Failure of a half-section of the heating section
This failure initiated a temperature drop of approx. 20 °C. Despite the missing external heating of the half-section, the differences between the individual temperature measuring points amount to a maximum of 10 °C only.

Heat transition properties with flooded catalyst bed
After the residual water has evaporated, steeper temperature gradients can be observed as compared to the "dry test". In the upper bed region, temperatures of approx. 250 °C will be reached after approx. 3.5 hrs. The total catalyst bed will reach its maximum temperature of 300 °C after approx. 10 hrs.

A comparision to the "dry test" shows that heat-up of a flooded catalyst bed of the design size will take only about 0.5 hrs. longer.

7. Next Steps

Project completed.

8. Relation with Other Projects

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9. References

R. Heck, W. Rüffer, P. Sydow
Investigation and development of systems for limitation of the hydrogen concentration in the BWR containment
Final report

10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	7.3	RS 246

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

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.8.1977	28.2.79	G. Langer
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	DM 648.900

1. General Aim

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

2. Particular Objectives

–

3. Research Program

The scheduled research program includes nine basic experiments to be performed at room temperature. The hydrogen, also at room temperature, is fed from a single large-area source into a central containment compartment. The local hydrogen concentration is measured by numerous detectors. The experiments are scheduled to last between 5 and 10 hours. The experimental configurations include one- and two-compartment models that extend vertically or horizontally. The following experimental parameters are varied:

- location of the hydrogen source
- [illegible]
- [illegible] of the containment compartments and the connecting

overflow openings.

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

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

4. Experimental Facilities, Computer Codes

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

The hydrogen supply system is operated with a bottle battery.

Pre- and post-experiment calculations are performed partly at the Gesellschaft für Reaktorsicherheit in Munich with the computer code RALOC, and partly by the Kraftwerk-Union (KWU).

5. Progress to Date

The following work packages have been completed in the period under review:

- assembly of the internals of the containment and installation of the detectors;
- [illegible] basic experiments, including 6 using 2 cylindrical compartments in vertical tandem arrangement and 3 using 2 toroidal compartments in horizontal tandem arrangement; in both cases the total volume of the two compartments was 80 m^3;

- preparation of 2 quick-look reports including a first evaluation of the experimental results.

6. Results

The following important results should be noted:

- Hydrogen spreads relatively fast both in the vertically and in the horizontally arranged compartments (homogenous distribution).
- The dispersion is hampered only if the connecting openings between the two compartments are very small (< 0.5 m^2).
- Thermal stratification in model compartment (temperature at the top higher than at the bottom) clearly hampers hydrogen dispersion.
- If the hydrogen source is so arranged as to produce strata of different concentrations, equalization due to diffusion is relatively slow ($t > 24$ h).

7. Next Steps

Writing of the final report with detailed evaluation of the results.

8. Relation with Other Projects

Research activities at the Gesellschaft für Reaktorsicherheit in Munich toward further development of the computer code RALOC.

Program RS 50 "Investigation of the phenomena occurring within a multi-compartment containment following the rupture of a coolant pipe in a water-cooled reactor".

9. References

Jahn, H. L.: RALOC - A New Model for Calculation of Local Hydrogen Concentrations in Subdivided Containments Under LOCA Aspects. Paper presented at the Thermal Reactor Safety Meeting, Sun Valley, Idaho, August 1977

10. Degree of Availability of the Reports

-

	CLASSIFICATION: 7.3
TITLE (ORIGINAL LANGUAGE): Controllo della concentrazione di idrogeno nel contenitore dopo LOCA.	COUNTRY: ITALY
	SPONSOR: CNEN-CNR
TITLE (ENGLISH LANGUAGE): Control of Hydrogen Concentration in Containment Following a LOCA	ORGANISATION: University of Pisa
	PROJECT LEADER: Salvatore LANZA

INITIATED:	COMPLETED:	SCIENTISTS:
1975		Fabio FINESCHI
STATUS: in progress	LAST UPDATING: June 1979	

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
   Similarity problems in hydrogen mixing.

3. Experimental facilities and Programme
   3.1 Facilities:
   - PSICO 10 facility
   - hydrogen detection assembly
   - hydrogen injection system

   3.2 Program: Theoretical considerations on:
   a) scaling up experiments with hydrogen-air mixtures;
   b) using the other gases in simulating hydrogen behaviour.

4. Project status
   Item 3.2a) is the point to be completed.

5. Next steps
   Improvement of the detection performance, i.e. type and number of detecting devices.
   Analysis of results and comparison with other researchers'.
   Mathematical modelling of the phenomenon.
   Supporting experiences.

6. Relation to other projects and codes: none

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Control of Hydrogen Concentration in Containment following a LOCA	7.3

7. 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 di Impianti Nucleari, Pisa RP 286(77).

2- BIANCHI F. et alii
"Esperienze di trasporto dell'idrogeno all'interno del contenitore PSICO 10"
Istituto di Impianti Nucleari, Pisa RP 296(77).

3- FINESCHI F., LANZA S.
"Mixing of Hydrogen in a closed containment"
Paper presented at ANS/ENS Meeting on Nuclear Power Reactor Safety, Brussels 16.10.1978.

4- GIOVANNETTI S. et alii
"Misura della concentrazione di gas (H_2, H_e) all'interno del contenitore PSICO 10 per gascromatografia"
Istituto di Impianti Nucleari, Pisa HYMT02/76.

5- BIANCHI F. et alii
"Risoluzione dell'equazione di diffusione: I parte"
Istituto di Impianti Nucleari, Pisa HYMT03/76.

6- MARTINO N. et alii
"Rassegna dei sistemi di rivelazione dell'idrogeno all'interno di un ambiente chiuso"
Istituto di Impianti Nucleari, Pisa HYMT04/76.

7- BIANCHI N. et alii
"Risoluzione dell'equazione di diffusione: II parte"
Istituto di Impianti Nucleari, Pisa HYMT05/76.

8- MARTINO N. et alii
"L'impiego di rivelatori a semiconduttore per la misura della concentrazione dell'idrogeno nel contenitore PSICO 10"
Istituto di Impianti Nucleari, Pisa HYMT07/76.

8. Degree of availability
Papers 1-3 are free, while papers 4-8 can be available with the authorization of the sponsoring organizations.

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

Classification 7.3

Title 1	Water Radiolysis and Hydrogen Generation under Accident Conditions	Country	U.K.
		Sponsor	C.E.G.B.
Title 2		Organisation	BERKELEY NUCLEAR LABORATORIES
Initiated 1976	Completed	Project Leaders	
Status Continuing	Last updating	Dr. C.J. Wood Dr. T. Swan	

1. General Aim

To measure the radiolysis rates of aqueous phases in water reactors under both normal and accident conditions.

2. Particular Objectives

To measure the rate of production of H_2 from the radiolysis of contaminated water in water reactor cores after a LOCA.

3. Experimental Facilities and Programme

Febetron Pulse Radiolysis Facility in which the kinetics of radiation induced processes can be followed by kinetic spectroscopy. The influence of likely contaminants on the H_2 generation rate will be measured.

4. Project Status

Experiments are presently limited to room temperature tests. The apparatus will be modified to allow elevated temperatures to be investigated.

142.2.01/4111.6.11		7-4
TITRE PIEGEAGE DE L'IODE DANS LE BETON.		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) IODINE - TRAPPING IN CONCRETE.		Organisme exécuteur CEA/DTech/STA Responsable
Date de démarrage 1/75	Etat actuel EN COURS	Scientifiques
Date d'achèvement FIN 79	Dernière mise à jour 12/78	

1. OBJECTIF GENERAL

Cette étude a pour but :
de déterminer la rétention de l'iode par le béton des enceintes de confinement afin de mieux évaluer les conséquences radiologiques en cas d'accident de perte de réfrigérant primaire des réacteurs à eau PWR.

2. OBJECTIFS PARTICULIERS :

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

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

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME :

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

Le programme comprend les étapes suivantes :

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

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

4. ETAT DE L'ETUDE :

1/ Avancement à ce jour :

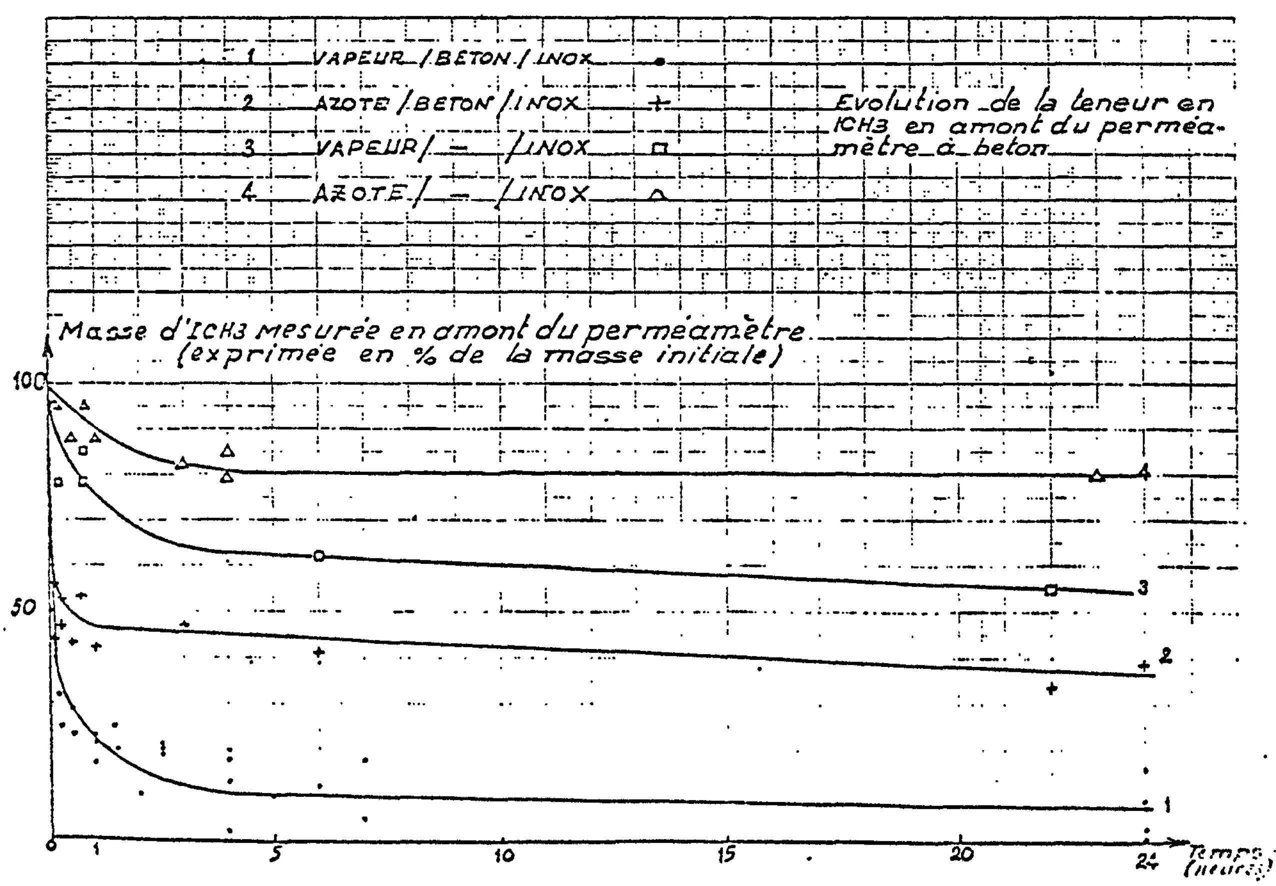
Les principaux travaux réalisés à ce jour sont les suivants :

- poursuite du programme de dosage de l'iodométhane en phase gazeuse et identification des zones de piégeage (phase solide : béton, joint et phase liquide : eau de condensation).
- 18 essais d'injection d'ICH3 ont été effectués durant cette période
- mise au point du dispositif de dosage de l'iode moléculaire. Début des essais de piégeage.

2/ Résultats essentiels :

Les résultats acquis en 78 permettent de mettre en évidence les conclusions suivantes :

- dans les conditions de l'accident de référence (4 bars 140° C) la diffusion de l'iodure de méthyle à travers une épaisseur de 20 cm de béton est inférieure à 1 % de la quantité injectée.
- la série d'essais effectuée en 78 pour déterminer les facteurs susceptibles d'entraîner la disparition de l'iodure de méthyle dans la partie amont du perméamètre (piégeage par le béton, le joint ou la vapeur d'eau) est résumée sur les courbes suivantes :



- on peut noter en conclusion que les conditions d'ambiance dans l'enceinte après l'accident de référence sont les suivantes :

  . les réactions de l'ICH3 avec les matériaux (béton, eau) sont pratiquement terminées 5 heures après l'accident,

  . après 24 heures, on constate (courbe 1) que 90 % de l'ICH3 est piégé (béton et eau)

  . l'influence du piégeage par le béton est prépondérente. Elle intervient au moins pour 50 % (courbes 1 et 3).

- les mesures effectuées avec l'iode moléculaire ont montré que 99 % de l'I_2 était pratiquement fixée instantanément sur les parois du perméamètre.

5. PROCHAINES ETAPES :

Lors de la réunion du 5 décembre 78, (note STA/78.1141 du 22/12/78) le comité technique a décidé d'engager les actions suivantes :

51. Programme ICH3

- Etude de la pénétration de l'iodométhane dans le béton en utilisant I_{131} comme traceur.
- Les modalités de définition de cet essai sont en cours d'examen (lieu des essais, carottage des éprouvettes, mesures).
- En fonction des résultats acquis, étude éventuelle de la désorption de l'iode piégé dans le béton.

52. Programme I_2

- Poursuite du programme avec l'iode stable pour confirmer les premiers résultats et déterminer la cinétique d'absorption sur les parois (cas de l'enceinte avec peau d'étanchéité)
- mesures avec I_2 stable sur perméamètre béton (utilisation des perméamètres de la STA ou construction d'une enceinte béton). Il s'agit d'évaluer la cinétique d'absorption sur du béton seul (cas de l'enceinte double).

6. RELATIONS AVEC D'AUTRES ETUDES :

Néant.

7. DOCUMENTS DE REFERENCE :

Les documents établis en 78 sont les suivants :

- DSN-SETSSR-T-78.1663 du 27.1.78 : programme d'essais 78
- Notes DRA/SEA/SCAA 78.42 du 27.4.78 } définition des essais complémen-
  " DRA/SEA/SCAA 78.67 du 14.6.78 } taires avec ICH3
- Note STA/78.527 du 15.6.78 : compte rendu de réunion de coordination
- Note DSN/SESTR 78.102 du 8.6.78 } procès verbaux d'essai
  Note DSN/SESTR 78.590 du 16.11.78 }

- Note DSN/SESTR/78.559 du 22.11.78 : Examen des résultats des essais 78
- Note SEA/SCAA/78.136 du 29.11.78 : Mesures effectuées par SCAA
- Note STA/78.1141 du 22.12.78 : Compte rendu de réunion de coordination.

8. DEGRE DE DISPONIBILITE :

Les procès verbaux établis par SESTR et SCAA sont des documents internes de travail.

NETHERLANDS ENERGY RESEARCH FOUNDATION (ECN)		CLASSIFICATION : 7.4
TITLE : Advies inzake periodieke kontrole van insluitvaten op lekkages		COUNTRY: THE NETHERLANDS
TITLE (ENGLISH LANGUAGE): Advice on periodic supervision on leakage tightness of containers.		SPONSOR : Ministry of Social Affairs ORGANIZATION : ECN PROJECTLEADER : H.J. van Grol
INITIATED : June 1974	LAST UPDATING : June 1979	SCIENTISTS : J.W.H. van den Bergh, H. Pruimboom
STATUS : ending	COMPLETED : end 1980	

General aim

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

2. Particular objectives

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

3. Experimental facilities:Not applicable

4. Project status

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

5. Next and final steps

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

6. Relation with other projects: -

7. Reference documents: See under 4 and 5

8. Degree of availability

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

9. Budget: -

10. Personnel: 1.0 manyear

8. INSTRUMENTATION, CONTROL AND COMPUTERISED PROTECTION

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	8	RS 240
Vorhaben/Project Title Untersuchungen zur Funktionstüchtigkeit der Druckhalter-Sicherheitsventile und des Abblasetanks bei Abblasen von heißem Druckwasser Investigations on the Functioning of the Pressurizer Safety Valves and the Relief Tank during Blowdown of Hot Pressurized Water		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 523, Erlangen R 111, Erlangen
Arbeitsbeginn/Initiated 1. 12. 76	Arbeitsende/Completed 29. 2. 80	Leiter des Vorhabens/Project Leader Landgraf/Helf
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 78	Bewilligte Mittel/Funds

1. General Aim

Verification of ability of PWR pressurizer safety valves and relief tank to function during blowdown with 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 the structural changes to the main safety valves, the control valves and arrangements necessary in order to retain the response characteristics in the event of a rapid change of phase (steam-hot pressurized water) of the blowdown medium.

In parallel to the above tests, the pressure build-up in the discharge line and the relief tank will be determined analytically.

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 pressure 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

Tests were run with steam at 170 bar to set the required pressure gradient. After this, tests with a pilot safety valve were made for the vapor phase at an opening pressure of 170 bar.
The pilot safety valve for reference tests at a lower pressure was modified; the opening pressure set to 150 bar and tests were run. Differential pressure measuring points were newly installed so that the pressure gradient and opening pressure for the tests could be determined more accurately.

During the tests, some disturbances occurred on the testing and measuring rig, involving a series of non-scheduled operations: several inspections of the main safety valve and the pilot safety valve were required on account of intensified opening pressure scatter (measurement channel VI).
As the tests had stopped for a prolonged period of time, all the pickups were removed from the testing rig and recalibrated. A check was made of the measuring chain from the pickups to the recording instruments.
The main safety valve and the pilot valve were dismantled, the sealing surfaces ground, and re-assembled.
A closed-circuit television facility with video recording was installed for a better function at assessment of the valves.

To 3.2 The computing program BABS was developed and tested, which not only calculates the pressure rise in the pressurizer relief tank dome while it is being blown free water, but also the pressure rise in the pressurizer relief tank after free blowing. The BABS computing program is made up of the individual computing programs developed to date, DOMFREI and ABDRUNA, with the modified versions of DOMFREI and ABDRUNA being used in each case.

6. Results

To 3.1 Tests to date have shown that the test rig instrumentation and data acquisition equipment are suitable for simulating with high accuracy the parameters of a reactor system and for making an assured statement on the mode of functioning and loads imposed on the pilot valve and piping during the assumed accidents and malfunctions.

Because of the complexity of the processes involved statements on pilot valve functioning can only be made with reservations at the present status

- Tests with saturated steam:
  The pilot valve demonstrated the typical opening behaviour of a full-stroke safety valve with the limits of the permissible differential opening pressure (the valve opens steadily to the stop position). The pressure reduction (due to pipe friction and dynamic flow) measured in the inlet branch (signal line) during the opening process, does never reach the permissible full stroke safety valve closing pressure and this shows, therefore, a stable opening behaviour. The opening pressure scattered in a number of tests despite an unchanged valve setting.

- Tests with hot pressurized water:
  The pilot valve opened only gradually with water to the design stroke limit. The pressure gradients were here approx. 10 bar/s (requirement 4 bar/s). A higher

pressure gradient is to be identified with high excess energy assisting the valve opening.
The pressure reduction (as a consequence of pipe friction and dynamic flow) measured in the inlet branch signal line) in the opening process, reaches at times approximately the pilot valve closing pressure and explains why opening is gradual.
Tests with pressure gradients of 4 bar/s must still be run to assess definitively the valve behaviour during relieving hot pressurized water.
Forecasts relating to the behaviour on transition from steam to hot pressurized water are not yet possible even today.

To 3.2 Computing program BABS is at present in the trial phase. Results are available of test runs made until now.

7. Next Steps

To 3.1 Commissioning of the test rig and measuring set-up.
Check of response pressure of the pilot valve using saturated steam.
Conduct of tests with hot pressurized water and saturated steam with transition to hot pressurized water.

To 3.2 An attempt will be made after completion of the BABS test runs to prepare the report on results.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	8	RS 281
Vorhaben/Project Title Schutzbegrenzungs-System mit Kleinrechnern Protection Limitation System with Mini-Computers		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 1, Erlangen
Arbeitsbeginn/Initiated 1. 7. 77	Arbeitsende/Completed 30. 6. 80	Leiter des Vorhabens/Project Leader W. Aleite
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 78	Bewilligte Mittel/Funds

1. General Aim

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

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

2. Particular Objectives

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

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

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

3. Research Program

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

3.2 Development of portable programs (programming language)

3.3 Model for reliability verification

3.4 Optimization of systems by means of the reliability model.

4. Test Facilities

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

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

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

5. Progress to Date and 6. Results

To 3.1 Transcription of the currently available program into a CP550 with little extension of storage completed. Modifications to the KWU-CP-550 compiler will only be required for Bit 13 storage, a special storage in the control and instrumentation system.

A draft of the specification for the monitoring units was prepared. With it were defined the points of the systems with which the DNB module is associated.

Cabinets for the computer system were installed in the laboratory. Current is supplied by a 24 V/200 A grid section with bus bars between floor and false floor. Cabinet IH 50 was chosen for recording block in Grafenrheinfeld and connected to the grid.

The process signals cabling was reworked to separate the binary and analogous signal cables. For recording of the thermoelements a page printer cable and control lines were provided between the recording block (IH 50) and computer room.

A design was produced for the component carrier for the thermoelement signal distribution.
A program for the determination of the DNB calculation constants was put into operation. Revision of the overall system tasks is being continued.

Within the scope of the PDV working group "testing and verifying of process computer soft ware" a comparison was made between the analyses of the Brunsbüttel computer protection system and the C-E core protection calculator, entitled "computer systems in nuclear power plants and their appraisal".

7. Next Steps

To 3.1 Erection of the 4-computer system for protection against DNB and high power density.

- Start-up of measurement data acquisition and processing programs on the KWU PR 320 in cooperation with the analog computer (ANDI-80-model, circuit model) and correction of the program system based on findings to date.

- Erection of the computer to computer coupling between the PR 320 and CP 550. This coupling assists the start-up of a Closed-Loop-Computer, which has to operate without a recorder device.

- Erection and check-out of the computer self-monitoring system and the hard-ware monitoring section with the aid of the PR 320.

- Transfer of the program system into the CP 550 in the laboratory, first, channel by channel, and then by multiple channel test and verification calculations.

- Systems evaluation with respect to area of operation, safety of operation, disturbances
  Special attention is planned for the examination of input signals, long-term behaviour of the structural elements and transient disturbances.

To 3.2 The experience gained during the construction and examination of the system is to generalized so that they can also be used in the construction of other computer systems. In this connection a diversity of computers has to be programmed.

To 3.3 In cooperation with the GRS (Reactor Safety Organ.) key topics are supposed to be identified which will contribute to safety, simpicity, care of handling and possibly standardization of the computer system. Relations between design features and reliability data are to be established. This can lead, for example, to restrictions on the complexity of the programming language.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.10.78 - 31.12.78	8	RS 351
Vorhaben/Project Title Rahmenpflichtenheft zur KKW-Leittechnik Specification for instrumentation and control in nuclear power plants		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Brown Boveri & Cie AG Mannheim GK/TE2
Arbeitsbeginn/Initiated 1.10.78	Arbeitsende/Completed 31.12.78	Leiter des Vorhabens/Project Leader H. Zimmermann
Stand der Arbeiten/Status completed	Berichtsdatum/Last Updating 31.12.78	Bewilligte Mittel/Funds

1. General Aim

With the instrumentation and control systems used in nuclear power plants at the time being the possibilities to meet the increasing demands on safety and availability are exhausted, to a far extent.
On that account, the aim of the project is to develop new instrumentation and control systems which satisfy the requirement for greater operational safety and which, at the same time, can be realized in an economical way.

The aim can be attained by means of new system structures with the following solution characteristics:

- BUS-data transmission systems
- sequential processors with semiconductor stores
- modern means of communication and documentation (e.g. display units)
- application of highly developed programming languages

2. Particular Objectives

Taking into account the possibilities of modern technologies, a basic requirement book will be compiled as basis for the development of new instrumentation and control systems specific for nuclear power plants.

This basic requirement book is supposed to serve several purposes:

a) To agree upon uniform basic conditions and solutions which can be applied universally and help to improve the prerequisites for the standardization of the instrumentation and control systems. These specifications, however, should still leave enough latitude to realize various technical instrumentation related solution.

b) To allow for all the requirements of the authorities giving the approval and for the extension of existing regulations and guidelines in order to ensure the approval of new systems.

3. Research Programm

3.1 Determination of the scope of application
3.2 Elaboration of the premises for the technical instrumentation related solution
3.3 Listing of the functional process requirements
3.4 Listing of the requirements related to plant erection and environmental influences
3.5 Listing of the safety and reliability requirements
3.6 Compilation of the specifications and guidelines to be observed

4. Experimental Facilities

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5. Progress to Date and 6. Results Obtained

the results appertaining to 3.1 to 3.6 are available.

7. Next Steps

Supplement to the basic requirement book consisting of the elaboration of the ergonomical requirements.

8. Relations with Other Projects

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9. References

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143.1.03		8
TITRE PROCEDURES DE VIEILLISSEMENT ACCELERE POUR TESTS DE QUALIFICATION DE MATERIAUX POLYMERES UTILISES DANS LES SYSTEMES DE SECURITE		Pays FRANCE Organisme Directeur CEA-DgCS
TITLE (Anglais) Accelerated aging nuclear qualification tests of polymeric materials.		Organisme exécuteur DSA/SCAPR (Saclay) Responsable
Date de démarrage 1.4.1977	Etat actuel en cours	Scientifiques
Date d'achèvement 31.12.1980	Dernière mise à jour 31.12.1978	

1. OBJECTIF GENERAL

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

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

2. OBJECTIFS PARTICULIERS

Le premier objectif est de déterminer les conditions limites d'accélération des essais lors des essais de simulation en marche normale et en cas d'accident de référence d'un réacteur PWR.

Afin de déterminer ces limites, l'influence des principaux paramètres

- dose et débit de dose (4 débits ont été retenus : 2.10^3 - 10^4 - 10^5 et 5.10^5 $rad.h^{-1}$)
- température
- ambiance

a été analysée sur une quinzaine de matériaux.

Les polymères sont choisis en fonction de leur caractère représentatif et de leur intérêt technologique pour la construction des matériels utilisés dans les centrales.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

- Laboratoire de chimie macromoléculaire de physico-chimie et d'essais des plastiques
- sources de cobalt 60 de 20.000 et 200.000 Ci.
- accélérateurs d'électrons de 0,5 MeV (3 Kwatt)et de 300 KeV (30 Kwatt)

4. ETAT DE L'ETUDE

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

- un comité de coordination
- 5 groupes de travail par famille de produits : moteurs et isolants - joints et pièces mécaniques - matériels de raccordement et connexions - matériels d'automatismes industriels - câbles et peintures.

Le programme d'essais a démarré en janvier 1978. Certains essais réalisés aux plus forts débits de doses sont terminés et ont mis en évidence l'influence du débit de dose sur les performances des polymères.

5. PROCHAINES ETAPES

- finir les essais en cours, analyser et interpréter les résultats obtenus
- irradier de nouveaux matériaux
- entreprendre l'étude du vieillissement thermique.

6. RELATION AVEC D'AUTRES ETUDES

Cette action entre dans le cadre plus général de la fiche d'action : qualification des appareillages de mesure et des matériels utilisés pour la sûreté des réacteurs dans des conditions de fonctionnement normales suivies en fin de vie d'un ADR

7. DOCUMENTS DE REFERENCE

- J. LAIZIER et R. STOURM

  Choix des conditions d'essais de vieillissement accéléré sous irradiation
  Rapport CAPRI n° 001-23-12-75.

- J. BERTHET - G. GAUSSENS - J. LAIZIER - F. LEMAIRE

  La qualification nucléaire des plastiques
  Rapport CAPRI n° 048-7-9-77

- Détermination des conditions d'essais en vue de l'établissement d'un code pour la qualification nucléaire des matériaux incorporant des polymères.

  G. GAUSSENS - F. LEMAIRE
  Rapport CAPRI - N042 11/8/77
  Rapport CAPRI - N050 3/10/77
  Rapport CAPRI - N003 18/1/78

  J. BERTHET - G. GAUSSENS

Rapport CAPRI N° 015 23/5/78
Rapport CAPRI N° 025 9/8/78.

8. <u>DEGRE DE DISPONIBILITE DES DOCUMENTS</u>

Confidentiel CEA.

143.1.04		8
TITRE ETUDE DE L'EVOLUTION DES CARACTERISTIQUES ELECTRIQUES DES MATERIELS DE CLASSE 1E APRES VIEILLISSEMENT ET ADR.		**Pays** FRANCE **Organisme Directeur** CEA/DgCS
TITLE (Anglais) ELECTRICAL CHARACTERISTICS EVOLUTION OF CLASS 1E EQUIPMENTS AFTER AGING AND ACCIDENTS SIMULATION.		**Organisme exécuteur** CEA-SES/SACLAY **Responsable**
Date de démarrage 1.01.1977	**Etat actuel** EN COURS	**Scientifiques**
Date d'achèvement 31.12.1981	**Dernière mise à jour** 30.6.1978	

1. OBJECTIF GENERAL

Cette action rentre dans le cadre général de l'action de qualification des équipements utilisés pour le contrôle-commande des réacteurs dans les conditions de fonctionnement normal suivies en fin de vie d'un ADR.

2. OBJECTIFS PARTICULIERS

Ces essais portent sur des câbles de différentes natures, des connecteurs, des contacteurs et des transmetteurs de pression.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME :

CAPRI - Irradiations γ.

4. ETAT DE L'ETUDE

Des mesures ont déjà été effectuées et se poursuivent sur des câbles ayant été soumis au vieillissement accéléré et aux contraintes thermodynamiques et d'irradiation d'un ADR. Il s'agissait de câbles de puissance, de mesure et de contrôle comportant des isolants de nature différentes : PVC, polyéthylène, ... et dont les propriétés électriques furent plus ou moins conservées.

D'autres câbles à isolant minéral essayés dans ces conditions présentèrent des dégradations importantes sous l'effet de contraintes thermodynamiques, dégradations dues aux mauvaises liaisons câbles-connecteurs. Le remplacement de câbles à la de BRENNILIS permit d'effectuer des mesures sur les câbles retirés câbles ayant subi des irradiations allant jusqu'à 10^{10} rad.

Par ailleurs, une mission d'information a été effectuée auprès de différents laboratoires et organismes américains : Franklin Institute - Westinghouse NRC - Sandia Laboratories en vue de confronter nos idées et nos méthodes en ce domaine.

5. PROCHAINES ETAPES

- Poursuite des campagnes d'essais sur les câbles comportant de nouveaux isolants :

- Etude du comportement d'autres composants : connecteurs, contacteurs, capteurs.

- Participation à l'élaboration de profils de qualification et de normes concernant les matériels de sûreté.

- Etude du système d'acquisition des mesures de température et de pression du caisson d'essai vieillissement et ADR du petit matériel (CAPRI)

- Analyse des résultats observés sur les câbles prélevés sur la Centrale de BRENNILIS.

6. RELATIONS AVEC D'AUTRES ETUDES :

Cette étude entre dans le cadre de l'étude générale des problèmes de qualification des matériels de classe 1E

143-1-05		8
TITRE DOSIMETRIE APPLIQUEE AUX ESSAIS DE QUALIFICATION DES MATERIELS DE CLASSE 1E UTILISES DANS LES REACTEURS NUCLEAIRES.		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) DOSIMETRY APPLIED TO QUALIFICATION TESTS ON CLASS 1E DEVICES USED IN NUCLEAR REACTORS.		Organisme exécuteur DCh/DRIS/LCRI (SACLAY) Responsable
Date de démarrage 01/01/1975	Etat actuel En cours	Scientifiques
Date d'achèvement 31/12/82	Dernière mise à jour 30/06/78	

1. OBJECTIF GENERAL

La qualification des matériels de classe 1E destinés au contrôle commande et à la sûreté des installations nucleaires exige l'exposition de ce matériel à des irradiations correspondants d'une part :

à

- L'irradiation permanente de ce matériel en ambiance normale et d'autre part :

à

- L'irradiation accidentelle et post-accidentelle en cas d'ADR en fin de vie du réacteur.

2. OBJECTIF PARTICULIER.

Cette étude coordonnée par le DSN, l'EdF et FRAMATOME fait l'objet d'une coopération entre plusieurs unités du CEA dont le LCRI - CERI concerné par la métrologie des rayonnements utilisés au cours de ces opérations d'irradiation.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME.

CAPRI - TRITON - EL.4...

.../...

4. ETAT DE L'ETUDE.

Dans le cadre de l'élaboration des normes, le LCRI-CERI a :

- apporté son assistance à différents groupes de travail dont la synthèse a abouti à la rédaction d'un document provisoire soumis à l'approbation du DSN [1]

- recherché etdéfinir [2], en accord avec la LMRI, une grandeur réellement significative pour la métrologie des essais sous rayonnement

- participé à la réception d'une mission américaine dans le but d'une harmonisation des procédures de qualification.

En ce qui concerne la dosimétrie des rayonnements le LCRI-CERI a poursuivi l'étude de moyens de mesure de doses absorbées photonique élevées.

Pour réaliser ce programme, il était nécessaire de disposer d'un champ de rayonnement γ étalonné et d'intensité élevée. Le champ de l'irradiateur "Pagure" du SARR-SCAPR a été étalonné ; cette opération a fait l'objet d'un procés verbal [3].

L'utilisation de ce champ de référence en absorbée a permis d'effectuer l'étalonnage du film dosimétrique "TAC" et d'étudier l'influence de divers paramètres tels que :

. Le débit de dose de 5.10^{4} à 5.10^{5} rad.h^{-1}

. la dose cumulée jusqu'à 20 Megarads

. la température

. le conditionnement

La détermination des incertitudes de mesure a montré qu'elles sont compatibles avec la dosimétrie des essais de qualification envisagés.

Ces résultats ont ... firmés à l'occasion des mesures effectuées dans un puits de ... sion de EL.4. Cette opération, demandée par le DSN dans le ... l'étude de l'effet des rayonnements sur les cables d'alimentation, a consisté à établir le diagramme donnant la répartition de la dose le long du circuit [4].

5. PROCHAINES ETAPES.

Le programme envisagé se poursuit par :

- l'exploration des propriétés du dosimètre pour des doses élevées jusqu'à 100 Megarads

- la continuation de l'étude des grandeurs d'influence

- la recherche et la mise au point d'une méthode complémentaire, par ionométrie, permettant des mesures instantanées et affranchissant de l'influence de l'énergie. .../...

143-1-07
153-1-03

8

TITRE DETECTION DES DEFAILLANCES DANS LES ENSEMBLES A BASE DE MICROPROCESSEURS UTILISES DANS LES SYSTEMES DE PROTECTION.	Pays FRANCE Organisme Directeur CEA/DGCS
TITLE (Anglais) FAILURES DETECTION IN MICROPROCESSORS DEVICES USED IN PROTECTIVE SYSTEMS .	Organisme exécuteur CEA/SES (Saclay) Responsable

Date de démarrage 1979	Etat actuel Etude nouvelle	Scientifiques
Date d'achèvement Fin 1980	Dernière mise à jour	

1. OBJECTIF GENERAL

L'objectif principal de cette action est de contribuer à l'élaboration de recommandations pour l'utilisation des microprocesseurs dans les systèmes de protection des chaudières nucléaires : les procédures de qualification des logiciels faisantl'objet d'autres actions, cette action sera orientée vers la mise au point de méthode pour détecter les défaillances des ensembles utilisant des microprocesseurs, pendant leur fonctionnement.

2. OBJECTIF PARTICULIER.

On s'interessera, en particulier, aux problèmes de communication entre microprocesseurs appartemant à des ensembles redondants de façon à établir des recommandations pour la conception et l'utilisation de ces matériels.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES.

Il sera réalisé en 1979 un banc d'essai comprenant deux unités d'échange, une unité émettrice et une unité réceptrice permettant la transmission sur fibres optiques de messages à cadence élevée ($\leqslant$ 125.000 bauds). On utilisera ce banc d'essai pour essayer différentes méthodes de détection des défaillances (défaillances matériel, erreur sur la transmission des informations...) Les résultats des essais devraient permettre d'établir des comparaisons et d'évaluer l'efficacité des différentes méthodes de façon à fournir une aide aux équipes chargées de l'analyse de la sûreté des systèmes utilisant des microprocesseurs.

.../...

Les microprocesseurs mis en oeuvre seront deux équipements Motorola 6800 déjà approvisionné par les SES.

Les unités émettrices réceptrices seront des équipements HARRIS UART-HED 6402.

4. ETAT DE L'ETUDE.

On prévoit en juillet 1979 le montage en laboratoire d'un banc d'essai permettant de réaliser une liaison multiplexée entre deux microprocesseurs.

L'étude sur le banc d'essai des méthodes de détection des défaillances doit aboutir en fin 1979.

L'établissement de recommandations sur ce type de liaisons multiplexées et de dialogue entre Microprocesseurs devrait aboutir avant la fin de 1980.

5. PROCHAINES ETAPES.

6. RELATIONS AVEC D'AUTRES ETUDES.

Cette étude est la suite logique de l'étude 143-1-07 et 153-1-03 intitulée "Critères de Sûreté pour l'utilisation de systèmes séquentiels dans les systèmes de protection".

143-1-08		8
TITRE DEVELOPPEMENT D'UNE ARCHITECTURE MULTIPROCESSEURS DE HAUTE FIABILITE POUR UN SYSTEME DE PROTECTION		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) DEVELOPMENT OF A HIGHLY RELIABLE MULTIPROCESSORS STRUCTURE FOR A PROTECTIVE SYSTEMS		Organisme exécuteur LETI-MCTE (Grenoble) Responsable
Date de démarrage 1979	Etat actuel Etude nouvelle	Scientifiques
Date d'achèvement Fin 1980	Dernière mise à jour 1.1979	

1. OBJECTIF GENERAL

Cette action fait suite à l'évaluation d'un système de protection basé sur une structure redondante à 3 processeurs.

- Les phases principales suivantes sont prevues en (1979 et 1980)

  . Etude sur la fiabilité de l'architecture de base à partir de données de simulation.
  . Extension de la structure à un système à vote Majoritaire 2/4
  . Etude et réalisation des modules d'acquisition analogiques et digitals.
  . Etude d'un cas concret.
  . Réalisation et mise en service d'un prototype sur une installation opérationnelle.
  . Validation des solutions retenues. Enseignements tirés de l'exploitation en vue de l'établissement de recommandations pour la mise en oeuvre et l'utilisation de calculateurs dans le système de protection.

Parrallèlement sera menée une étude sur la fiabilité du logiciel et sur diverses méthodologies de conception, d'acriture et de test du logiciel. L'étude en endurance de la fiabilité du système devrait faire l'objet d'une étude ultérieure.

.../...

2. OBJECTIFS PARTICULIERS.

Etude sur la fiabilité de l'architecture de base à partir de données de simulations. Evaluation de la disponibilité et de la sûreté de fonctionnement sur divers types de pannes matérielles et logicielles provoquées.

Extension de la structure à un système de vote majoritaire 2/4

Etude et réalisation des sous ensembles d'acquisition analogiques et digitale décentralisable à haute fiabilité et disponibilité.

Projet détaillé de l'implantation du prototype sur une installation réelle.

Methodologie de conception d'écriture et de test de logiciel pour systèmes à haute fiabilité et disponibilité.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES.

L'installation comporte 2 calculateurs Solar 16-05 de 8000 mots de mémoire rapide supervisés par un calculateur plus puissant Solar 16-40 de 32000 mots de mémoire assisté d'un disque et d'une unité de bande magnétique.

Le programme de gestion actuellement réalisé permet de mettre en oeuvre les échanges entre ces trois calculateurs tant en considérant le système le plus performant comme appartenant au système logique 2/3.

4. ETAT DE L'ETUDE.

L'état de l'ancienne étude concerne le système à vote majoritaire 2/3. L'extension au système à vote majoritaire 2/4 sera l'objet de cette nouvelle étude qui nécessitera également la mise en oeuvre du système d'acquisition des données.

5. PROCHAINES ETAPES.

6. RELATION AVEC D'AUTRES ETUDES.

Les études antérieures sur l'utilisation de calculateurs dans les systèmes de protection : réf. 143-1-08/411-4-01 ont permis :

- de définir à partir d'une architecture 2/3 un mode de fonctionnement d'un ensemble multiprocesseur destiné à assurer la fonction de protection, ce système pourra fonctionner en 2/4.

- d'élaborer au niveau du logiciel, à partir d'outils de programmation standard, une méthodologie possible de conception, écriture et test des programmes, avec pour aboutissement la définition de règles et de recommandations.

... / ...

- de réaliser le système informatique assurant le fonctionnement, l'exploitation du système à partir de données simulées.

  Cet ensemble de programmes comprend :

- un programme de simulations de données d'acquisition avec possibilités de génération d'incidents de fonctionnement au niveau des processeurs.

- Le logiciel d'exploitation du système de protection travaillant à partir des données d'acquisition simulées et procédant à l'enregistrement de la décision.

- Un programme de dépouillement des résultats permettant à partir des données d'entrées et d'éventuels evènements externes et de l'ordre élaboré de valider le bon fonctionnement de l'ensemble du système.

7. DOCUMENTS DE REFERENCE.

- Note technique LETI/MCTE N° 1277 février 1978.
  Proposition d'un système de protection utilisant des calculateurs.
  Application au contrôle des réacteurs nucléaires.

  H.D. MAIER
  P. DARIER

- Note technique LETI/MCTE N° 1278 février 1978.
  Programmation des systèmes de haute fiabilité et disponibilité.

  P.DARIER
  H.D.MAIER

- Note technique LETI-MTCE N° 1316 décembre 1978.
  Système de protection utilisant trois calculateurs
  Logiciel d'exploitation

  D.LALLEMENT

- Communication au congrès de Pittsburg du 20 au 22/7/77
  "Un système de protection utilisant des calculateurs, conception du système et du logiciel".

  P.DARIER
  H.D.MAIER

- Colloque International sur la commande et l'Instrumentation des Centrales Nucléaires. Cannes 24-28 avril 1978
  "Autocontrôle d'un système logique de protection utilisant une structure redondante de calculateurs".

8. DEGRE DE DISPONIBILITE.

Les trois premières notes techniques sont des rapports internes CEA.

143.1.09	8
TITRE EFFETS DES PERTURBATIONS ELECTROMAGNETIQUES D'ORIGINES DIVERSES SUR L'INSTRUMENTATION DE CONTROLE-COMMANDE - MOYENS DE PROTECTION.	**Pays** FRANCE **Organisme Directeur** CEA/DgCS
TITLE (Anglais) EFFECTS OF DIFFERENT KINDS OF ELECTROMAGNETIC PERTURBATIONS ON INSTRUMENTATION AND CONTROL EQUIPMENTS - PREVENTION METHODS	**Organisme exécuteur** CEA/SES (SACLAY) **Responsable**

Date de démarrage	Etat actuel	Scientifiques
1.01.1976	EN COURS	
Date d'achèvement	**Dernière mise à jour**	
31.12.1980	30.6.1978	

1. OBJECTIF GENERAL

Cette action concerne l'étude des effets des perturbations électromagnétiques d'origines diverses sur les appareillages qui entrent dans les systèmes de protection des réacteurs. On recherchera les moyens de lutter contre ces perturbations et l'on définira les contrôles à effectuer pour en vérifier l'efficacité.

2. OBJECTIFS PARTICULIERS :

Dans l'immédiat, trois axes ont été retenus afin de répondre aux problèmes actuels :

- établissement de spécifications techniques pour les liaisons bas niveaux. On définira les caractéristiques des liaisons à utiliser pour les mesures thermodynamiques et neutroniques. Ces spécifications seront établies en fonction des normes en vigueur et des résultats d'expérience acquis sur le comportement des liaisons en milieu perturbé.

- étude sur les effets des champs électromagnétiques de haute intensité. On examinera les incidents dus à la foudre survenus sur différents réacteurs*. Une analyse des mécanismes d'action de ces perturbations sera effectuée afin d'envisager des moyens préventifs. Des expérimentations pourront avoir lieu en collaboration avec les laboratoires spécialisés sur les problèmes de la foudre et un programme de travail sera établi en liaison avec l'IPSN.

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* PHENIX, BRENNILIS, BIBLIS.

- action des perturbations sur les mesures thermodynamiques.
  Les émetteurs-récepteurs portatifs provoquent dans les centrales des perturbations importantes des systèmes de sécurité, en particulier sur les mesures thermodynamiques. On examinera dans quelles conditions ces perturbations risquent de provoquer une paralysie des sécurités et de mettre les systèmes en état de pannes non sures.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Laboratoire d'étude de la foudre de St Privas d'Allier.

4. ETAT DE L'ETUDE

Des travaux importants ont été effectués dans le domaine de la protection des ensembles de mesure neutronique vis-à-vis des perturbations d'origine électrique.

Différentes méthodes d'essais des installations ont été expérimentées sur site et une méthode particulièrement adaptée aux chaines de mesure neutronique a été développée. Cette méthode expérimentée sur les réacteurs de FESSENHEIM a donné des résultats intéressants.

Un document relatif aux recommandations, caractéristiques et méthodes d'essai des liaisons, est actuellement en cours de rédaction dans le cadre du MCH.

5. PROCHAINES ETAPES (calendrier, dates repères, délais de principe)

- Spécifications techniques des liaisons pour les mesures neutroniques,	1er semestre 79
- spécifications techniques pour les mesures thermodynamiques	courant 1980
- recueil d'incidents dus à la foudre,	1er semestre 79
- expérimentation sur la foudre,	1979-1980
- moyens de protection,	1981
- collation des incidents dus aux Walkies-Talkies	1979
- examen technologique de matériels perturbés	1979-1980
- établissement de spécifications techniques	1980

6. RELATIONS AVEC D'AUTRES ETUDES :

Effets de la foudre sur les installations de contrôle-commande

143-1-12 4112-01		8
TITRE ETUDES DE DETECTEURS ET D'ELECTRONIQUES POUR LA SURVEILLANCE DES ELEMENTS COMBUSTIBLES.		Pays FRANCE Organisme Directeur CEA - DGCS
TITLE (Anglais) STUDY OF DETECTORS AND ELECTRONICS FOR FUEL ELEMENTS SURVEY		Organisme exécuteur CEA - SES (Saclay) Responsable
Date de démarrage 1.01.1976	Etat actuel ——	Scientifiques
Date d'achèvement 31.12.1981	Dernière mise à jour	

1. OBJECTIF GENERAL

Cette action dont l'objectif général est l'étude de moyens de surveiller l'état du combustible pendant son irradiation et après son irradiation, a pour objectifs particuliers, d'une part l'étude de détecteurs permettant de contrôler le combustible depuis sa sortie du coeur du réacteur jusqu'à son retraitement, et d'autre part l'observation du rapport des activités des différents produits de fission émetteurs de photons gamma pour vérifier que la détection précoce des défauts de gainage et l'identification de leur nature est possible par ce moyen.

2. OBJECTIF PARTICULIER.

Surveillance des éléments combustibles irradiés.
Il est necessaire de contrôler le combustible depuis la fin de son irradiation jusqu'à son retraitement. Des ruptures de gaine peuvent en effet se produire lors de manutention ou de stockage. D'autre part, il est interessant soit de démontrer, soit de confirmer, le type de défaut défaut d'un élément combustible irradié.

Le contrôle en cours de manutention sert à la protection du personnel, par exemple lors du déchargement d'éléments de la filière à eau pressurisée. L'examen avec caractérisation de défaut permet un tri des éléments combustibles avant traitement, par exemple avant la mise en étui des assemblages de la filière à neutrons rapides.

L'étude doit tenir compte des conditions de cheminement du combustible :

.../...

- pour les réacteurs à eau légère : piscine du réacteur, canal de transfert, piscine de désactivation, château de transport.

- pour les réacteurs à neutrons rapides : bras de transfert, barillet de stockage, couloir à étuis de stockage.

Suivant le temps de désactivation, les éléments radioactifs à détecter et à doser seront :

- des gaz de fission :
  . Xénon 135, énergie gamma 250 KeV, période 9 heures
  . Xénon 133, énergie gamma 81 KeV, période 5 jours

Le rapport Xénon 135, Xénon 133 permettant de caractériser le défaut.

  . Krypton 85, énergie gamma 514 KeV, béta 672 KeV, période 10 ans ;

ce dernier gaz étant pratiquement le seul gaz de fission restant dans des éléments combustibles après 6 mois de refroidissement.

- d'autres produits de fission :
  . iode 131, énergie gamma 364 KeV, période 2 jours.
  . césium 137, énergie gamma 662 KeV, période 30 ans.

- et des transuraniens :
  . neptunium 239, énergie gamma 106 KeV, période 2 jours.
  . américium 241, énergie gamma 60 KeV, période 400 ans.

pouvant être observés lorsque le combustible n'est plus maintenu à l'intérieur de sa gaine.

3. INSTALLATION EXPERIMENTALE ET PROGRAMMES.

Cette étude nouvelle comprendra 3 phases principales :

- une phase de recherche bibliographique suivie de calculs préliminaires pour l'adaptation aux problèmes actuels

- une phase expérimentale en laboratoire avec poursuite de l'étude théorique

- une phase essais sur réacteur à eau légère et à neutrons rapides en fonctionnement

Ces deux dernières phases pourront être menées en collaboration avec les exploitants de réacteurs.

Etude des moyens de détection précoce des défauts de gaine

En régime normal ou transitoire, le relâchement des produits de fission dépend de la nature du défaut . Par exemple, si on n'observe pas d'émetteurs de neutrons différés et que les seuls produits de fission présents dans le caloporteur soient des gaz, on peut en conclure que le défaut est du type fissure et non pas rupture avec contact direct du caloporteur et du combustible. Dans cette partie de l'étude on se propose d'observer le rapport des activités de différents produits de fission, émetteurs de de photons gamma et pour cela on réalisera un appareillage adapté.

.../...

4. ETAT DE L'ETUDE.

Il s'agit d'une étude nouvelle dont les prochaines étapes seront les suivantes :

- Surveillance des éléments combustibles irradiés

  . Point des moyens actuels et évaluation théorique des possibilités de mesure 1979

  . Définition, étude et mise au point d'un appareil de détection. 1980

- Etude des moyens de détection précoce des défauts de gainage

  . Réalisation d'un tiroir délivrant les quotients. 1978

  . Mise au point de l'ensemble et suivi des activités et rapport sur Isabelle. 1979

5. PROCHAINES ETAPES.

6. RELATION AVEC D'AUTRES ETUDES.

Cette étude est la suite logique de l'étude "Investigation et développement des méthodes d'identification et de localisation des défauts de gaine pour les réacteurs à eau".

143-1-14		8
TITRE **MISE AU POINT DE METHODES D'ANALYSE DES DONNEES POUR LA DETECTION D'ANOMALIES DANS LES REACTEURS.**		Pays FRANCE Organisme Directeur CEA/DGCS
TITLE (Anglais) DEVELOPMENT OF DATA ANALYSIS METHODS FOR DEFECTS DETECTION IN REACTORS.		Organisme exécuteur CEA/LETI (Grenoble) MCTE Responsable
Date de démarrage 1/1/1978	Etat actuel nouvelle étude	Scientifiques
Date d'achèvement Fin 1981	Dernière mise à jour Fin 1978	

1. OBJECTIF GENERAL

Etude et mise au point d'une méthode de scrutation et de traitement des mesures en processus complexe (type réacteur nucléaire), dans le but de détecter et localiser le plus tôt possible l'apparition d'un défaut.

Le principe consiste à utiliser la corrélation existante entre les paramétres surveillés pour effectuer leur classement, et définir des zones caractéristiques du bon ou mauvais fonctionnement. La méthode permet de détecter des tendances et donc d'anticiper la détection du défaut.

2. OBJECTIFS PARTICULIERS.

Le modèle sera mis en oeuvre en temps réél sur le réacteur SILOE à partir de l'enregistrement en pile des mesures physiques, débit, température, puissance neutronique, temps de doublement.
Ceci devant aboutir à la réalisation des interfaces permettant d'effectuer la campagne de mesure puis le traitement différé sur le banc traitement du signal.
La méthode d'enregistrement étant au point, elle sera appliquée à des équipements instrumentés spécialement (vannes par exemple)

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES.

Les installations existantes au LETI ont permis en 1978 d'effectuer la mise au point du programme en centre de calcul (calculateur Solar 16-65) la numérisation des paramètres étant effectuée sur un équipement de traitement du signal (Pluminat S). .../...

4. ETAT DE L'ETUDE.

Le programme permet à partir d'un jeu de données d'obtenir :

- La paramètrisation des signaux
- L'introduction de perturbations simulées
- La méthode de classification basée sur l'analyse en composante principale.
- La sortie graphique des résultats.

5. PROCHAINES ETAPES.

6. RELATIONS AVEC D'AUTRES ETUDES.

Le principe developpé ici est en relation directe avec les méthodes de calcul du traitement du signal developpées au LETI depuis de nombreuses années.

143-1-16		8
TITRE CONTROLE DE LA REACTIVITE DEPUIS L'ARRET DU REACTEUR JUSQU'A SA PUISSANCE NOMINALE .		**Pays** FRANCE **Organisme Directeur** CEA/DgCS
TITLE (Anglais) CONTROL OF THE REACTIVITY OF A REACTOR FROM SOURCE LEVEL TO FULL-POWER.		**Organisme exécuteur** CEA/SES/SAI (Saclay) **Responsable**
Date de démarrage 1/1/1979	**Etat actuel** nouvelle étude	**Scientifiques**
Date d'achèvement 31/12/1980	**Dernière mise à jour**	

1. OBJECTIF GENERAL

- Mesure de l'antiréactivité d'un réacteur à l'arrêt : évaluation des différentes méthodes de mesure. Essais des matériels disponibles, dépouillement et interprétation des résultats.
- Mesure de la réactivité dans la période de démarrage. Possibilité de détection d'un mouvement de barre, malgré la contreréaction de l'effet Doppler.
- Possibilité de mesurer la réactivité sur toute l'étendue de contrôle des mesures neutroniques.

2. OBJECTIFS PARTICULIERS.

- Déterminer les avantages et inconvénients réciproques des technologies analogiques et numériques.
- Déterminer les avantages et inconvénients réciproques des mesures de période et de réactivité.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES.

Mise en place d'un détecteur au réacteur ULYSSE.

.../...

- Evaluation et étude comparée des différents appareillages disponibles 1er semestre 1979.

- Essais sur simulateurs analogiques à grande dynamique, sur calculateurs hybrides éventuellement et sur réacteurs : 2ème trimestre 1979.

- Poursuite des essais : 1er semestre 1980.

4. ETAT DE L'ETUDE.

Début en 1979.

5. PROCHAINES ETAPES.

6. RELATIONS AVEC D'AUTRES ETUDES.

Etude sur les réactimètre de sécurité pour réacteurs rapides.

143.1.18/			8
TITRE EFFETS DE SYNERGIE DANS LA REALISATION D'ESSAIS SIMULTANES DE QUALIFICATION. DE MATERIELS DE CLASSE IE			Pays FRANCE Organisme Directeur CEA/SCAPR
TITLE (Anglais) SYNERGIC EFFECTS IN QUALIFICATION TESTS OF CLASS IE EQUIPMENTS			Organisme exécuteur DCA/SCAPR Responsable
Date de démarrage 1.7.78	Etat actuel NOUVELLE		Scientifiques
Date d'achèvement 31.12.80	Dernière mise à jour 12.78		

1. OBJECTIF GENERAL

Les cycles d'essais actuellement définis pour la qualification nucléaire reposent sur des opérations séquentielles, alors qu'au cours du service réel les contraintes se trouvent appliquées simultanément, donc selon un mode combiné.

Actuellement on simule ces contraintes en les appliquant l'une après l'autre, essentiellement pour des raisons de simplicité expérimentale et de coût d'essai. L'étude a pour but de déterminer dans quelle mesure cette procédure donne, ou non, des résultats significatifs ; de relier les résultats constatés à la suite d'un profil d'essai donné (séquentiel ou simultané) aux conditions réelles de travail simulées, donc de définir un ou des profils conduisant à une qualification satisfaisante réalisable dans de bonnes conditions de fiabilité et d'économie.

2. OBJECTIFS PARTICULIERS

- Réalisation, essais et démarrage des enceintes

- **Analyse paramètres d'essai :**

  Comparaison sur polymères modèles et quelques assemblages de séquences d'essai simulatives de l'ADR.

  Contraintes : température, pression, irradiation et mécanique.

  Mesures : propriétés physiques, électriques, déterminations physico-chimiques.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Laboratoire de chimie macromoléculaire, de physico-chimie et de contrôle des plastiques.

Sources de cobalt 60 de 20.000 et 200.000 Ci
Accélérateurs d'électrons de 0,5 à 3 MeV, puissance 3 Kwatt, et de 300 KeV puissance 30 Kwatt.

4. ETAT DE L'ETUDE

Les cahiers des charges définissant les paramètres essentiels des appareils ont été établis, les appels d'offre lancés, les déterminations des coûts de revient établis. Les premiers essais sont prévus dans le deuxième semestre de 1979.

5. PROCHAINES ETAPES

- Réalisation des différentes séquences d'essais (simultanées ou séquentielles) sur des polymères.

- Dépouillement, analyse des résultats et conclusions.

6. RELATION AVEC D'AUTRES ETUDES :

Cette action entre dans le cadre plus général de la fiche action :

"Qualification des appareillages de mesure et des matériels utilisés pour la sûreté des réacteurs, dans des conditions de fonctionnement normales suivies en fin de vie d'un ADR"

<table>
<tr><td colspan="2">143.1.20</td><td>8</td></tr>
<tr><td colspan="2">TITRE
EFFET DE LA FOUDRE SUR LES INSTALLATIONS DE CONTROLE-COMMANDE.</td><td>Pays
FRANCE

Organisme Directeur
CEA Dg CS</td></tr>
<tr><td colspan="2">TITLE (Anglais)
LIGHTNING EFFECT ON CONTROL-COMMAND EQUIPMENT.</td><td>Organisme exécuteur
CEN-G/LASP

Responsable</td></tr>
<tr><td>Date de démarrage
1.1.1979</td><td>Etat actuel
NOUVELLE</td><td rowspan="2">Scientifiques</td></tr>
<tr><td>Date d'achèvement
31.12.1980</td><td>Dernière mise à jour
-</td></tr>
</table>

1. OBJECTIF GENERAL

Le matériel de contrôle-commande concernant la sûreté des installations est sensible aux effets électromagnétiques liés à certains phénomènes physiques tels que la foudre. Il s'agit donc de mieux connaître ces effets et d'en rechercher des remèdes.

2. OBJECTIFS PARTICULIERS

Les caractéristiques des coups de foudre varient notablement d'un éclair à l'autre. Les valeurs limites relevées sont :

amplitude maximale : $5,10^{4}$ A
durée de front : $3,10^{-7}$ seconde.
surtension sur câble à 50 m de distance : $4,5.10^{4}$ V

Le champ électromagnétique lié à la foudre peut donc avoir une influence énorme sur les chaînes participant à la sûreté d'une installation nucléaire L'objet de cette action est de mettre à la disposition des expérimentateurs sous la forme de coups de foudre déclenchés par des tirs de fusées à partir du sol, des rayonnements électromagnétiques dont les effets sur des équipements peuvent être mis en évidence afin d'en réduire les conséquences.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Station de Recherche sur la foudre de St Privat d'Allier.

4. ETAT DE L'ETUDE

La station fonctionne depuis 6 ans en liaison avec l'EdF - le CNES - le SEP de Saclay et le CESTA et a délivré 80 éclairs.

5. PROCHAINES ETAPES :

- Etudes bibliographiques concernant les incidents dus à la foudre sur les centrales nucléaires (BIBLIS)

- apprécier la qualité des normes UTE dans ce domaine

- exposer au rayonnement électromagnétique de la foudre différents ensembles : capteurs et composants (circuits intégrés, microprocesseurs..)

- déterminer les moyens de protection de ces ensembles contre les effets parasites de la foudre.

6. RELATION AVEC D'AUTRES ETUDES

7. DOCUMENTS DE REFERENCE :

Développement de la recherche sur la foudre en France : communication à la 14e conférence internationale pour la protection contre la foudre - GDANSK 22-26 mai 1978 - par le groupe de recherche de St Privat d'Allier.

	CLASSIFICATION: 8
TITLE (ORIGINAL LANGUAGE): Diagnostica e sicurezza tramite analisi di rumore	COUNTRY: Italy
	SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Diagnostics and safety via noise analysis	ORGANISATION: CNEN
	PROJECT LEADER: A. Serra
INITIATED: 1/1/1976	COMPLETED:	SCIENTISTS:
STATUS: In progress	LAST UPDATING: July 1979	

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 and self-checking instrumentation for nuclear particle detection have been developed and operated.

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 is going to be realized.

6) Relation to Other Projects

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

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Diagnostics and safety via noise analysis	8

7) Degree of Availability

Documents are not classified material.

(A.Serra, CNEN, RIT, CSN Casaccia, CP 2400, I-00100 Roma).

	CLASSIFICATION: 8
TITLE (ORIGINAL LANGUAGE): Misura di spettri di neutroni mediante rivelatori fissili	COUNTRY: ITALY SPONSOR: ENEL
TITLE (ENGLISH LANGUAGE): Neutron spectra evaluation by fissile detectors	ORGANISATION: Politecnico di Milano (*) PROJECT LEADER: G.Sandrelli - ENEL V.Sangiust - Politecnico
INITIATED: 1973 / COMPLETED: 1980 STATUS: In progress / LAST UPDATING: July 1979	SCIENTISTS: P.Barbucci - ENEL A.Cesana - Politecnico G.Sandrelli - ENEL V.Sangiust - Politecnico M.Terrani - Politecnico

(*) Istituto di Ingegneria Nucleare - CESNEF

1. General aim

To set-up a quick experimental procedure for the evaluation of fast neutron flux density spectral shapes by counting the delayed neutrons emitted by small fissile monitors.

2. Particular objectives

2.1 To select a suitable set of fissile isotopes taking into account their availability on the market at high degree of purity, the quality of the information about their nuclear properties and their sensitivity in a fast neutron standard field.

2.2 To reevaluate nuclear parameters when the accuracy by which they are known was inadequate to our purposes.

2.3 To develop a version of common unfolding codes adapted for minicomputers, with the view of employing it in on-line analyses.

2.4 To check the whole procedure by performing measurements of neutron fluxes of different shapes.

3. Experimental facilities

All the experimental facilities are at the Nuclear Engineering Institute of the Polytechnique of Milan. They consist of:

- a neutron filter of sintered B_4C inserted in a beam channel near the core of the L54 water boiler reactor (50 kW), inside which a neutron standard field is realized;
- a set of fissile monitors to be used for setting up the new experimental procedure;
- a pneumatic system for tranferring the fissile monitors from the irradiation to the counting position;
- a BF_3 detector embedded in paraffin, coupled to a 100 channels analyzer, for neutron counting;
- a 60 cm^3 Ge-Li detector, coupled to a 4096 channels analyzer, for some supporting analyses by γ-ray counting;
- a LABEN 701 minicomputer, 12k words, for the treatment of the experimental results.

45.897

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TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Neutron spectra evaluation by fissile detectors	8

4. Project status and essential results

The points 2.1 - 2.2 - 2.3 have been completed. A suitable set of fissile targets turned out to be that one formed by ^{232}Th, $^{233/235/238}U$, ^{237}Np, $^{238/239/240/241}Pu$ and ^{241}Am.

As far as nuclear data are concerned fission cross sections were derived from ENDF/BIV file, while the status of the delayed neutron data for ^{237}Np, $^{238/240/241}Pu$ and ^{241}Am required a new evaluation of the decay constants and delayed neutron yields. These latter measurements were performed by comparing the delayed neutron intensity with the activities of some γ -emitter long-lived selected fission products in the targets.

The code SAND II, at present one of the most commonly used for neutron spectra unfolding, was adapted in a version at 27 energy groups for mini-computers. A comparison between the results obtained in the standard spectrum by this version and those obtained by the original code at 620 groups was fully satisfactory.

The measure time required for the evaluation of a neutron spectrum by this procedure is about 1 day, more than an order of magnitude lower than that required by the traditional application of the Multiple Foil Activation technique.

The experimental data can be unfolded on-line by means of a minicomputers, within the reach of any laboratory, while in the traditional application large computers, which moreover make only off-line analyses possible, are indispensable.

5. Next steps

As a check of this technique the determination of the fast spectrum in a irradiation facility of the TAPIRO reactor operating at CSN-Casaccia (CNEN) is foreseen.

6. Reference documents

Internal progress reports.

7. Degree of availability

To a limited extent, available with the authorization of the sponsoring organization.

ENEL- CRTN, Bastioni Porta Volta 10, I-20121 Milano

Politecnico di Milano, CESNEF, Via Ponzio 34/3, I-20133 Milano

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

General aim

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

Particular objectives

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

Experimental facilities

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

Project status

Objective 1&2: Several experiments per core since December 1974. In-core results since December 1977.
Objective 3 : Off-line separation in operation for noise above 5 Hz.

Next steps

Objective 1: To be continued for present and next cores.
Objective 2: Continuous improvement of understanding, especially of pressure noise.
Objective 3: Extension to lower frequencies and for effects hither to not considered; on-line analysis.
Objective 4: Pattern recognition type of techniques are envisaged for real time application.

Relation to other projects: -

[illegible] CLASSIFICATION : 8.10.4

Reference [illegible]

[1] [illegible] Measurements and Analysis of Ex-core Neutron Detector Noise of [illegible] (PWR) at Full Power, Reaktortagung, Düsseldorf (1976), p. [illegible]77.

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

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

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

[illegible] of availability

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

Budget: -

Personnel: -

		CLASSIFICATION: 4-8
TITLE (ORIGINAL LANGUAGE): SIMULAZIONE E ANALISI DINAMICA DI IMPIANTI NUCLEARI		COUNTRY: ITALY SPONSOR: C.N.E.N.
TITLE (ENGLISH LANGUAGE): SIMULATION AND DYNAMIC ANALYSIS OF NUCLEAR POWER PLANTS		ORGANISATION: C.N.E.N. PROJECT LEADER: M.DI BARTOLOMEO
INITIATED: 1962	COMPLETED:	SCIENTISTS: T.G. BISERNA F. CIAMPA O. MODONESI
STATUS: In progress	LAST UPDATING: June 1979	

9. OTHER SAFEGUARDS

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	9	RS 294

Vorhaben/Project Title	
Auswertung des Vorversuchs V und Abschluß des Förderungsvorhabens RS 104 Evaluation of the Preliminary Investigation V and Completion of R+D-Task RS 104	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 2, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 10. 77	31. 7. 78	Dr. Dorner
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	31. 12. 78	

1. General Aim

Since for concrete reasons task RS 104 will not be continued, a document will be produced giving the current results obtained from the preliminary tests performed to date, and a presentation of the state of knowledge based upon recent experiences. This work will complete task RS 104.

2. Particular Objectives

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

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

3. Research Program

3.1 Burst Process, Pressure-Relief- and Blowdown Process

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

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

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

3.2 Preliminary Investigation Carried Out

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

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

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

4. Test Facilities

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

5. Progress to Date

- Dimensional measurement of the test tube
- Evaluation of the measurement results
- Comparison between measurement results and those of previous preliminary tests
- Preparation of test report draft.

6. Results

Test results show favourable agreement between preliminary tests V and III (retardation of boiling).

7. Next Steps

Project completed.

8. Relation with Other Programs

RS 104, RS 108

9. References

Dr. H. Kopp et al.
Evaluation of the Preliminary Investigation V and Completion of R+D-Task RS 104
Final report BMFT RS 294, Oct. 1978.

10. Degree of Availability

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

General aim

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

Particular objectives

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

Experimental facilities and programme

Computer code REBOR.

Project status

Operational for Dodewaard BWR.

Next steps

Not applicable.

Relation with other projects

Not applicable.

Reference documents

Internal KEMA reports.

Degree of availability

Free on basis of exchange with other programmes.

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

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

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

10. CORE AND PRIMARY CIRCUIT IN STEADY STATE CONDITIONS

	CLASSIFICATION: 10-14
TITLE (ORIGINAL LANGUAGE): ANALISI STATISTICA DI SEGNALI	COUNTRY: ITALY SPONSOR: C.N.E.N.
TITLE (ENGLISH LANGUAGE): STATISTICAL ANALYSIS OF SIGNALS	ORGANISATION: C.N.E.N. PROJECT LEADER: A.FEDERICO
INITIATED: 1966 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: June 1979	

1. - General aim

Concern the developments of statistical methods for acquisition and elaboration of experimental data coming from nuclear power plants and experimental loops.

2. - Particular objectives

Apply statistical methods to study:
Reactor physics, Thermohydraulic and mechanical effects, Acoustic noise, Fuel coolant interaction etc.

3. - Experimental facilities and programme

Transducers, d.c. amplifiers, filters, magnetic records.
A package of programmes for statistical analysis.

4. - Project status

The main efforts are now devoted to the BWR and LMFBR reactors.

5. - Relation to other projects

These studies are made in collaboration with responsibles of experimental facilities.

6. - Reference documents

1) L.Cimorelli - A.Federico
Applications of spectra analysis techniques 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 signal elaboration developed in CNEN laboratories. IAEA Specialis Meeting on Analysis of Measurements to Diagnose Potential Failures.
Roma, Aprile 10-11, 1972.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
STATISTICAL ANALYSIS OF SIGNALS	10-14

7. - Degree of availability

Know-how and facilities for statistical analysis are available.

Berichtszeitraum/Period 1.1.1978 - 31.12.1978	Klassifikation/Classification 10.1	Kennzeichen/Project Number RS 232
Vorhaben/Project Title In-Core Data Acquisition and Processing with the Aim of Early Fault Detection in Light Water Reactors. In-Kern-Meßdatenerfassung und -verarbeitung mit dem sicherheitstechnischen Ziel der Früherkennung von Schäden in Leichtwasserreaktoren.		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor T.U. Hannover Institut für Kerntechnik
Arbeitsbeginn/Initiated 1.12.1976	Arbeitsende/Completed 31.12.1979	Leiter des Vorhabens/Project Leader Prof. Dr.-Ing.D.Stegemann
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General aim

Development of a complex system for acquisition, processing and presentation of measured in-core data with the safety oriented aim of early fault detection in light water reactors.

2. Particular Objectives

Basing upon the experience in application of statistical analysis methods in power reactors (RS 68A), an on-line surveillance system will be developed, allowing the early detection of certain anomalies in reactor operation and supporting the operators to avoid dangerous operating conditions. The additional informations consist of a number of nuclear and thermohydraulic data. The system design includes any future tasks of combined analysis of neutron noise and vibration measurements.

3. Research Program

3.1 Construction of a highly flexible electronic measuring system, including an analog magnetic tape recorder and a digital magnetic tape recorder for the data acquisition and preparation of selected safety relevant in-core signals of a nuclear power plant.

3.2 On line measuring and processing of these data via fast Fourier analysis. Computation of characteristic functions in the time and frequency domain like correlation functions, spectra, phase and coherence functions. Computation of significant thermohydraulic data like steam bubble velocities, NRMS-values and local power densities. Comparison of these data to reference values, obtained under normal reactor operating conditions.

3.3 Presentation of the processed data on a color video display, plotting of spectra and automatically registering of selected data on a line printer.

3.4 Theoretical considerations concerning the influence of steam bubbles to the detector signal, especially for self powered neutron detectors.

4. Experimental Facilities, Computer Codes

System of 32 amplifiers, multiplexer, analog to digital converter, 14-track analog magnetic tape recorder and a 32 K process computer with I/O-facilities, including a digital tape recorder and a color video display.

5./6. Progress to Date and Results

3.1 Completition of 32 differential, self-compensating, low-noise amplifiers, suitable for the in-core fission chamber instrumentation as well as for the signals of self powered neutron detectors. Test of all amplifiers under full power conditions at the power plant Brunsbüttel. Recording of several in core detector signals on analog magnetic tape with the constructed amplifiers was performed.

Completition of the signal multiplexer which switches computer controlled the input of a 32 channel ADC from the stationary detector signal to the random part of the signal. A tube multiplexer is developed and under construction. This multiplexer switches, also computer controlled, the signals of 12 detectors of 3 measuring tubes selectable out of 8 to the inputs of the analog magnetic tape recorder.

The total system for data acquisition and processing of 32 in-core detectors was put together including all further necessary hardware and successfully tested with signals of an analog magnetic tape simulating a power plant.

3.2 Assembling of all developed computer programs for the automatic data acquisition and processing of 32 signals was successfully performed. The initiation of the system and the delivery of the necessary process data to the system is performed by a consol dialog. The Fast Fourier transformation and the further processing of the data blocks is performed by microprograms. These data are then stored on a magnetic disc for 50 measuring cycles back, presenting a reactor operating history of about five to six hours. The following values are stored (always 50 cycles):

(I) 32 normalized auto spectral densities
(II) 24 normalized cross spectral densities
(III) 32 local power density values
(IV) 32 NRMS-values
(V) 24 local steam bubble velocities

The fiftyfirst cycle is always extinguished for the benefit of the new values. Programs for comparing the measured data to reference values are developed but not yet tested.

3.3 The processed data are mainly presented on a color video display. For the automatic operation of the system one can select an automatic display of either a measuring tube presentation, showing four NRMS-values and three steam bubble velocities of a selected tube for the last 50 cycles. This gives a vertical view of the local situation in a cooling channel. Or one can select a plane presentation which shows a horizontal survey of the local conditions of the core consisting of eight NRMS-values and eight steam bubble velocities.

Furthermore one can present by request the following processed data on the video display:

(I) The auto spectral density of any detector of the last fifty cycles
(II) Every cross spectral density of any pair of detectors which are situated one upon each other, also of every of the last fifty cycles
(III) The corresponding auto-and cross correlation functions
(IV) The coherence functions and the phase functions
(V) The 32 local power densities of the last 50 measuring cycles For comparison one can display several spectra in one presentation

All 32 auto spectral densities are automatically plotted in [illegible] time intervals of a few days for the comparison of the form of the spectra. Furthermore an automatic line printer output of selectable thermohydraulic parameters is done in intervals of some hours.

3.4 The results of the theoretical considerations concerning the detailed influence of steam bubbles to detector signals will be published in the next future.

7. Next steps

Integration of checking programs to the system and tests. Development of criteria for a significant exceeding of set limits and of a table with adequate action levels. Installation of the system at a working nuclear power plant, gaining experience in the behaviour of the system under power plant working conditions and to get experience concerning the on-line surveillance of a power plant. Furthermore it is expected to get more information about the long time behaviour of the cooling channels and the physics of the random signal of the in core instrumentation with the help of detailed off-line analysis of recorded signals from the magnetic tapes.

8. Relation with other projects

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9. References

M. Zeller	Verarbeitung von Rauschsignalen an der Rechenanlage HP 2100 mit Hilfe von Mikroprogrammen, Übersicht und Programmbeschreibungen IKH-Bericht 90/77
O.P. Singh, D. Stegemann	Theoretical Treatment of Noise Analysis in Boiling Water Reactors Atomkernenergie 31, 1978
P. Gebureck, O.P. Singh, D. Stegemann	Experimental and Theoretical Noise Analysis Investigations in Boiling Water Reactors Reactor Noise (SMORN II), Progress in Nuclear Energy, Vol. 1, 1977

1.1.1978 - 31.12.1978 RS 232

P. Gebureck, G. Grondey
D. Stegemann, M. Zeller

In Core Data Aquisition and Processing with the Aim of Early Fault Detection in Light Water Reactors
Enlarged Halden Programme Meeting, Loen, 1978

10. Degree of Availability of the Reports

Institut für Kerntechnik, Universität Hannover, Elbestr. 38A, 3000 Hannover 21

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	10.2	RS 234
Vorhaben/Project Title Verifikation von Transientenprogrammen (DWR und SWR) Verification of Transient Programs (PWR and BWR)		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 11, Erlangen
Arbeitsbeginn/Initiated 1. 7. 78	Arbeitsende/Completed 31. 12. 80	Leiter des Vorhabens/Project Leader Dr. Preusche
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 78.	Bewilligte Mittel/Funds

1. General Aim

The accuracy and range of validity of the computing programs for the analysis of accidents without any loss of coolant shall be tested and demonstrated. Commissioning tests and comparison calculations shall be carried out to this end.

2. Particular Objectives

The range of validity of the point and 1-dimensional (1D) models shall be delimitated by comparison with experiments and 3-dimensional benchmark calculations. Any quality defects detected in the programs are to be eliminated by improvements.

Special tests to clarify the behaviour of the test of the reactor plant are neither specified nor run. However, the information of plant behaviour accumulated in the scheduled core behaviour tests will be processed and evaluated.

3. Research Program

3.1 Specification of model tests with thermohydraulic and nuclear disturbances. This should involve especially a study how far disturbances entailing a positive reactivity addition can be performed without any hazard to the plant. Coordination with the overriding commissioning program.

Verification of steady-state conditions **on the basis of** commissioning tests already carried out in GKN and Biblis B (PWR), KKB, KKI and KKP (BWR). (For the BWR, even partial checking of transient, spatial behaviour).

3.2 Specification, compilation and checking of the required instrumentation

3.3 Quality Assurance of Measuring Signals:
Check and documentation of signal formation in the measuring channels of process instrumentation. If necessary, provision of separate facilities for data acquisition.

3.4 Performance of tests under clearly defined and Documented Marginal Conditions
For the BWR, test are to be carried out during the commissioning of the Krümmel (KKK) Power Plant and for the PWR, in Grafenrheinfeld (KKG/BAG) and, if possible, also in plants already handed over.

3.5 Evaluation of Tests

3.6 Re-calculation using the available computing models, tracing of weak points, introduction of improvements, delimitation of range of application of individual programs.

3.7 Specification of complementary tests on the basis of experience so far gained until now.

3.8 Evaluation and re-calculation of complementary tests.

4. Experimental Facilities, Computing Programs

A process-computer-controlled data processing system with 48 channels suitable for these tests is already available.

The programs to be verified are in particular:

IQSBOX (3D transient PWR core simulator)
IQSBWR (3D transient BWR core simulator)
COSBWR (1D transient BWR core simulator)
[illegible] (PWR plant simulator with 1D core)
LOOP 7 (PWR plant dynamics program)

5. Progress to Date

To 3.1 - Drafting of a PWR test program
- preliminary clarification of a BWR test program and data recording in Philippsburg NPP (KKP I)

To 3.3 Tests for the quality assurance of measuring signals have been carried out in the KKU plant with in the scope of these preliminary tests.
The data processing system was installed in the fourth quarter. After the behaviour of single signals had been tested in steady state operation, transient commissioning tests could be recordes at 50 % reactor power, such as shutdown of one reactor coolant pump, load rejection to station service requirements and turbine trip. d

6. Results

To 3.1 Suitable tests for the KKU plant were selected to check the methods and measuring procedures. Data recording was prepared on the plant site and measurements made to ascertain the transfer response of the measuring channels.

To 3.3 Records of the measuring signals from KKU are available on magnetic tape. A brief preliminary evaluation of the eccentric rod drop test confirmed the high resolving power of the measuring system and behaviour on allocation of the control rod drop process to the remaining signals.

7. Next Steps

To 3.3 Recording of further transients during the commissioning of KKU.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

		CLASSIFICATION: 10.2
TITLE (ORIGINAL LANGUAGE): Statisk Reaktorfysik		COUNTRY: **Denmark** SPONSOR: Risø National Laboratory
TITLE (ENGLISH LANGUAGE): Development of calculation methods for static and quasistatic reactor physics in light water reactors.		ORGANISATION: Risø National Laboratory PROJECT LEADER: T. Petersen
INITIATED: 1968	COMPLETED:	SCIENTISTS: C.F. Højerup L.Z. Jensen G.K. Kristiansen H. Neltrup B. Schougaard
STATUS: Progressing	LAST UPDATING:	

1. General aim

Provision of static and quasistatic reactor physics information relevant for safety assessment of light water reactors.

2. Particular objective

Reliable methods for calculating -
Power shape as well within fuel boxes as overall for the reactor as function of burn up.
Reactivity as function of burnup.
Reactivity control and reactivity budgets.
Reactivity coefficients.
Stability.

3. Experimental facilities and programmes

4. Project status

1. Progress to date

a) Multi-group cross section generation from UKNDL completed
b) Multi-group $G \leq 76$, collision probability methods for fuel rods completed and coupled to multi-group diffusion theory calculations completed.

45.897

c) Few group 2D and 3D diffusion theory calculation codes, using finite difference techniques, finite element, flux synthesis and nodal theory for the reactor core completed.
d) 3D flux synthesis or 3D nodal theory coupled to hydraulic channel calculation for BWR including burn up completed.

2. Essential Results

Reactor physics code system assesed in calculation on Yankee Rowe, Connecticut Yankee and Dresden and partially in calculations on a BWR.

5. Next Steps

1. Test the system further against BWR measurements.
2. Further refinement of methods concerning fuel element calculation and development of fast 3D methods.
3. Develop methods for determining power distribution by a combination of measurements and calculations.

6. Relation with other projects

Provision of cross-sections and parameters for dynamics projects and fuel management studies.

7. Reference Documents

1. A.M. Hvidtfeldt Larsen, H. Larsen and T. Petersen. Calculation on a Boiling Water Reactor as a Test of the Risø Reactor Code Complex. Risø Report No. 268, 1972.

2. Torben Petersen. Aspects of Prediction of the Performance of a Boiling Water Reactor. Risø Report No. 289, 1973.

3. H. Neltrup and Per B. Suhr. Survey Calculation on the Hadam Neck (Connecticut Yankee) Power Plant as a Test of the Risø Reactor Physics Code System. Risø Report No. 298, 1973.

4. C.F. Højerup, The Cluster Burn up Programme CCC and Comparison of its Results with NPD Experiments, Risø-M-1898 (1976) 6 pp.

8. Degree of availability

Partly available, partly available on exchange basis.

	CLASSIFICATION: 10.2
TITLE (ORIGINAL LANGUAGE): Maintenance and development of LWRWIMS (RPK 1.1)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Winfrith)
	PROJECT LEADER: J R Askew (Winfrith)
INITIATED: March 1978 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: March 1979	

Background

LWRWIMS is the UK standard design and best estimate code for lattice calculations on LWR. It produces cell averaged data for input to JOSHUA.

Objectives

To maintain and develop LWRWIMS and its nuclear data library as required.

Programme

1. General maintenance.

2. Validation by comparison with experimental data.

3. Development of new models.

4. Revision of library data (especially U^{235} to improve calculation of moderator temperature coefficient).

Facilities

A working ICL 4/70 version has been frozen. All development is now on the Harwell IBM 370.

Reference Documents

AEEW - R 785 LWRWIMS. A Modular Computer Code for the Evaluation of Light Water Reactor Lattices.

Part 1.	Description of Methods	F J Fayers et al
Part 2.	Users Guide	M J Halsall
Part 3.	Programmers Guide	M J Halsall

The current users' guide is an on-line computer file that can be printed at any time by any LWRWIMS user.

	CLASSIFICATION: 10.2
TITLE (ORIGINAL LANGUAGE): Maintenance and development of JOSHUA (RPK 2.1)	COUNTRY: UNITED KINGDOM SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Winfrith) PROJECT LEADER: J R Askew (Winfrith)
INITIATED: March 1978 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: March 1979	

Background

JOSHUA is a 3D neutronics/thermal hydraulics depletion code used for fuel management studies on all types of water reactors.

Objectives

To maintain and develop JOSHUA and extend it to include xenon transients that could arise during normal operation.

Programme

1. General maintenance.

2. Validation by comparison with operating data from existing reactors.

3. Extension to include time dependent xenon and samarium build up (as in TRALEB) primarily for xenon stability studies.

Facilities

Development for the PWR programme will be on the Harwell IBM 370.

Reference Documents

AEEW - R 1165 A review of the equations solved by the Computer programme JOSHUA. M J Roth.

Berichtszeitraum/Period 1.01.-31.12.1978	Klassifikation/Classification 10.3	Kennzeichen/Project Number RS 81
Vorhaben/Project Title Mischungseffekte bei parallel durchströmten Kanälen Mixing effects in parallel channels with water two-phase flow		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Euratom C.C.R. Ispra / Italy
Arbeitsbeginn/Initiated 01.01.1975	Arbeitsende/Completed 31.12.1979	Leiter des Vorhabens/Project Leader Dr.Herkenrath/Dr.Hufschmidt
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds DM 1.303.590.-

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 to providing experimental data on the local distribution of mass-flow, enthalpy and voidfraction so as to achieve a better thermohydraulic description and more precise DNB margin for a LWR core. Furthermore, studies in steady-state conditions are a necessary preliminary to studies of the mixing process in transient conditions such as those occurring during power transients or blowdown. Therefore, the first phase of the programme will be devoted to investigations under steady state conditions.

The actual programme provides the studies of steady-state measurements of the subchannel interactions in two-phase flow conditions with 16 rod cluster test-sections in BWR-geometry (pressure 70 bars), and in PWR-geometry (pressure 160 bars) with appropriate splitting devices at the outlet and the studies of transient mixing measurements for partial depressurization (to about 80% of the initial pressure) with the above mentioned testsections. For not damaging the loop components - especially the pumps - a complete blowdown of the loops is not possible.

For a more fundamental understanding of the mixing process investigations with two-channel test-sections of different shapes (firstly an eight-shaped channel with different heat-inputs) are foreseen also in both steady-state and transient conditions.

2. PARTICULAR OBJECTIVES

The mixing programme was started in the frame of collaboration contracts between CNEN (Rome) and BMFT (Bonn) on one hand and EURATOM on the other. The contracts involved the studies of the boiling mixing phenomena using two different test sections:

- A two-channel test section for fundamental studies (pressure up to 150 bars; high pressure water loop PRIL) and
- a 16-rod cluster test section with BWR-geometry (pressure 70 bars; high pressure water loop BOWAL).

Later, the BOWAL studies should be extended to a pressure of 160 bars using PWR geometries.

After first operational tests with the two-channel test section in the PRIL loop at the beginning of 1974, this part of the programme was interrupted and effort was concentrated on the preparation and execution of the 16-rod bundle tests in BOWAL.

3. RESEARCH PROGRAMME

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

- Steady state measurements of the mixing effect with a 16-rod cluster in BWR geometry (70 bars)
- Mixing studies in transient conditions with 16-rod clusters in BWR- (70 bars) and PWR geometries (160 bars).

4. EXPERIMENTAL FACILITIES

The high-pressure water loop BOWAL (3.6 MW power input) to be used for the boiling mixing experiments with the 16-rod cluster test-sections in BWR- and PWR-geometry and the high-pressure water loop PRIL (0.6 MW power input) to be used for the fundamental mixing studies and for calibration tests with void-meters have been described in detail in / 1 / and the BOWAL loop is schematically shown in Fig. 1. The main data of the two loops are the following:

PRIL LOOP

This loop consists of:

- controllable [illegible] circulation
- partial blowdown (20%) facility for the transient tests

- pressurizer, condenser, subcooler and preheater (360 kW).

The characteristics of PRIL are:

- maximum power: 600 kW
- maximum pressure: 200 bar
- maximum temperature: 365°C
- special instrumentation: different void meters.

BOWAL LOOP

This loop consists of:

- controllable forced circulation
- partial blowdown (20%) facility for the transient tests
- pressurizer, condenser, subcooler and preheater (360 kW).

The characteristics of BOWAL are:

- maximum power: 3,6 MW
- maximum pressure: 250 bar
- maximum temperature: 365°C
- special instrumentation: distribution of temperature and power along a number of the rod, pressure distribution along the test section. Outlet enthalpy distribution among the 6 basic subchannels of the cluster.

The 16-rod cluster test section PELCO-S for the BWR investigations (70 bars) is described in / 2 /. The PWR test section EUROP (16rods, 160 bars) is actually in the final state of construction.
The X-ray voidmeter devices to be used for the transient measurements has been developed and will be described in a special report after testing under real operation conditions.

5. PROGRESS TO DATE

In 1978 the steady state BWR mixing experiments with PELCO-S have been completed and the final report including the comparison with computer codes is in preparation.

6. RESULTS

After the modification of the BOWAL loop (Fig. 1) / 2 / the mixing tests with the BWR-test section PELCO-S (70 bars) were started with simultaneous sampling and enthalpy analysation (Fig. 2). After different calibration tests with special respect to the isokinetic measurement technique, the comparison of the summed outlet mass-flows $\left[Q_{m_{tot}} = 8.Q_{m_1} + 4.Q_{m_2} + 4\ Q_{m_3} + 1.Q_{m_4} + 8\ Q_{m_5}\right.$, the numbers are referred to the sampled subchannels]

with the total inlet mass-flow (measured with the diaphragms between pump and preheater - see Fig. 1), showed deficits to the order of 6%. After renewed inspections of the upper part of the 16-rod cluster test-section this mass-balance in adiabatic conditions could be improved to the order of $\pm$ 1% by eliminating subchannel 3 which showed great deviations with respect to the isokinetic Δp-measurement. The mass balance has now been established by

$$Q_{m_{tot}} = 12\ Q_{m_1} + 4\ Q_{m_2} + Q_{m_4} + 8\ Q_{m_5}$$

In analog manner the heat balance over the test section has been made by

$$Q_{h_{tot}} = 12\ Q_{m_1} h_{o_1} + 4\ Q_{m_2} h_o + Q_{m_4} h_{o_4} + 8\ Q_{m_5} h_{o_5} - Q_{m_{tot}} h_i$$

where h_i is the inlet enthalpy and h_o are the subchannel outlet enthalpies.
The mean outlet enthalpy can be calculated by:

$$h_{o_{tot}} = h_i + \frac{Q_{h_{tot}}}{Q_{m_{tot}}}$$

In the reference period the steady state mixing tests with the BWR test section PELCO-S have been completed in the following test range:
- pressure: about 70 bars
- mass flow density: 1000, 1500, 2000 kg/m^{2}s
- inlet quality: - 12% to -2%
- heat flux density: up to 60 W/cm^2 (2,1 MW)
- outlet quality: up to 30%

In total about 200 test series have been run under various combinations of the above listed test parameters. A detailed final report of these tests with tables and diagrams, including the isokinetic calibration tests, with the respective analysis of the results is in preparation / 3 /. From this report three representative diagrams

(Fig. 3 a-c) have been chosen showing the outlet quality of the sampled subchannel versus the mean outlet quality of the test section PELCO-S. Each point of these plottings is an arithmetic mean value of up to 15 test series under nearly equal conditions. Whilst the outlet qualities of the corner subchannel 2 are always considerable below the mean values (dotted line = 45° diagonale) the qualities of the other sampled subchannels are near the mean value. Because of the smaller weightness of subchannel 2, however, the heat balance in any case is guaranteed with acceptable tolerances.

PWR-16-rod cluster EUROP

After successful pretests for the sealing of the heater rods under real operation conditions the PWR-16-rod cluster test section EUROP (cross section see Fig. 2 - PWR) has been mounted in loop BOWAL. Before delivery, different tests of the completed test-section under pressure and temperature have been performed at CISE, Milan to study the behaviour of the assembly.

X-ray voidmeters

The X-ray voidmeters composed of three high voltage generators KRISTALLOFLEX 800 and six tubes AG-W 61 T (60 kV; 3000 W) for the five sampling pipes and the main outlet tube have been tested in detail and it has been found that the ripple of the high voltage caused by the rectifying has to be diminished (the furnisher has garanteed a drastic reduction in ripple). The detector device for the voidmeters have been developed by the Lehrstuhl für elektronische Schaltungen of the Ruhruniversität/Bochum (FRG) and the now existing measurement chain with ionisation chambers is able to detect void-fraction variations from zero to 100% in a response time of better than 5 ms with the necessary precision. An intensive test-program for the calibration of the whole voidmeter device on an optical bank and under real test conditions (70 and 160 bar) have been started.

7. NEXT STEP

- Continuation of the mixing experiments with test section EUROP (PWR-geometry, 160 bars) under steady state conditions in the following range:

pressure: 160 bars
mass flow density: 2000 ÷ 4000 kg/m^2s
inlet temperature: about 285°C
heat flux: up to 3.6 MW

- Comparison of the experimental results with existing mixing codes in collaboration with NUCLITAL, Genova
- Calibration of the X-ray voidmeters under real operation conditions (up to 160 bars) in PRIL loop
- Mixing tests under transient conditions with EUROP.

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

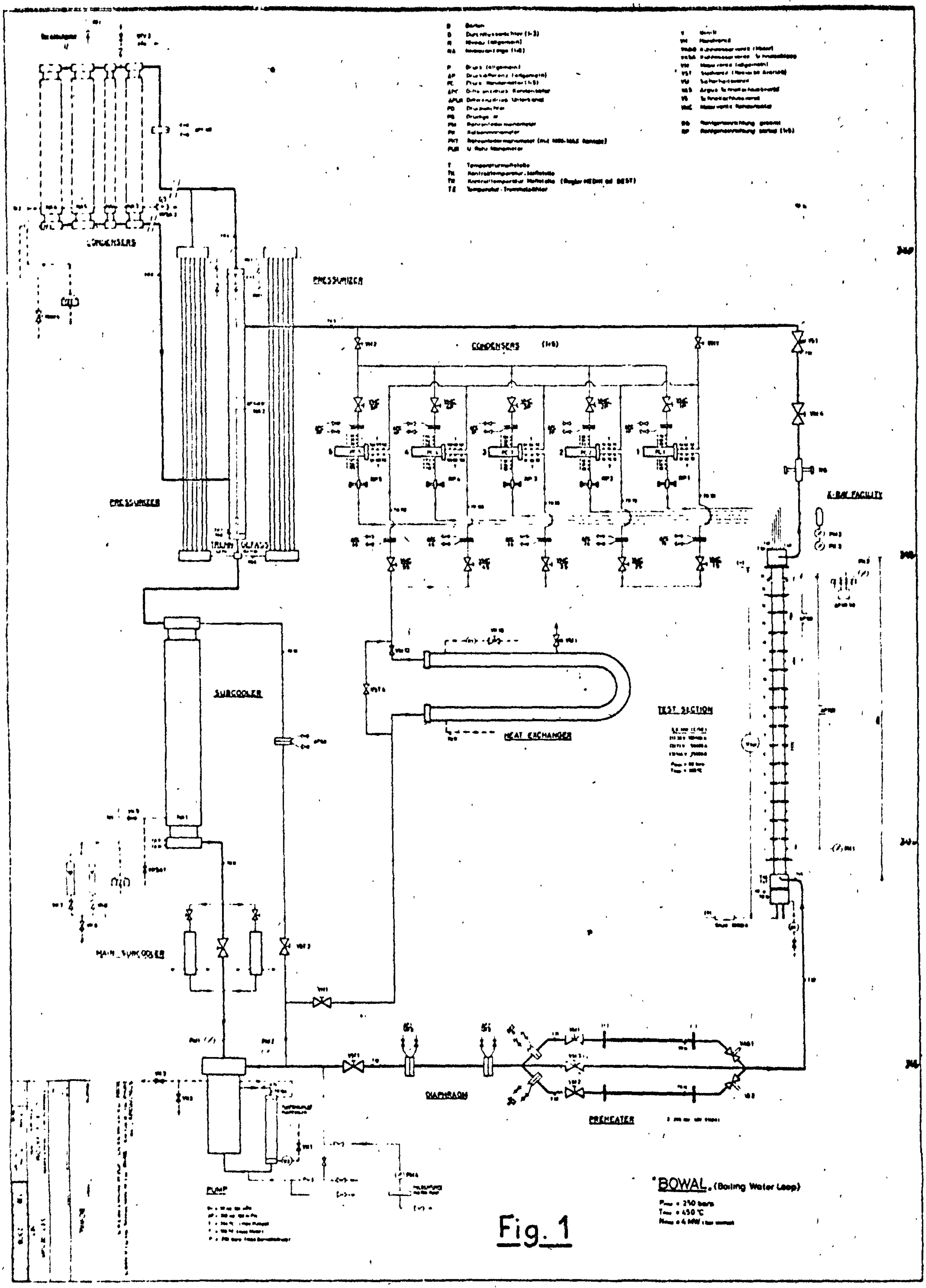
/ 1 / GASPARI, G.P., GERMANI, G.F., LUCCHINI, F., MARELLI, A. "PELCO-S : A BWR 16-Rod Test Section for Subchannel Experimental Analysis". CISE-Doc. Service, PELCO no.4, May 1975

/ 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 the JRC Ispra" EUR 6045 en (1978)

/ 3 / HERKENRATH, H., HUFSCHMIDT, W. "Experimental Investigation of the Enthalpy and Mass Flow Distribution between Subchannels in a BWR Cluster Geometry (PELCO-S)" EUR-Report (in preparation)

10. DEGREE OF AVAILABILITY OF THE REPORTS

Reports mentioned in 9 are not classified and, except ref. 1, are on sale at the office for Official Publications of the European Communities, 37, rue Glesener, Luxembourg.



BOWAL (Boiling Water Loop)

Fig. 1

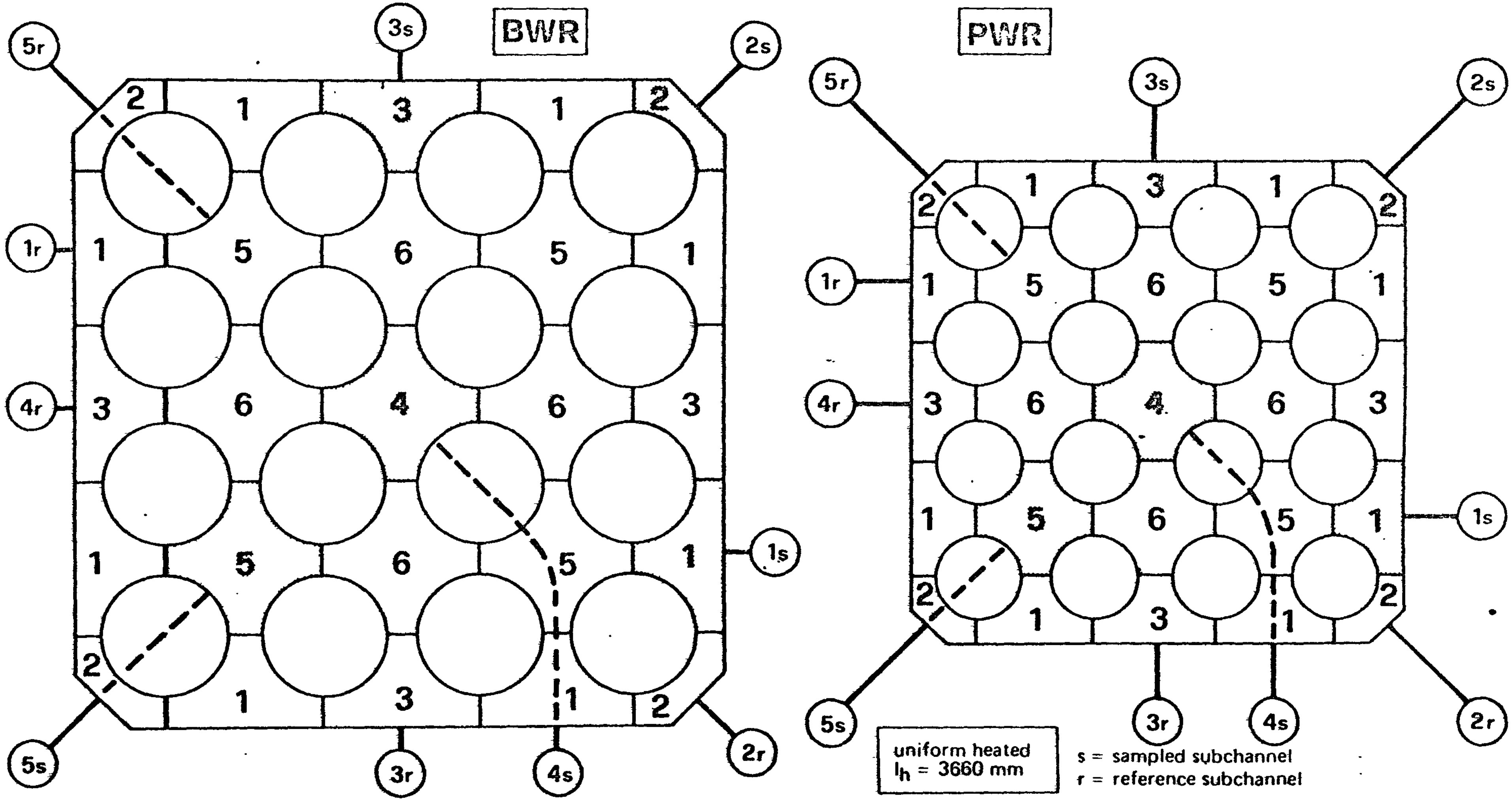


Fig. 2 – Cross-Section of the Test-Section Outlet with Splitters Between the Subchannels and Appropriate Taps for Static Pressures (Isokinetic Measurement)

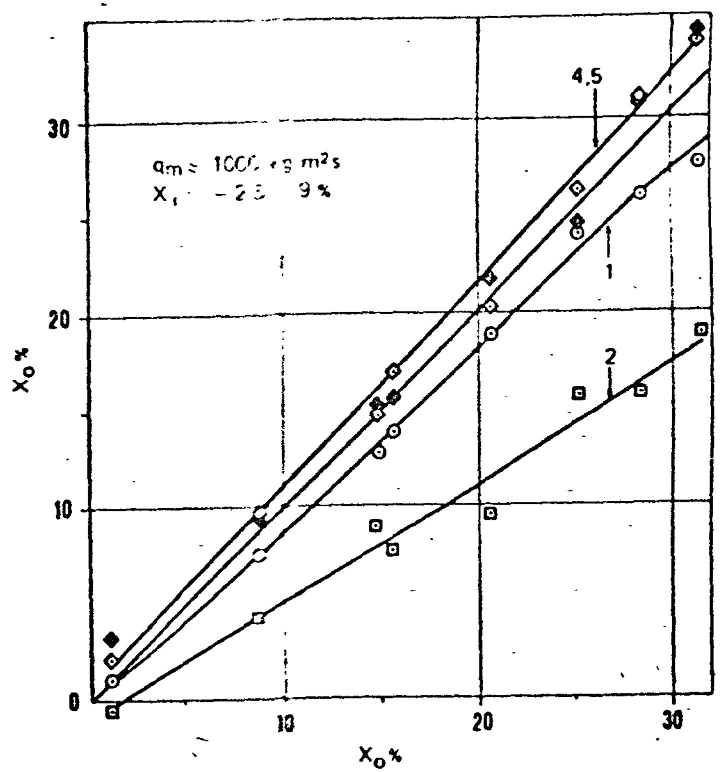
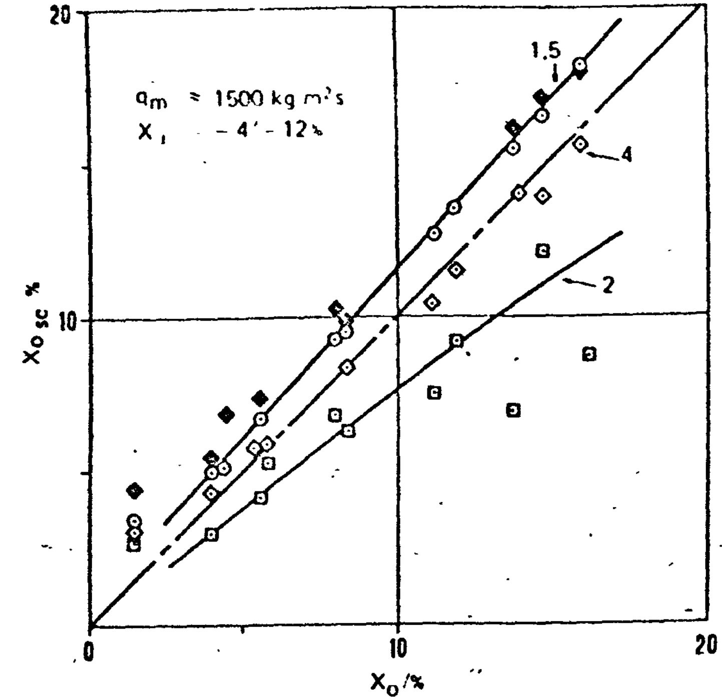
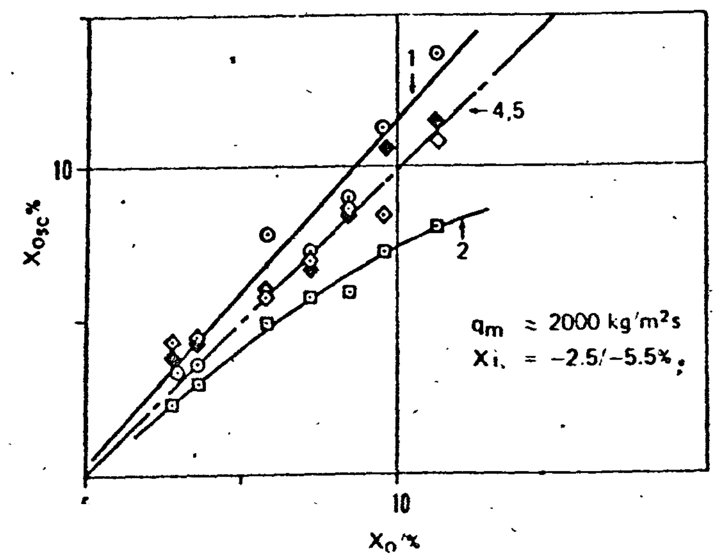


Fig. 3a-c-Outlet Quality of Subchannels Vs Mean Outlet Quality of
Test-Section PELCO-S at About 70 Bars
Subchannel 1 ⊙
Subchannel 2 ⊡
Subchannel 4 ◇
Subchannel 5 ◆





	Classification: 10.3
Title: I	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: II: SDS, a thermohydraulic subchannel programme for steady-state analysis of water reactors	Organization: Risø National Laboratory
Initiated date: May, 1970 Completed date: Autumn, 1974 Status: Code made, Verification finished Last updating: Autumn, 1974	Scientists: A. Olsen, F. Cortzen, V.S. Pejtersen

1. General aim

To provide a general subchannel code, for the analysis of thermal-hydraulic, steady-state conditions in water reactors.

2. Particular objectives

The objective is to make a calculation model (code) which can predict, based on sound physical models, local void, mass flow, enthalpy and burn-out margin for safety evaluation and design.

3. Experimental facilities and programme

The experimental programme has been executed concurrently with the code development to supply data on local phenomena in fuel elements with respect to thermo-hydraulics.

Experiments were made on a 7-rod and a 9-rod BWR geometry, as well as on (concentric and excentric) annular geometries.

4. Project status

1. Progress to date

A FORTRAN 4 code exists, capable of solving, on a subchannel basis, steady-state thermal-hydraulic problems, for BWR's and PWR's. Specifically predicting void- and mass-flow-distributions on a fuel-box or core basis. (No limit on number of subchannels, so far up to 400 have been tested).

2. Essential results

On the modelling side a substantial improvement has been made with respect to the formulation of the subchannel equations, physically, mathematically and codewise. This in turn has made it possible to simulate, on a sound basis, hitherto not approachable experiments, such as flow-blockages.

5. Next step

Physical models and correlations must be verified in respect to new and further experimental data.

6. Relation with other projects

The basic subchannel project was a joint venture of three Scandinavian organizations: Danish AEC, AB Atomenergi, Sweden, and Institutt for Atomenergi, Norway. The nordic co-operative project was completed 1973, and the Danish project was an extension of the work. The experience gained with this project has been utilized for the work with the transient subchannel code, TINA, developed within the framework of the NORHAV project.

7. Reference documents

Z. Rouhani: Review of Momentum balance in Subchannel Geometry Eur. Two-phase flow meeting (ETPFM), Brussels 1973.

J. Bosio, O. Imset: Two-phase flow investigations in a 7-rod bundle. ANS European Meeting, Karlsruhe, Oct. 1973.

8. Degree of availability

Not available.

	CLASSIFICATION: 10.3
TITLE (ORIGINAL LANGUAGE): Esperienze di frazione di vuoto in sezioni di prova 8x8 tipo BWR/6	**COUNTRY:** ITALY **SPONSOR:** NUCLITAL
TITLE (ENGLISH LANGUAGE): Void fraction experiments in 8x8 BWR/6 test sections	**ORGANISATION:** NUCLITAL/CISE **PROJECT LEADER:** F. LUCCHINI (NUC)
INITIATED: January 1977 — **COMPLETED:**	**SCIENTISTS:** G.P. GASPARI (NUC) R. RAVETTA (CISE)
STATUS: In progress — **LAST UPDATING:** June 1979	

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 tecnique.
   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 at end 1979/beginning 1980. The analysis completion is scheduled in December 1980.

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.

F. Lucchini, Nuclital, V.G. D'Annunzio 113, I-16121 Genova

	CLASSIFICATION: 10.3
TITLE (ORIGINAL LANGUAGE): Esperienze di dryout con sezioni di prova 8x8 tipo BWR/6	COUNTRY: ITALY SPONSOR: NUCLITAL
TITLE (ENGLISH LANGUAGE): Dryout (CHF) experiments in 8x8 BWR/6 test sections	ORGANISATION: NUCLITAL/CISE PROJECT LEADER: G.P. GASPARI (NUC)
INITIATED: January 1977	COMPLETED:
STATUS: In progress	LAST UPDATING: June 1979
SCIENTISTS: F. LUCCHINI (NUC), R. RAVETTA (CISE)	

1. General Aim: verification and development of design 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^{2}s.
   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 1980 first half. The analysis completion is scheduled in December 1980.
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.

G.P. Gaspari, Nuclital, V.G. D'Annunzio, 113, I-16121, Genova

		CLASSIFICATION: 10.3
TITLE (ORIGINAL LANGUAGE): Esperienze di dryout con sezioni di prova anulari		COUNTRY: Italy SPONSOR: NUCLITAL
TITLE (ENGLISH LANGUAGE): Dryout (CHF) experiments in annular geometry		ORGANISATION: NUCLITAL/CISE PROJECT LEADER: L. Perticaro (NUC)
INITIATED: April 1978	COMPLETED: April 1979	SCIENTISTS: M. Capelli (CISE) I. Ceresa (CISE)
STATUS: Completed	LAST UPDATING: June 1979	

General Aim

Evaluation of the effect of the axial heat flux profile on the dryout power.

Particular Objectives

Obtaining CHF experimental data in annular geometry using 12 heated rods with different axial heat flux profiles.

Experimental Facilities and Programme

The experiments have been carried out on the CISE IETI-1 loop at 70 bar, specific mass flowrate 250-2000 kg/m^2s, power up to 300 kW.

Project Status

The whole experimental program is completed, as well as the analysis of data.

Next Steps

None

Relation with Other Projects

These experiments follow and complete a previous dryout programme carried out by a 4x4 rod bundle.

Reference Documents

G.P. Gaspari, F. Lucchini, L. Perticaro, "Dryout tests in annular geometry with different axial heat flux profiles", NUCLITAL RP 78/019.

Degree of Availability

Proprietary.

L. Perticaro, Nuclital, V. G. D'Annunzio 113, I-16121 Genova.

45.897

	CLASSIFICATION: 10.3
TITLE (ORIGINAL LANGUAGE): Comportamento termoidraulico stazionario e transitorio dei canali di potenza del reattore CIRENE nella fase di avviamento.	COUNTRY: ITALY SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Steady-state and transient thermal-hydraulic behaviour of CIRENE power channels during reactor start-up	ORGANISATION: CISE PROJECT LEADER: R. Granzini, R. Ravetta
INITIATED: June 1975 — COMPLETED: 1979 STATUS: in progress — LAST UPDATING: June 1979	SCIENTISTS: G. Masini, C. Medich, O. Vescovi

1. General aim: to study the thermoydraulic behaviour of boiling power channels at positive inlet quality in the low flowrate range both in steady state and transient conditions.

2. Particular objectives: to obtain design and procedure information on reactor start-up and experimental data to validate TILT simulation code.

3. Experimental facilities and programme: CIRCE loop simulating in a closed circuit one or two full-scale power channels.

   Programme: 1) Steady-state tests with positive inlet quality to determine two-phase flow patterns and to measure critical power
   2) Transient tests of steam flowrate reduction to zero with one or two power channels in parallel.

45.897

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Steady-state and transient thermal-hydraulic behaviour of CIRENE power channels during reactor start-up.	10.3

4. Project status: steady-state and transient tests with one power channel were completed and analyzed; transient tests with two power channels in parallel are completely set up.

5. Next steps: The experiments will be carried out in the second half of 1979.

6. Relation with other projects: parallel-channel instability tests in the start-up conditions of CIRENE reactor.

7. Reference documents: classified memoranda.

8. Degree of availability: to a limited extent.

(CISE, C.P.3986, I-20100 Milano)

		CLASSIFICATION: 10.3
TITLE (ORIGINAL LANGUAGE): Termoidraulica dei separatori di vapore		COUNTRY: Italy SPONSOR: AMN-CNEN
TITLE (ENGLISH LANGUAGE): Steam separators thermohydraulics		ORGANISATION: AMN PROJECT LEADER: F. Zanzi
INITIATED: 1978	COMPLETED:	SCIENTISTS: E. Giammari, F. Zanzi (AMN); G. Guglielmini, C. Isetti, G. Milano, G. Reale (University of Genova)
STATUS: In progress	LAST UPDATING: September 1979	

1) General Aim

To obtain a full comprehension of phenomena related to steam separators and dryers in a BWR plant. In particular aspects connected with safety requirements are considered.

2) Particular Objectives

The work is going on following these principal lines:

a) Interactions of the components with the system (the requirements on carryover and carryunder are studied as affected by safety requirements due to the limits on contamination in the turbine and on void fraction at the inlet plenum).
b) Analysis of the steam separator and its components.
c) Mechanical design of the steam separator.

Item "b" is being developed in collaboration with a research university centre.

3) Reference Documents

G. Guglielmini, C. Isetti, G. Milano, G. Reale, "Analisi dei separatori di vapore per BWR. Aspetti fenomenologici". AMN, IV78-01B, 2/11/78.

A. Benazzoli, A. Isoppo, "I separatori di vapore nei BWR tipo General Electric". AMN, NT-DCA-7802.

A. Verri, "Introduzione al modello matematico di un separatore di vapore". CNEN, NT-78(25)VBA-1/G71A-17.

E. Giammari, F. Zanzi, "Interazioni dei separatori di vapore con il sistema", AMN, IV79-08.

P. Cerretti, "Analisi di fattibilità in Italia dei separatori di vapore per BWR-6", [illegible]

45.897

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Steam separators thermohydraulics	10.3

4) Degree of Availability

Restricted distribution on some of the documents above.

Project leader:
F. Zanzi, AMN-VPR/RTI, Via G. D'Annunzio 113, I-16121 Genova.

5) Additional Information

Stydies are performed in the frame of CNEN-AMN research agreement.

	CLASSIFICATION: LWR 10.3
TITLE (ORIGINAL LANGUAGE): Mischungseffekte bei parallel durchströmten Kanälen	COUNTRY: C.E.C.
	SPONSOR: BMFT/C.E.C.
TITLE (ENGLISH LANGUAGE): Mixing effects in parallel channels with water two-phase flow	ORGANISATION: JRC - Ispra
	PROJECT LEADER: H. Herkenrath
INITIATED: 1975	COMPLETED: 1979
STATUS: Continuing	LAST UPDATING: April 1979
	SCIENTISTS: W. Hufschmidt

1. General aim

Basic studies of thermohydraulic mixing between the subchannels of a boiling fuel cluster.

2. Particular objectives

Experimental investigation of the mixing phenomena in two-phase flow conditions (steady-state and transient) with water at 70 bars (BWR) and 160 bars (PWR) in 16 rod cluster geometries (BOWAL loop). Validation of cluster thermohydraulics code. 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 and preheater (360 kW)

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
- pressurizer, condensor, subcooler and preheater (360 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 rods, pressure distribution along the test section. Outlet enthalpy distribution among the 6 basic subchannels of the cluster.

3.3. Test sections

Two particular 16 rod test sections simulating typical BWR-geometry - PELCO-S - and typical PWR- geometry - EUROP-, active length 3.66 m, with subchannel fluid extraction equipment (iso-kinetic probe) at the heated length top end.

4. Project status

4.1. Progress to date

Steady state measurements of mixing effects with the 16 rod cluster in BWR geometry (70 bars) are completed. Analysis and comparison with thermohydraulic codes are nearly terminated.
Steady state measurements with the 16 rod cluster in PWR geometry (160 bars) have been started.
The development of X-ray voidmeters is continued.

4.2. Essential results

The BWR results will be published in the near future.

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5. Next steps

Steady state measurements of mixing effects with a 16-rod cluster in PWR geometry (160 bars).

Mixing studies in transient conditions with 16-rod cluster in PWR geometry.

Development of X-ray voidmeters.

7. Reference documents

[1] HERKENRATH, H., HUFSCHMIDT, W.
The Pressurized and Boiling Water Loops BOWAL and PRIL for Boiling Mixing Studies of the Heat Transfer and Fluid Mechanics Division of the JRC Ispra
EUR 6045 EN (1978)

[2] GASPARI, G.P., GERMANI, G.F., LUCCHINI, F., MARELLI, A.
PELCO-S: A BWR 16-Rod Test Section for Subchannel Experimental Analysis
CISE-Doc. Service, PELCO No. 4, May 1975

[3] REACTOR SAFETY PPR No. 3524 for Period January-June 1978, pp 45-49

[4] HERKENRATH, H., HUFSCHMIDT, W.
Experimental Investigation of the Enthalpy and Mass Flow Distribution between Subchannels in a BWR Cluster Geometry (PELCO-S)
EUR-Report (to be published)

[5] HERKENRATH, H., HUFSCHMIDT, W., LUCCHINI, F.
Experimental Subchannel Investigation in a 16-Rod Test Section by Means of the Isokinetic Sampling Technique
Paper to be presented at 2nd Multi-Phase Flow and Heat Transfer Symposium-Workshop, Miami Beach, USA, 16-18.4.79

ENERGIEONDERZOEK CENTRUM NEDERLAND (ECN)		CLASSIFICATION: 10.3
TITLE: Burnout onderzoek aan 9-staafs bundel		COUNTRY: THE NETHERLANDS
		SPONSOR: Ministry of Social Affairs ORGANIZATION: ECN
TITLE (ENGLISH LANGUAGE): Burnout experiments on 9-rod bundles		PROJECTLEADER: S.B. van der Molen
INITIATED : 1972	LAST UPDATING : May 1978	SCIENTISTS: H. Hoogland D.W. Middleton D. Werner
STATUS: nearly finished	COMPLETED : End 1978	

General aim

Study of the burnout phenomena in bundles

Particular objectives

Experimental study of the influence of the position of the supporting grids on the value of the burnout-flux and the place where the burnout occurs.

Experimental facilities and program

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

Project status

The burnout flux of the central rod of a 9-rod bundle is determined at a pressure of 70 bar as a function of a massflow rate, in the range of 700 - 1800 kg/m^2 sec, the heatflux from the outer rods and as function of the position of (58, 126 and 240 mm) of the supporting grids with respect to the outlet side of the bundle. The project concerns experiments performed in a bundle with an axial uniform heat flux distribution, whereas the radial heat flux distribution can be varied.

Next steps

Calculations with Cobra III c and comparison with experimental results. Completion of the study through a report, scheduled end of 1978.

Relation to other projects: no

Reference documents

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

Degree of availability:

Through ECN library channel

Budget: -

Personnel: -

ENERGIEONDERZOEK CENTRUM NEDERLAND (ECN)		CLASSIFICATION: 10.3
TITLE: Experimenteel onderzoek naar mixing en de invloed van rooster op mixing in een 36-staafs bundel		COUNTRY: THE NETHERLANDS SPONSOR: ECN ORGANIZATION: ECN
TITLE (ENGLISH LANGUAGE): Mixing experiments in 36-rods bundle and the influence of supporting grids on mixing		PROJECTLEADER: S.B. van der Molen
INITIATED : 1973	LAST UPDATING : May 1978	SCIENTISTS: H. Hoogland A. Warmenhoven
STATUS : finished	COMPLETED : July 1978	

General aim

Experimental investigation of the mixing phenomena in a bundle geometry.

Particular objectives

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

Experimental facilities and program

Testloop for low-pressure experiments.

Project status

A 6x6 bundle is provided with a length of 6 heating rods in special positions. At this moment at two axial positions in the bundle the temperature of the flowing liquid is measured by a number of thermocouples over the cross-section of the bundle in order to investigate the heat exchange between the subchannels in case of the nonhomogeneous heatflux distribution due to the heat production by only 6 rods of the total of 36. Measurements have been carried out with and without a supporting grid positioned between the two temperature measuring planes.

Next steps

The obtained data show that for low mass flows mixing does not increase when a supporting grid is installed. A report of the measurements-results is planned in July 1977.

Relation to other projects: No.

Reference documents: No.

Degree of availability: Through ECN library.

Budget: -

Personnel: -

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

	CLASSIFICATION : 10.3
TITLE : Ontwikkeling rekenprogramma VITESSE Berekening van stationaire snelheid en temperatuurverdelingen in een bundelgeometrie voor onsamendrukbare fluïda.	COUNTRY: THE NETHERLANDS SPONSOR : ECN ORGANIZATION : ECN
TITLE (ENGLISH LANGUAGE): Development computer code VITESSE Computation of steady state velocity and temperature distribution in bundle geomtry for incompressible fluids	PROJECTLEADER : Slagter, W.
INITIATED : Febr. 1975 — LAST UPDATING : July 1979 STATUS : progressing — COMPLETED : 1981	SCIENTISTS : Roodbergen, H.A. Dekker, N.H.

General aim

The development of a computer code based on a local approach for the prediction of velocity and temperature distributions in a fuel rod bundle cooled by an incompressible fluid.

Particular objectives

The finite element method is being applied to describe the distributed parameter approach for subassembly thermal-hydraulics.

Experimental facilities and program

Hydraulic experiments are carried out with the Laser Doppler Anemometry (LDA). Thermal-hydraulic experiments have been planned.

Project status

The code allows for the prediction of 2D-fully developed turbulent velocity and 3D temperature distributions in fuel rod subassembly of bare smooth rods with single phase cooling. Two models of turbulence have been incorporated in the code. Code validation is carried out with respect to turbulent flow predictions.

Next steps

Recalculation of thermal-hydraulic experiments to verify the thermal-hydraulic part of the code. Predictions of developing turbulent flows in a parallel plate channel to simulate hydraulic behaviour of gridded bundles.

Relation with other projects

- LDA mesurements of local transport phenomena in support of "local approach" calculations.
  Coehoorn, J. et al; ECN memo No: 79-1
- Feasibility study for the experimental task "Measurements of temperature fields in enlarged rod bundle geometries with sodium flow"
  Möller, R. et al; private communication.

Reference documents

- Slagter, W., Roodbergen, H.A.
  TRIP: A finite element computer program for the solution of convective heat transfer problems.
  RCN-243; Petten, January 1976.
- Slagter, W.
  Finite element analysis for turbulent flows of incompressible fluids in fuel rod [illegible] Nucl. Sc. and Eng. 70 ([illegible]) [illegible].

- Slagter, W., Roodbergen, H.A. and Dekker, N.H.
  Prediction of fully developed turbulent flow in noncircular channels by the finite element method
  Turbulent forced convection in channels and rod bundles. Proc. of NATO Advanced Study Institute, Istanbul (1978), Washington DC, Hemisphere publishing Corporation, 1979.

Degree of availability

Upon mutual agreement at ECN-Petten

Budget: -

Personnel: -

	CLASSIFICATION: 10.3
TITLE (ORIGINAL LANGUAGE): CRITICAL 2-PHASE FLOW MODELLING IN VARIABLE AREA FLOW CHANNELS	COUNTRY: UK
	SPONSOR: SRD
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD UKAEA
	PROJECT LEADER: A R EDWARDS

INITIATED: DECEMBER 1976	COMPLETED: OPEN ENDED	SCIENTISTS: S F HALL
STATUS: STATIC AT PRESENT	LAST UPDATING: MAY 1979	

1. GENERAL AIM

Theoretical description of non-equilibrium 2-phase flow in variable area passages - calculation of steady-state critical flowrates.

2. PARTICULAR OBJECTIVES

To gain understanding of physics involved in non-equilibruim 2-phase flow, to provide boundary conditions for large blowdown codes.

3. EXPERIMENTAL FACILITIES

None.

4. PROJECT STATUS

   1. Progress to Date

      Study of literature (costing). Develo ment of 1-D, 1 Fluid, non-equilibruim homogeneous model and computer program, CRIBBLE.

   2. Essential Results

   1-D 1 Fluid model not enough to tell us anything new about physics.

5. NEXT STEPS

Development of 2-D, 2 Fluid model, with better bubble dynamics and equilibruim assumptions.

6. RELATION TO OTHER PROJECTS AND CODES

Some of the work similar to development of HUBBLE-BUBBLE, CRITS programs.

7. REFERENCE DOCUMENTS

SRD R97, SRD R127

8. DEGREE OF AVAILABILITY

No restrictions

PROJECT TITLE: Thermal Performance of LWR Fuel Cluster	CSNI INDEX NUMBER 10.3
SPONSORING COUNTRY: United Kingdom	ORGANISATION Atomic Energy Establishment, UKAEA, Winfrith
DATE INITIATED:) Continuing Programme) from 1966 DATE COMPLETED:)	PROJECT LEADER: D H Lee

DESCRIPTION

1. General Aim

To determine the thermal and hydraulic performance of fuel rod clusters under steady state and transient conditions in order to improve both performance and safety of water cooled reactors.

2. Particular Objectives

To provide high quality experimental data needed for the improvement of computer codes used for reactor assessment. In particular to determine:-

(a) Dryout power under steady state and transient conditions

(b) Post-dryout heat transfer coefficients

(c) Rewetting behaviour

3. Experimental Facilities

Large scale high pressure test facilities capable of mounting 5 x 5 full length electrically heated rod clusters with powers up to 9MW, to operate under BWR and PWR conditions. Test section and instrumentation laboratories, high pressure test loops and mini computer data acquisition systems.

4. Project Status

Dryout studies have included 36-pin and 60-pin SGHWR cluster and 5 x 5 PWR clusters.

5. Next Steps

Modifications of the large test facilities are in hand to provide transient flow and eventually blowdown capability.

Reference Documents

A Review of Dryout Data in SGHWR Fuel Bundle Simulations. N A Bailey, D H Lee. AEEW R1001.

The Winfrith 9MW Heat Transfer Rig. J D Obertelli. AEEW M 1386.

Description of the Winfrith 6MW Heat Transfer Rig. C H Robinson. AEEW M 1387.

Classification

10.3

Title 1 DRYOUT IN BOILING CLUSTERS	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION AEE WINFRITH
Title 2	Project Leader J OBERTELLI Scientists:
Initiated 1966 Completed : Status : Last updating	

Description:

1. General Aim

To control the conditions for a boiling-type reactor so that dryout is sufficiently unlikely to occur; this implies a knowledge of the consequences of dryout.

2. Particular Objectives

To observe dryout in an out-of-pile rig under conditions representative of the reactor.

3. Experimental Facilities

The 9 MW electrically heated rig capable of taking a full-sized bundle.

4. Project Status

Dryout has been studied as a function of the bundle geometry, flow, etc.

5. Next Steps

Further measurements to study the effect of power distribution and bundle geometry, and the effect of a continuously-running spray.

Reference Documents

Internal documents.

Obertelli and Owen AEEW-M935 (Unclassified).

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	1o.4	RS 29o

Vorhaben/Project Title	
Geräuschmessungn an Kernreaktoren im Frequenzbereich von ca. 1 kHz bis ca. 1 MHz Noise measuring on the primary circuit of the cooling system in nuclear power stations	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Allianz-Zentrum für Technik GmbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.6.77	31.3.79	Raible
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General aim

The object of this project is the development of a Leak Detection and Monitoring System based on acoustic emission, with the aid of which leakages can be detected and located on pressurized components.

As opposed to the hitherto applied methods of leak detection the significant features of an acoustic emission system are:

- very short response time
- higher sensitivity
- precise leak location capability.

2. Particular objectives

In continuation of the development of a Leak Detection and Monitoring System for the reactor coolant loop based on acoustic emission, the particular object of this project is the measuring and analysing of noises, which are emitted during operation of the PWR and BWR and a test circuit in the frequency range from 1 kHz to 1 MHz. By the installation of various types of sensors, data is obtained on which type of sensor is best suited for the most favourable frequency range.

The so-called "operating noises" depend on the operating conditions, e.g. on the parameters of the medium in the cooling system.

Typical "operating noise patterns" are measured in the above-mentioned frequency range and compared with the "leakage-noises" of simulated leaks.

Furthermore, the propagation of sound in the actual structures of various components is investigated with the aid of an artificial noise source.

When the noise pattern dependent on the various locations in a reactor cooling system and on the working conditions is known, an "atlas of noises" can be compiled, which then represents the basis for an acoustic monitoring system.

3. Research program

The points of main emphasis of this project work are measurements of the operating noise in a wide frequency band during the phase of initial start-up (zero power phase) and during power output. The project embraces the following 5 points:

3.1 extensive operating noise measurements using 1o measuring points on the reactor cooling system of a Pressure Water Reactor (PWR) during the non-nuclear phase of the initial start-up (first hot-check-operation) and at the same time, measurements of the sound propagation, too.

3.2 comparative measurements in a PWR at 4 measuring points on the cooling system during power output (nuclear phase).

3.3 comparative measurements in a BWR at 4 measuring points on the reactor cooling system during power output.

3.4 operating noise measurements at a test plant to compare the noises in a reactor plant and a test plant using 4 measuring points.

3.5 the development of an artificial noise source for sound propagation and attenuation measurements (Battelle-Frankfurt).

In cooperation with the Kraftwerk Union Erlangen (KWU), the Battelle-Institut Frankfurt and the Gesellschaft für Reaktorsicherheit Garching (GRS), the research program was elaborated and will be carried out in joint partnership.

4. Measuring facilities

The measurements are carried out on two types of reactor power plants, the BWR and PWR and on a test plant of the KWU Erlangen.

Measurement facilitation depends on timing and on the moment when a PWR is in the start-up phase or a PWR or a BWR is in the refuelling phase.

5. Progress to date

In 1976, the noise of simulated leakages on a test loop was measured using accelerometers for the low frequency range and ultra-sonic transducers for the high frequency range. The spectrum of the leakage noises and the measurements of the narrow-band RMS values show that the noise level depends on the leak cross-section as well as the leakage rate, in both the low frequency and the high frequency range. From the background noise existing of the time, it was shown that there was good sensibility for leak detection.

On the other hand, before deciding on the optimum frequency range and on the best type of sensor for a leak monitoring system, the actual operating noise in the above-mentioned frequency range, under various working conditions and on various components, must be known.

In 1977 and 1978, all points of the research measuring program were carried out except the point 3.4. The operating noise was measured at the reactor cooling systems in the power plants

- PWR Kernkraftwerk Gösgen (first and second hot-check-operation)
- PWR Kernkraftwerk Biblis A (during nuclear phase and power

output)

- BWR Kernkraftwerk Isar (during first nuclear start-up and power output).

The propagation of sound in the actual structures of various components was investigated with the aid of an artificial noise source on the main steam line of the PWR Gösgen.

6. Essential results

The first survey shows that the highest level of the operating noise was measured on the primary circuit of the PWR at the recirculating loop inlet. The operating noise in the BWR is generally lower than the one in the PWR.

The results of the propagation-measurements shows a decrease of the noise level within a short-range field (up to about 4 m). Within a long-distance field (between 4 and 2o m), however, an attennation is not measurable. This is valid within the frequency range from 15 kHz $<f<$ 3o kHz. Additional evaluations shown an attennation from about o,5 dB/m in the long distance-field from 1o to 2o m (band-pass 88 kHz - 96 kHz).

7. Next steps

A lot of datas have been recorded and have to be evaluated for the final report. These are the next steps to do. Results about that are to be discussed together with Battelle, GRS and KWU.

8. Relation with other projects

RS 97 A, Allianz-Zentrum für Technik
"Solid-Borne Sound Measuring on Reactor Pressure Vessels and on the Primary Circuit of the Cooling System.
1st part, Pre-Experiments for a Leak Monitoring System."

RS 193, Battelle-Institut Frankfurt
"Pre-Experiments for Leak Monitoring by means of Acoustic Emission Analysis."

RS 289-RS 292. The project RS 29o must be seen in close connection with the projects RS 289 to RS 292.

9. References

P. Jax, Battelle-Institut Frankfurt
"Vorversuche zur Leckageüberwachung mit Hilfe der Schallemissionsanalyse"
BF-R-62.944-1

B. Raible, Allianz-Zentrum für Technik Ismaning
"Körperschallmessungen am Reaktordruckbehälter und am Primärkreislauf von Kernkraftwerken."
Teil 1: Vorversuche für ein Leckageüberwachungssystem.
Bericht zum Forschungsvorhaben RS 97 A.

1o. Degree of availability

All reports specified under 9. are available at the "Gesellschaft für Reaktorsicherheit, Glockengasse 2, 5ooo Köln.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78-31.12.78	10.4	RS 289
Vorhaben/Project title: Geräuschmessungen an Kernreaktoren im Frequenzbereich 1 kHz bis 1 MHz Noise Measurements on Nuclear Reactors in the Frequency Range Between 1 kHz and 1 MHz		Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Battelle-Institut Frankfurt Abteilung Werkstoffverhalten
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.8.77	31.5.79	Dr. P. Jax
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	December 31, 1978	

1. General Aim

evelopment of a surveillance system for leaks on an acoustic basis for the primary system of a nuclear power plant. Compared with the methods previously used for leak detection, this surveillance system is to show fast response and higher sensitivity at a substantially improved accuracy of locating leaks.

2. Particular Objectives

The structure-borne sound emitted by nuclear reactors and experimental plants during operation in the frequency range from about 1 kHz to about 1 MHz is to be recorded and analyzed. Use of various sensor types - the institutions participating in this project, i.e. Allianz-Zentrum für Technik, Ismaning; Gesellschaft für Reaktorsicherheit, Garching; and KWU, Erlangen, will make measurements in the low-frequency range up to 80 kHz, while Battelle will perform measurements in the ultrasonic range - is to provide information about the optimum frequency range for a surveillance system for leaks and the most suitable sensor type. For this purpose the noise patterns typical of PWR und BWR systems will be examined and compared with the "leakage noise" measured on experimental plants and artificial leaks.

In addition, the acoustic wave propagation in the real structures is to be investigated by using an artificial noise source.

1.1.78 - 31.12.78 RS 289

3. Research Program

3.1 Extensive measurements of the operating noise at the cooling system of a PWR while putting it into operation (hot trial runs) at ten measuring points, and measurements of acoustic wave propagation

3.2 Reference measurements on a PWR during power operation, with four measuring points at the reactor cooling system

3.3 Reference measurements on a BWR during power operation, with four measuring points at the reactor cooling system

3.4 Measurements of operating noise on an experimental loop of the KWU in Erlangen, with four measuring points, for comparison with the operating noise measured on nuclear reactors, to be used as a basis for future investigations

3.5 Development of an artificial noise source for acoustic wave attenuation measurements

4. Experimental Facilities, Computer Codes

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5. Progress to Date

Re 3.1

Recording of the operating noise on magnetic tape during the first and second hot trial runs performed in the Gösgen-Däniken nuclear power plant, measurement of the r.m.s. values and determination of the frequency spectra. Comparison of the operating noise with the useful signals from a chance leak at the site of a pressure transducer. Transmitting of continuous test signals by means of the noise source (Item 3.5) to simulate a leak, determination of the attenuation of the signal along the steam line and of the useful ratio of signal to operating noise at various sites in the primary circuit.

Development of simplified, theoretical concepts for supering background noise and calculating on the basis of this, the optimum ratio of useful signal to background noise for leak surveillance systems.

Re 3.2
Further evaluation of the measured data stored on magnetic tape, with regard to frequency spectrum and r.m.s. values.

Re 3.3
Installation of nine sensors at five measuring points in the primary system of the Isar nuclear power plant. Recording of operating noise during operation at 55 to 98 percent capacity and during a shutdown phase of the reactor when several operating systems were switched on and off in the "cold" state. Measurement of the r.m.s. values and determination of frequency spectra.

6. Results
Re 3.1 to 3.3
The operating noise of highest intensity is measured on the primary loop of the two pressurized-water reactors. The "cold" main coolant pipe (connection between pump and reactor pressure vessel) has the highest absolute noise level. The noise levels measured for the boiling-water reactor, on the other hand, are more than 20 dB lower. On average the frequency spectrum decreases with increasing frequency.

The operating noise measured was compared with the leakage noise measured under project RS 193. An estimate for the most unfavorable case shows that local surveillance of the primary systems of nuclear reactors for the detection of leaks is possible, and that it should also be feasible to detect a minor leak (orifice nozzle with a leakage rate of about 50 kg/h) in the high-frequency range around 500 ± 100 kHz at a distance about 2.5 m. Because of the lower operating noise level of BWRs, the range of operation and the possible frequency range of a leak surveillance system seem wider for this type of reactor.

A chance leak at the site of a pressure transducer of the Gösgen nuclear power plant (in the main coolant pipe; estimated leakage rate 10 to 50 kg/h)was detected due to the rise in r.m.s. values at neighboring sensors (by up to 100 % of the background noise level). It should be noted, however, that in this operating phase the background noise level was lower than in the case of normal power operation by a factor of 3, as only one main coolant pump was switched on.

The result available so far, according to which the signal/background noise ratio of the leakage noise improves with increasing frequency, was substantiated by the theoretical considerations: Thus, for a pipe of infinite length, with given sensor distance a, an optimum signal/background noise ratio is obtained if $a \lesssim x_0/2$ (formula for propagation of a point source: $A \sim A_0 \cdot \exp(-x/x_0)$). For a finite length of pipe (with reflection loss at both ends), where no external noise is received at the ends, the optimum is $x_0 \rightarrow \infty$ and thus given at low frequencies, since $x_0 \sim 1/f$ (see below). However, in the realistic case of a finite length of pipe (length a), where external signals are received, the optimum signal/background noise ratio again results to have a finite value of x_0. Depending on whether the noise level outside the pipe is higher or lower than inside it, the optimum is shifted with respect to the above value $a \lesssim x_0/2$ of the pipe of infinite length towards smaller or larger values of x_0.

Re 3.1

The results obtained for the attenuation of the signal along the steam line are in agreement with a theory developed by Dr. Fischer, KWU, under project RS 291. According to this theory, the decrease in the r.m.s. value depends on the relation U/x_0 (U = circumference). With respect to measuring accuracy, the following relation applies: $x_0 \sim 1/f$ (f=center frequency of measuring cascade). Locating of the noise source is possible through measurement of the relative signal level at two measuring points $\bar{A}_1$ and $\bar{A}_2$ (amplitude in dB units). With the exception of a small area around the two measuring points, the following equation is applicable: $\bar{A}_1 - \bar{A}_2 = \alpha . x$ (x = distance of the leak from measuring point 1). α is a constant which is dependent on U/x_0.

Re 3.3

In contrast to normal operating conditions, operation of the plant in cold state involves high noise levels when various operating systems are switched on. The "TD" system, which supplies the pressurized bearing water for the axial-flow pumps, was found to be a dominating noise source. As to measuring accuracy, the following equation still applies: $\bar{A}_1/\bar{A}_2 \sim \alpha . x$, with $\alpha \sim 1/f$.

7. Next Steps

Re 3.1 to 3.3

Further evaluation of the measurements

8. Relation with other Projects

The research is being conducted in close cooperation with the Allianz-Zentrum für Technik, Ismaning; with KWU, Erlangen; and with Gesellschaft für Reaktorsicherheit, Garching (projects RS 290 to RS 292). It is based on results of the research-projects RS 193.

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	10.4	RS 291

Vorhaben/Project Title	Land/Country
Geräuschmessungen an Kernreaktoren im Frequenzbereich ca. 1 kHz bis ca. 1 MHz Noise Measurements on Nuclear Power Plants within a Frequency Range of Approx. 1 kHz up to Approx. 1 MHz	FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 214, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 8. 77	31. 3. 79	Dr. Fischer
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	31. 12. 78	

1. General Aim:

To develop a leakage monitoring system on an acoustic basis for pressure boundaries of safety related reactor systems.

Compared with current leakage detection methods, this type of monitoring system is expected to achieve quick response, enhanced sensitivity and a considerably improved localization of leakages.

2. Particular Objectives

Subsequent to the development of an acoustic leakage monitoring system for reactor cooling systems, it is the aim of the program to evaluate and analyse the sonic emission in the structure of operating reactor and test plants within a frequency range of approx. 1 kHz up to approx. 1 MHz. By the use of different types of transducers, the optimum frequency range for a leakage monitoring system, and the most suitable transducers will be determined.

This sound emission, here called "operating noise" depends - among other things - on the operating condition of a plant, i.e. on the parameter of the cooling system medium. Thus typical "operating noise" sound patterns will be established for both PWR and BWR plants within the above frequency ranges which then will be compared with "leakage sounds" observed in test plants

and simulated leakages.
Furthermore the sound emission caused by an artificial noise source in real structures will be investigated.

3. Research Program

Central point of the experimental part of the program will be broad-band measurements of operating noises during start-up and during power operation.
The test matrix is as follows:

3.1 Comprehensive measurements of the operating noises observed on a PWR during start-up operation (hot functional test) using 10 measurement points for the reactor cooling system and sound propagation and attenuation measurements.

3.2 Reference measurements on a PWR during power operation using 4 measurements points in the reactor cooling system.

3.3 Reference measurements performed in a BWR during power operation using 4 measurement points in the reactor cooling system.

3.4 Measurements of the operating noises observed in a test plant as compared to those obtained from reactors giving a basis for further investigations with 4 measurement points.

3.5 Development of an artificial noise source for attenuation measurements.

4. Experimental Facilities

Plant measurements performed both during commissioning phase and in operation as well as preparation for such measurements including necessary installation and assemblies, access to hard- and software required as well as close co-operation with commissioning and/or operating personnel.

5. Progress to Date

Zu 3.1

In January measurements were made during the first hot test operation in NPP Gösgen Däniken.
The 10 measurement locations were:
- RPV head
- main coolant loop (hot)
- main coolant loop (cold)
- S.G. primary side
- S.G. secondary side
- feed water line
- main steam line
- volume control line
- pressurizer

During the second hot functional test the operating noises were measured at the above ten locations for various operating conditions.
Operating conditions were: p = 150 bar, T = 290 °C
- all main coolant pumps running
- all main coolant pumps off
- only main coolant pump 2 operating
- pump switchings

In addition, broad-band, continuous noise was locally excited by an artifical noise source at various points on a main coolant line and recorded above the operating background noise.
Changes in the operating noises during shutdown of the reactor cooling system to the cold, pressureless condition at the end of the 2nd hot functional test were determined through analog recordings of the RMS-value as observed at selected measuring points.

Systematic attenuation measurements on a main steam line were performed in the cold pressureless condi-

tion. The main steam line was instrumented with sensors at the S.G. outlet nozzle as well as approx. 20 m distant from the outlet. Between the two sensor positions, continuous noise was locally excited by the Battelle noise source placed at 17 different locations with two values of distance between the orifice and the surface, and this noise was then measured by the two sensors.
Also, in the cold conditions, the artificial noise source was used for different measurements on the main coolant line and the background signal of all measurement channels recorded.

To 3.2 Evaluation of the Nov. 77 measurement data was continued.

To 3.3 Measurement of the BWR KKI during nuclear test operation was prepared for. The sensor-locations were polished, sensor mountings and sensors were assembled, measurement cables were laid; an instrument cabinet was installed outside the reactor building, equipped and the measurement chains were checked. 5 measuring points for 4 sensors each were installed in the following locations on the reactor coolant circuit:
1 m. p. on a main steam line in the vicinity of the RPV outlet nozzle;
1 m. p. on the same line in the vicinity of the pipe penetration outside the pressure reduction system (DAS);
1 m. p. on a feed water line in the vicinity of the RPV inlet nozzle;
1 m. p. on the same feed water line in the vicinity of the pipe penetration outside the DAS;
1 m. p. in the nozzle area of the RPV bottom.

In the cold condition during different operating modes the following measurements were performed: background noise, noise caused by large purge loop, by make-up water supply by seal water supply and by axial pumps together with these auxiliary systems.

During the conduct of the power tests operating noise was measured at different power levels and at corresponding operating conditions:
55 %, 70 %, 75 %, 85 %, 96 %, 98 %, 100 %
(thermal power).

6. Results

To 3.1 In the above mentioned operating conditions, RMS values were measured in different frequency bands. Spectra were prepared by means of analog tape recordings.
The RMS values of the primary system during 2nd hot functional tests were higher than those obtained during the first hot functional test (in the 2nd hot functional test the core has been loaded).
The attenuation measurements were evaluated for frequency ranges between 40 - 70 MHz and 70 - 100 kHz. Path attenuation is low, however it increases with increasing frequency.
The representative recordings obtained during shut-down indicate that the primary system noise initially increases with decreasing pressure and decreasing temperature, and then increases to more than 20 dB.

To 3.2 The periodically increased noise observed on the two main coolant lines in Biblis-A was further investigated.
The reason seems to be a pump-induced vibration of the main coolant lines.

To 3.3 The RMS values in various frequency ranges were measured in the above mentioned operating conditions. At full power the RMS values at a single location decrease in the following order:
- main steam line near the RPV and near the pipe penetration
- feed water line near the RPV - RPV bottom - feed water line near the pipe penetration
  (RMS values measured by 10 kHz high-pass filter).

7. Next Steps

Evaluation of current measurement data is supposed to be completed by end of 1978.
Operating noises will be investigated with regard to their intensity and spectral composition.
The data obtained will be compared with these noises obtained from actual or simulated leaks.
With these results as basis the question of the basic feasibility of an acoustic leakage monitoring system will be resolved.
Final conclusions pertaining to the optimum frequency range will be drawn from the dependence of the frequency on operating and leakage noises as well as from path attenuation.

8. Relation with Other Projects

The RS 291 program will be performed in conjunction with the RS 289 program (Battelle-Inst. Frankfurt), RS 290 (AZT, Garching) and RS 292 (GRS, Garching).

9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	10.4	RS 292

Vorhaben/Project Title	Land/Country: FRG
Geräuschmessungen an Kernreaktoren im Frequenzbereich von ca. 1 kHz bis ca. 1 MHz Noise Measurements in Nuclear Reactors within the Frequency Range from ca. 1 kHz to ca. 1 MHz	Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Gesellschaft für Reaktorsicherheit (GRS) mbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.8.1977	31.3.1979	Dr. D. Wach
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	31.12.1978	

1. General Aim

Fundamental investigations to develop a new leakage detection system for pressurized components of nuclear power plants based on acoustic information. The advantages compared with commonly used systems should be: higher sensitivity, prompt alarms and improved localizability of the leak.

2. Particular Objectives

Analysis of operational background noise in the primary systems of light water reactors measured with different sensors within the projects RS 289, 290 and 291 in order to answer the questions for optimal frequency ranges, qualification of sensors and transferability of measurement results from a test loop to real reactor conditions. The influence of the structure to the sound propagation should be investigated by use of correlation techniques (2.1). Further on the available signals measured by the different partners should be evaluated by a standardized analysis system for comparative reviewing purposes (2.2). In this context special importance has to be laid on the tests where an artificial noise source is used. By comparison of the analysis results with and without the artificial noise source the form-filter influence of the reactor structures, the real sound propagation mechanisms and the transferability of corresponding test loop results should be investigated (2.1).

3. Research Program

3.1 Cooperation within the working committee of the research programs RS 289 to RS 292 with respect to measurement planning and performing as well as with respect to the discussions of the analysis results.

3.2 Copying, digitalization and signal analysis of the different detectors at defined test phases using analyzers and computer programs of the stochastic signal analysis. Preparing of the results for comparative reviewing.

3.3 Evaluation of defined measurements with correlation methods in order to investigate the signal source structures, coherences of distance-depending sound intensities, and the reactor-specific structures effects on the sound propagation and the filtering influence to the operational background noise.

4. Experimental Facilities, Computer Codes

The digital programs DIGI, DIGCHN, TUMX92, TUMX68 are used as well as storage- and special analyzer systems.

5. Progress to Date

Ad 3.1 At the nuclear power plant Gösgen measurements of the operational background noise were performed in cooperation with the other groups during the hot functional tests at 10 representative primary system sections. In addition artificial noise sources (blowing, brushing, etc.) were used to measure the sound propagation in real reactor structures with and without background noise. Corresponding measurements at already operating light water reactors were performed at representative positions in the PWR Biblis A and the BWR Isar I.

Ad 3.2 The signals of the different detectors used in the research programs RS 289 to RS 291 were copied on magnetic tapes, digitalized and evaluated by computer codes and stochastic analysis systems. Using one and the same analysis system the conditions were provided for objective comparative reviewing of the different detector results.

Ad 3.3 The signals of detectors of the same type were investigated by use of correlation techniques. Auto- and cross-correlation functions were evaluated as well as cross-power spectral densities and coherence functions. This was done when applying different types of artificial sound sources with and without background noise.

6. Results

Ad 2.1 The correlation results with leakage-generated sound at a test loop showed that in the signals of low-frequency detectors the structure transfer plays a decisive role as a forming filter to the noise and sound signals (and this means that important consequences have to be derived for the usefulness of any simplified propagation laws). Therefor special interest was directed to coherence evalutions of measurements with artificial leakage sound. The results of the digital analysis showed that in the lower frequency range (< 50 kHz) resonance structures in the coherence functions can be found. Since these are at the detector eigenfrequencies they cannot be used for sound propagation determinations e.g. derived from phase relations. In the higher frequency range no correlations were found in signals of relatively closely positioned detectors and measurements without coolant flow, i.e. without background noise. Proposed that the copies of the magnetic tapes have no effects influencing the results especially in the higher frequency range in a negative manner, this means that non-linear effects (like highly dispersion-infected sound propagation, mode alteration, multiple-path problems) have to be considered. These are important results with respect to model conceptions for sound source structures and propagation mechanisms.

Ad 2.2 The comparative analysis of different detectors are not yet completed, so that no final statement can be given. However, it should be mentioned that the spectral investigation of the measurements with and without artificial (point) noise sources applied at the hot loop of the main coolant pipe showed the most distinctive measuring effect in the broad-banded high-frequency probes.

7. Next Steps

The still missing analyses of coherence of the low-frequency sensor signals and the evaluations for the comparative reviewing will be performed. The results will be documented. In cooperation with the other partners the final report about operational background noise will be written.

8. Relation with other Projects

The project corresponds directly with the projects RS 289 (Battelle), RS 290 (AZT) and RS 291 (KWU). It is also based upon the results of the already finished projects RS 97 A and RS 193.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	10.4	RS 297
Vorhaben/Project Title Schallsignale in Kreiselpumpen bei 2-Phasenströmung Sound Signals from Centrifugal Pumps working in different Flow-Regimes		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Technische Universität Berlin, Institut für Kerntechnik
Arbeitsbeginn/Initiated 1.11.1977	Arbeitsende/Completed 31.10.1980	Leiter des Vorhabens/Project Leader Prof.Dr.-Ing.U.Wesser
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

The purpose of this project is the development and test of a flow-regime detection method for centrifugal pumps. To this end, sound signals in the andible and ultrasonic range are to be employed. The method is also to be used as an evaluation aid for the project RS 109 Blow-Down experiments.

2. Particular Objectives

In phase A of the project, preliminary experiments are to be carried out at the IKT (Institute for Nuclear Engineering, Berlin). It is thus to be determined whether the sound spectra of centrifugal pumps exhibit characteristics which are dependent on the flow regime concerned, and further whether the necessary measurement time is appropiate for Blow-Down conditions. The successful conclusion of these experiments is necessary for participation in the LOBI pump experiments at WCL-Hamilton-Canada (Phase B). In these experiments recordings will be made to enable evaluation of the later Blow-Down tests. The sound behaviour is to be described in such a way that a particular sound signal may be attributed to a corresponding flow regime. This prozedure will then be applied to recordings obtained from pumps in the Blow-Down rig of project RS 109. In phase C of the project, recordings made in Blow-Down experiments in Ispra will be evaluated by methods developed in phase B.

3. Research Program

There are three main research activities

3.1 Preliminary investigations with pump test facilities at IKT

3.2 Data collection during LOBI pump tests at WCL-Hamilton-Canada, and mathematical description of the results.

3.3 Participation in the Blow-Down experiments in Ispra (RS 109) and analysis of the collected data.

4. Research Facilities

The Blow-Down facility of project RS 109 at IKT was used after modification by the inclusion of a centrifugal pump.

5. Progress to date

5.1 Phase A of the project has been completed.

5.2 Data collection during the LOBI pump tests at WCL-Hamilton. Participation in Group I and Group II measurements. Suitable software is being developed.

6. Results

Detectors and measurement facilities suitable for the preliminary experiments at WCL were selected. In the selection, consideration was given to the expected operation conditions of the WCL facilities. After preparatory talks in Canada, the measurement system was set up and tested in July 1977. In the following Group I LOBI pump tests, thermohydraulic data was recorded parallel with sound data collection. In total, about 300 test points were taken for each of the five detectors. The sound data was obtained under operating conditions which approached stationarily, and was stored on magnetic tape. Initial evaluation shows that differing flow regimes in the pump lead to differing signal structures. The measurements were continued in the Group II pump tests in October/November 1977 and completed on the Nov. 18, 1978. After the return transport of the measurement facilities from Canada, a systematic analysis of the results was commenced. Software was developed and tested by way of preparation of a mathematical description of the sound behaviour of the pump.

7. Next Steps

Evaluation of data from the LOBI pump tests (Group I and II) will be carried out uwing previously developed computer programs. The software itself will be constantly revised and improved. Signal properties will be sought which characterise in a distinct fashion the sound behaviour of the pump under differing 2 phase flow-regimes. According to the project scheme, preparations will be made for Blow-Down experiments in Ispra.

8. Relations to other projects

1) Partial use of project RS 135 facilities.
2) Cooperation with project RS 284 "cavitation signals"
3) Planned inclusion of the measurement prozedure within the project RS 109 at the Blow-Down rig of EURATOM-Ispra and at the test rig of WCL-Canada

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	10.4	RS 284

Vorhaben/Project Title	
Statistical Analysis of Cavitation Sound from Emergency Cooling Pumps. Statistische Analyse von kavitationsspezifischen Schallsignalen aus Notkühlpumpen.	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Technische Universität Berlin, Institut für Kerntechnik

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.11.1977	31.10.1980	Prof.-Dr.-Ing.U.Wesser
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	Dec. 31. 1978	

1 General Aim

The purpose of the project is to determine the variation of statistical signal quantities with the onset and development of cavitation in emergency cooling pumps. The method is to be suitable for application under the conditions of a nuclear reactor, and at high unwanted noise levels.

2 Particular Objectives

Theoretically, the connection between thermofluiddynamic occurences in cavitation zones and the resultant sound generation will be investigated. Sound transducer already used in cavitation tests will be optimised with regard to the different measurement applications of the research project. Software for statistical analysis of signals will be adjusted to the characteristic features of cavitation signals, and extended. The experimental work aims are to find measuring devices suitable for detection of cavitation signals from emergency cooling pumps. Data from cavitation tests on an experimental pump yields characteristic signal quantities. To test and develop the methods of signal detection (and subsequent evaluation), experiments with original sized emergency cooling pumps are projected.

3. Research Program

There are five main research activities

3.1 Measurement of cavitation sound signals from original-sized pumps (in particular emergency cooling pumps)

3.2 Measurement of cavitation sound signals using laboratory test facilities

3.3 Further development of sound detector systems
3.4 Further development of signal evaluation methods
3.5 Theoretical treatment of cavitation sound phenomena

4. Experimental Facilities, Computer Codes

Experiments are carried out using experimental facilities at IKT (Institute for Nuclear Engineering, Berlin) and also facilities provided by industry. In the latter case the tests concern large pumps and original sized emergency cooling pumps.

5. Progress to Date

5.1 Measurement of cavitation sound signals from large pumps
Measurements on original sized reactor pumps under simulatea emergency conditions.

5.2 Measurement of cavitation sound signals using laboratory facilities; measurements on a radial pump, on a flow grating and on special experimental cavitation facilities in industry.

5.3 Further development of the detector systems
Design, building and tests of detector systems with integral amplifiers and filters.

5.4 Further development of the evaluation methods
Development and tests of software

5.5 Theoretical treatment of cavitations and phenomena.
Development and tests of software
Comparison of theoretical and experimental values to check evaluation method.

6. Results

The first evaluation of cavitation sound signals collected from a reactor emergency cooling pump under simulated emergency conditions showed a signal structure dependant on the operating conditions and the degree of contamination of the flow medium. In addition, in the case of cavitation tests on industrial test rigs, it was shown that the onset and development of visually recognizable cavitation phenomena resulted in characteristic signal forms. In order to handle the large amount of data, it was considered wise to first develop appropiate software. A more through evaluation of the data is thus still outstanding. The new computer based evaluation method works without external data store. The program calculates and displays the amplitude distribution and the statistical quantities for description of the distribution curves. The new evaluation method can be put into action in the first quarter of 1979, after further test runs.

High sensitivity detectors were used for the data collection, designed specially for ultrasonic cavitation signals. The detectors performed satisfactory in tests on an original reactor pump. Further development of detectors led to systems with integral amplification and filtering. These were tested successfully on the IKT rig in cavitation measurements.
The evaluation method applied to the above investigations was supported by theoretical considerations. With the aid of a mathematical model theoretical values were calculated which were compared with experimentally determined values. The results of the calculations confirmed the dependance of this quantity on the cavitation condition concerned.

7. Next Steps

The data from previous cavitation test runs on test pumps, on a flow grating and on an original reactor pump under simulated accident conditions are stored on magnetic tape. By use of appropiate software statistical signal quantities will be calculated for aa test runs. Characteristic statistical quantities, cavitation counts and the development of cavitation phenomena during the test will be compared. The evaluation should show in what way the detected cavitation sound signals are influenced by flow geometry and flow medium quality. With this information projected cavitation investigations will be prepared in accordance with the future course of the project.

8. Relations to Other Projects

1) Cooperation with project RS 259 "NaK-Testloop"
2) Cooperation with project RS 297 "Sound Signals from Centrifugal Pumps working in different flow Regimes"

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 – 31.12.78	10.4	GKSS – 212
Vorhaben/Project Title Untersuchungen des Kavitationsverhaltens von Nachkühlpumpen von Leichtwasserreaktoren für die langfristige Nachwärmeabfuhr unter normalen und extremen Betriebsbedingungen. Investigations on the cavitation behavior of LWR Residual Heat Removal Pumps for the Long-Term After-Heat Discharge under normal and extreme operation conditions.		**Land/Country** FRG **Fördernde Institution/Sponsor** BMFT **Auftragnehmer/Contractor** GKSS-Geesthacht Institut für Anlagentechnik
Arbeitsbeginn/Initiated 1.10.77	**Arbeitsende/Completed** 31.12.83	**Leiter des Vorhabens/Project Leader** Wietstock
Stand der Arbeiten/Status Continuing	**Berichtsdatum/Last Updating** December 1978	**Bewilligte Mittel/Funds**

1. General aim

It is a supposition for the long-term after heat discharge out of the safety-tank of a LWR to have knowledge of the behavior of the residual heat removal pumps under normal and extreme operation conditions. It is the general aim of our researches, to carry out hydrodynamical tests on the pumps, which are installed to today respectively in future and to evaluate the technological limits under working conditions and to reduce conservative aspects.

2. Particular objectives

The maximum pump-flow with cavitation is given by the point of intersection of the characteristics of NPSH available and NPSH required. It is the aim of this research programm to find out the influence of temperature, pressure quality of water and the inlet conditions on both mentioned characteristics. Furthermore we have to test the influence of the pump load under cavitations condition.

3. Research program

3.1 Investigation of the total head of the pump as a function of the NPSH-value until full developed cavitation (total head against zero). Parameters are load, temperature and gas content.

3.2 Determination of the under 3.1 mentioned NPSH-characteristics as a function of the impurity of water.

3.3 Determination of dynamical NPSH-values.
In the first place it has to be determined in what manner momentary changes of working conditions can put a pump into cavitation or can put it out off cavitation. Parameters are mentioned under 3.1 and 3.2.

3.4 Determination of the mode of oscillation as a function of cavitation conditions for the determination of additional stress conditioned by cavitation. (Impeller, bearing, ect.).

4. Experimental Facilities

For solving the above mentioned problems we use the pump test rig of the GKSS.
The rig can be operated at pressures up to 110 bar and temperatures up to 593 K. Due to the application of corrosion resisting steel (partly as plating) the use of fully desalinated water is possible. The installed flow meters can record flow quantities up to 6000 m^3/h. The tightness demanded for cavitation measurements has been achieved by careful execution of flange connections. The proved integral tightness amounts to 10^{-2} Torr liter/sec.
The essential component of the loop is a main circuit, consisting of a pressure tank with connected pipe loop. The pump for testing is mounted in the range of the NW 500 - loop, and discharges the medium through the main loop. Inserted flap controls the volume flow.
An auxiliary loop serves as purification circuit (when testing contaminated pumps) and partly as cooling system for the pump to be tested. Over a level control system water is led into the storage tank, which apart from storing serves for deaerating the water. The water level may be raised in any stage of service by means of a dosating pump. Furthermore it is possible to insert an additional pump into the circuit in order to amplify the investigations beyond the first quadrant.
The main dates of our test rig are:

System-pressure	110 bar
Temperature	max 320 °C
Flow-rate	160 - 6000 m^3/h

Power	2400 kW
Inner diameter of the vessel	1200 mm
Main pipe	500 / 300 mm
Material	plated steel / corrosion - resistant steel
Thightness	10^{-2} Torr liter/sec

For connection the pump to our test bed some adjustment work is to be done.

5. Progress to date

For definition of the specified research programm the discussion of a conception paper [1] began.

6. Results

7. Next steps

Definition and discription of the specified research program. Projecting the adjustment work.

8. Relation with other projects

RS 297 Sound Signals from Centrifugal Pump working in different flow regimes.

RS 284 Statistical analysis of cavitation sound from emergency cooling pumps.

9. References

[1] Investigations on LWR Residual Heat Removal Pumps. GKSS-Working paper 72 10 AT B 15

[2] R. Doose, A. Katsaounis, A. Saribaf, P. Wietstock. The pumpe test bed. GKSS-paper 73/I/16

10. Degree of Availability of the report

by GKSS.

	CLASSIFICATION: 10.4
TITLE (ORIGINAL LANGUAGE): Comportamento vibrazionale della colonna di combustibile Cirene - Fretting corrosion	COUNTRY: Italy
	SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Dynamics of fuel strings in axial flow - Fretting corrosion (HWR - Calandria tubes)	ORGANISATION: CNEN
	PROJECT LEADER: G. Lelli
INITIATED: 1974 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: July 1979	

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 facilities consist 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.

4. Project status

The dynamic behaviour of the fuel string has been analysed under various thermodynamic conditions (Specific flow 200 ÷ 260 g/cm^2s; steam quality 9 ÷ 21%; pressure 50kg/cm^2) The constraints condition has been varied as it has an influence on the dynamic behaviour.

5. Next steps

Other tests under different hydraulic and load conditions will be carried out in order to quantify the vibrational behaviour. Endurance tests will be carried out on a non instrumented test section.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Dynamics of fuel strings in axial flow - Fretting corrosion (HWR - Calandria tubes)	10.4

6. Relation to other projects and codes

   FULFLO

7. Reference documents

   - V.A. Mason, M.J. Pettigrew, G. Lelli, L. Kates, E. Reimer, "Dynamics of nuclear fuel assemblies in vertical flow channels: computer modelling and associated studies", Keswick Conference "Vibration in Nuclear Plant", May 1978, Paper 1.4.
   - G. Possa, P. degli Espinosa, G. Lelli, G. Vanoli, "Fuel string dynamics and pressure tube fretting-corrosion in the Cirene power channel", Keswick Conf. "Vibration in Nuclear Plant", May 1978, Paper 1.5.
   - V. Kuzelka, "Response of cohesive fuel bundles to random exciting forces due to a fluid flow-Part-1: mathematical model", TERM-RAP(77)6.

   Other classified technical documents have been edited.

8. Degree of availability

   To a limited extent.

   Contact person: G. Lelli, TERM-RAP, CNEN - CSN Casaccia, CP 2400, I-00100 Roma.

	CLASSIFICATION: 10.4
TITLE (ORIGINAL LANGUAGE): STATIC AND DYNAMIC TESTS ON "P40-D4" PROTOTYPE GRID FOR STEAM GENERATOR	COUNTRY: ITALY
	SPONSOR: Breda Termomeccanica (MI)
TITLE (ENGLISH LANGUAGE): STATIC AND DYNAMIC TESTS ON "P40-D4" PROTOTYPE GRID FOR STEAM GENERATOR	ORGANISATION: ISMES S.p.A. - Bergamo Dynamic Department
	PROJECT LEADER: Dr. Aldo Castoldi
INITIATED: January 1978 / COMPLETED: April 1979	SCIENTISTS: Dr. Carlo Colombo
STATUS: COMPLETED / LAST UPDATING: June 1979	

1. GENERAL AIM

The tests on the grid for a steam generator were performed to investigate its dynamic response and its mechanical resistance.

2. EXPERIMENTAL FACILITIES AND PROGRAMME

The tests performed were the following:

- static tests on two straight sections of the outer ring;
- dynamic tests with loads perpendicular to plane on the grid;
- dynamic tests with loads acting on plane on the grid;
- static tests with loads perpendicular to plane on the grid;
- static tests with loads acting on plane on the grid.

3. ESSENTIAL RESULTS

The analysis of the results obtained during the dynamic tests shows that the grid behaves with good approximation as a homogeneous system.

4. REFERENCE DOCUMENTS

- Static and dynamic tests on "P40-D4" prototype grid for steam generator. File No. 1431.

ISMES, V.le Giulio Cesare 29, I-24100 Bergamo.

N.V. KEMA	CLASSIFICATION : 10.4
TITLE : Ruisanalyse van de reactor en het primaire circuit	COUNTRY: THE NETHERLANDS SPONSOR : KEMA ORGANIZATION : KEMA
TITLE (ENGLISH LANGUAGE): Reactor and primary circuit noise analysis	PROJECTLEADER :
INITIATED : – LAST UPDATING : 1978 STATUS : – COMPLETED : 1977	SCIENTISTS : J. Hoekstra J. v.d. Veer E. Türkcan

General aim

On load surveillance of vital systems.

Particular objectives

Dynamic behaviour of a BWR.

Experimental facilities

Dodewaard nuclear power plant.

Project status

Still in progress.

Next steps

Not applicable.

Relation to other projects

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

Reference documents

None.

Degree of availability

Through the organizations KEMA and ECN.

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

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

11. MATERIALS AND MECHANICAL PROBLEMS IN NORMAL AND ACCIDENT CONDITIONS

Classification II	
Title 1 ZIRCALOY CORROSION STUDIES	Country U.K.
Title 2	Sponsor C.E.G.B
	Organisation BERKELEY NUCLEAR LABORATORIES
Initiated 1976 Completed 1979	Project Leaders
Status Complete Last updating	Dr.C.J. Wood Dr. T. Swan

1. General Aim

To determine the factors which influence the incidence of nodular oxidation of fuel and pressure tubes in boiling water reactors.

2. Particular Objective

To determine the combined influence of damaging radiation and galvanic couples on rate of zircaloy oxidation (in association with AERE, Harwell). To investigate the nature of the kinetic transition during oxidation.

3. Experimental Facilities and Programme

Studies have been carried out using damaging radiation from the Harwell Variable Energy Cyclotron. These investigated the effects of galvanic coupling and of the dissolved gas content in high temperature water.

4. Project Status

The project is complete and has been reported. No further work is planned.

Measurements during irradiation coupled with post-irradiation examination showed that the VEC experiments had successfully simulated a reactor environment, with greater oxidation enhancement in oxygenated than in hydrogenated water. This was attributed to an increase in the ionic conductivity of the oxide film. Galvanic coupling to stainless steel increased the rate further but did not initiate nodular oxidation possibly because the VEC runs were shorter than the necessary incubation period.

Classification 11.1	
Title 1 Fuel Element Behaviour under Irradiation **Title 2**	**Country** : Belgium **Sponsor** : BELGONUCLEAIRE **Organisation** : BELGONUCLEAIRE - CEN/SCK
Initiated : 1972 **Completed** : - **Status** : in progress **Last updating** : May 1979	**Project leader** : Mr. H. Bairiot

1. General Aim

Assessment of fuel element behaviour under irradiation (including mixed oxide).

2. Particular Objectives

(a) Fuel densification study and evaluation of the qualitative and quantitative effects (e.g. neutronic point-of-view, physical mechanisms, attempt to deduce empirical correlations for predetermination calculations) ;

(b) Fuel behaviour under accidental conditions ;

(c) Power ramp capabilities ;

(d) Surveillance of statistical behaviour of Pu fuel in BWR and PWR power plants.

3. Experimental Facilities and Programme

- In-pile (BR-2 reactor) and out-of-pile tests for fuels and fuel elements. Neutron radiography. Post-irradiation examination. Pool side inspection.
- VENUS critical facility for study of local power peaking effects with simulated axial gaps.
- Fuel assemblies irradiated in BR-3, SENA, DODEWAARD, GARIGLIANO, etc ...

4. Project Status

Progress to date : VENUS tests, irradiations in BR-2, BR-3 and SENA. Routine operation of codes COMETHE (steady state) for fuel behaviour, SPARTAN and THEATRE 3 for transient evaluation of fissile material inhomogeneity.

Essential results : Power peaking effects, fuel-clad interaction, fission products.
COMETHE was selected as "the best code" in a comparative evaluation performed by EPRI (March 1977), and used by about forty organizations (Utilities, Governmental, Manufacturers, etc.)

5. Next steps

- COMETHE transient under development.
- Experimental data base continuously expanded.

6. Relation with other Projects

Some actions in the frame of the "Pu recycle programme" sponsored by the ECC.

7. Reference Documents

8. Degree of Availability : proprietary

- Disclosure on a bilateral basis to be agreed.

	CLASSIFICATION: 11.1.
TITLE (ORIGINAL LANGUAGE): Fuel Rod Behaviour under Operating Transients	COUNTRY: BELGIUM
	SPONSOR: CEN/SCK
TITLE (ENGLISH LANGUAGE):	ORGANISATION: CEN/SCK
	PROJECT LEADER: Mr. HEBEL, W.
INITIATED: 1978 — COMPLETED:	SCIENTISTS: MM. DIRVEN, P. DETAVERNIER, W.
STATUS: Starting — LAST UPDATING: May 1979	

1. General Aim

Providing a versatile and compact inpile test rig for running dynamic transients of fuel rod power, of coolant flow and of cooler capacity.
Testing of fresh as well as of pre-irradiated LWR single fuel rods or small rod assemblies.

2. Particular Objectives

Verifying fuel rod operation limits and failure behaviour.

3. Experimental Facilities

Compact high pressure irradiation test rig with built-in forced cooling water circulation and control of flow-rate and of cooler capacity ("PCM" loop).
Reloadable test rig for BR2 reactor.

4. Project Status

Conceptual design of test rig terminated, work for construction in progress.

5. Next Steps

-

6. Relation with other Projects

Project proposal (called WRS) to test clusters of 45 PWR fuel rods in BR2 at LOCA and/or ATWS incidents.
Irradiation testing in BR2 reactor of LMFBR fuel rod behaviour under transient operating conditions (project "VIC") and with local blockage of sodium coolant flow (project "MOL 7 C").

7. Reference Documents

W. HEBEL (editor) : BR2 Reactor Review Meeting
MOL, June 1, 1978.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	11.1	06.01.06 (PNS 4235.3)

Vorhaben/Project Title	
Untersuchungen zum Einfluss des Oxidbrennstoffes und von Spaltprodukten auf die mechanischen Eigenschaften von Zry-Hüllrohren bei Störfalltransienten Investigations on the Influence of Oxide Fuel and Fission Products on the Mechanical Properties of Zry-cladding Tubes under Transient Conditions	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Kernforschungszentrum Karlsruhe Projekt Nukleare Sicherheit; IMF I

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1974	1980/81	Dr. P. Hofmann
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	December 1978	

1. General Aim

Evaluation of the extent of the inner Zry-4 cladding tube corrosion. Influence of the chemical interactions between oxide fuel, fission products and the Zry cladding material on the mechanical properties of Zry-cladding tubes under inert gas and oxidizing conditions at high temperatures.

2. Particular Objectives

- Investigation of the influence exerted by the oxygen potential of the UO_2 on the deformation and rupture behavior of Zry-4 tubing under transient conditions and in hypothetical loss-of-coolant accidents.
- Stress corrosion cracking behavior of Zry cladding tubes with respect to volatile fission products and fission product compounds.
- UO_2/Zry reaction experiments under Power Cooling Mismatch (PCM) conditions; determination of the reaction kinetics between UO_2 and Zry.

3. Research Program

3.1 Isothermal, isobaric as well as temperature and pressure transient experiments on short UO_2 or Argon containing Zry-cladding tube specimens under inert gas conditions.

3.2 Creep rupture as well as temperature and pressure transient burst tests under inert gas conditions on short Zry tubular specimens containing small amounts of iodine or other volatile fission products. Influence of UO_2, thin oxide layers on the stress corrosion cracking behavior of as-received and predamaged Zry cladding tubes. Determination of the critical iodine concentration. Examination of the internal tube surfaces of iodine containing burst specimens, using a scanning electron microprobe (SEM).

3.3 Isothermal annealing tests performed unter inert gas conditions on short Zry-tube specimen ... d with UO_2 pellets at temperatures up to 1500°C and UO_2/Zry contact pressures up to 80 bar. X-ray and electron microprobe investigations of the reaction products formed during the chemical interactions between UO_2 and Zry.

4. Experimental Facilities

ad 3.1 The burst tests are performed in the tube burst apparatus TUBA, In addition to the temperature of the cladding tube and the inner gas pressure in the tube specimen, the cladding tube diametral expansion is determined continuously using a light optical method.

ad 3.2 Tests on stress corrosion cracking (SCC) are mainly carried out in the SCC apparatus, part of them also in the TUBA apparatus.

ad,3.3 The UO_2/Zry reaction experiments are performed in a high-pressure gas autoclave system under inert gas conditions.

5. Progress to Date

- Burst experiments on as-received and pre-damaged Zry-4 cladding tubes to evaluate the influence of fission product elements on the deformation and rupture behavior of the cladding tubes.
- Burst experiments on as-received and on the inside preoxidized Zry-4 tube specimens to determine the critical iodine concentration above which Zry will fail as a result of stress corrosion cracking.
- Investigation on the change of the structure of Zry-4 in dependence of temperature and annealing time.
- Annealing Experiments to study the chemical interactions between UO_2 and Zry at high temperatures under Power Cooling Mismatch conditions.
- Destructive post-irradiation-examinations (metallography) of burst fuel rods of PNS task 06.01.08 (PNS 4237) under which in-pile experiments are carried out on the fuel rod behavior in a loss-of-coolant accident.
- Annealing experiments with irradiated UO_2 both isothermally irradiated samples and samples from PNS-LOCA-fuel rod F6 (PNS-task 06.01.08).
- Analytical chemical investigations to determine the oxygen concentration of Zry-4 tube specimens filled with UO_2 and iodine after burst experiments.

6. Results

- The burst experiments with CsI, I, Cs_2O, Cs_2ZrO_3, Cs_2MoO_4, ZrI_4, TeI_4, $ZrTe_2$, Cs_2Te, Cd, Sb, Sn, Te, Se and I_2O_5 filled tube specimens were performed unter inert gas conditions at about 800°C. All the above mentioned elements and compounds yielded a decreased burst strain of the Zry-4 cladding tubes in comparison to the argon filled reference specimens. However, only the tube specimens that contained elemental iodine or iodine compounds, with the exception of CsI, failed in a brittle mode at 800°C due to stress corrosion cracking of the Zry /1 - 5, 9/.

  As shown by first experiments on pre-damaged Zry-4 cladding tubes in the pressence of iodine, major defects in the cladding tube (notches in the cladding tube inner surface $\lesssim$ 150 μm) will result in tube failure due to stress corrosion cracking already at relatively low mechanical stresses. Both the burst strain and the time until rupture are greatly reduced by the cladding tube defects. The cladding tubes rupture at the notched points practically without any local necking /1 - 4, 9/.

- The critical iodine concentration, above which the Zry-4 cladding tube fails due to stress corrosion cracking, depends primarily on the Zry-temperature and on the presence of oxide layers on the inner cladding tube surface. Is the critical iodine concentration exceeded, the Zry-4 cladding tubes fail at temperatures below 850°C with a pronounced reduction at the burst strain. Above 850°C iodine has a negligible influence on the burst strain of the Zry-4 tubing. Thin oxide layers on the inner cladding tube surface (7 - 15 μm) cause a shift of the critical iodine concentration to smaller values. At 780 - 800°C the critical iodine concentration undergoes variations between 1 mg/cm^3 for cladding tubes used in the as-received state and 0.1 mg/cm^3 for cladding tubes subjected to preliminary inner oxidation.

- The structure of Zry free of stress depends on the temperature and annealing time. An assessment of the cladding material temperature by means of structure examinations seams therefore possible.

- The time and temperature dependent growth of the phases [α-Zr(O)$_a$, (U,Zr)-alloy, α-Zr(O)$_b$] formed during the chemical interactions between the UO_2 and Zry was determined experimentally between 1000 and 1500°C. The kinetic data of oxygen uptake from inside by the Zry cladding were corrected accordingly with a view to the initial UO_2/Zry-phase boundary and compared with the oxygen uptake from outside. The comparison shows that the oxidation of the Zry cladding from inside by UO_2 is but slightly lower than the oxidation from outside by steam. Besides, the oxygen content of the oxygen stabilized α-Zr(O)-phases was determined by means of Auger Electron Spectroscopy. The oxygen concentration varies between 4 and 6 wt.% depending on the temperature and annealing time /6,7/.

- The first burst fuel rods exposed to a loss-of-coolant accident transient in the FR 2 reactor have been subjected to metallographic postirradiation examinations to determine among other things the exent of inner corrosion. It becomes evident from the test results that the cladding tube inner surface does not undergo uniform axial and radial oxidation. In the vicinity of the burst region the inner oxidation of the cladding tube is strongest; at greater distances from the burst region there exist some zones free from oxidation. The inner oxidation of the cladding tube is due mainly to the steam penetrating intothe fuel rod after bursting. This is also indicated by the different oxide layer thickness on the cladding tube surface (generally stronger outside as compared to inside) /8/.
- UO_2 samples irradiated at 1000°C to high burnups crumbled during annealing at 1200 and 1600°C. UO_2 samples irradiated at ≤ 1000°C to 1% burnup showed an additional swelling of about 2% after annealing for 2 hrs at 1600°C.

  UO_2 samples of the PNS rod F 6 (2.4% mean burnup) were annealed at different times at temperatures between 1200 and 1600°C. The maximum swelling was found to be 4.3% after 3 hrs at 1500°C. The swelling at 1200°C was only 0.4%. The swelling depends on the irradiation conditions, too.
- The influence of iodine on the oxygen uptake of Zry-4 cladding tubes have been studied under transient LWR-conditions. The results show that iodine causes a lower oxygen uptake of preoxidized and UO_{2+x}-filled Zry-4 cladding tube specimens in comparison to specimens free of iodine. The influence of iodine is more pronounced at temperatures above 700°C where the ZrO_2-layers spall off more easily. Possibly, iodine also prevents the oxide layer formation in the case of the UO_{2+x}-filled Zry-tube specimens /1, 10/.

7. Next Steps

- Continuation of isothermal and transient tests on stress corrosion cracking of Zry cladding tubes due to iodine and other volatile fission products. Determination of the critical iodine concentrations. Influence of thin oxide layers and defects on the inner cladding tube surface on the critical iodine concentration.
- Stress corrosion cracking experiments on pre-cracked Zry cladding tubes (sharp edges, crack due to fatigue). Examinations of tubular specimens by the scanning electron microscope prior to and after the tests relative to stress corrosion cracking.
- Destructive examination of irradiated fuel rods of PNS task 06.01.08/01A (PNS 4237).
- Isothermal annealing experiments with irradiated UO_2 at various temperatures in oder to determine the fuel swelling and fission gas release behavior.
- Zry-4 cladding tube structure examinations in dependence of temperature and annealing time.
- Analytical chemical investigations of Zry-4, oxide fuel and simulated fission products.

8. Relation with Other Projects

PNS 4235.1, 4235.2, 4235.4 and 4237.

9. References

/1/ PNS second Semi-annual Report 1977, KfK 2600 (1978), p. 343

/2/ PNS first Semi-annual Report 1978, KfK 2700 (1978)

/3/ P.Hofmann
Hat Jod bei LWR-Störfällen einen Einfluss auf das Verformungs- und Bruchverhalten von Zry-4-Hüllrohren ?
Reaktortagung 1978, Hannover, S.545 - 548

/4/ P.Hofmann
Does Iodine exert an influence on the deformation and rupture behavior of Zry-4 cladding tubes in LWR accidents ?

Enlarged Halden Program Group Meeting on LWR Fuel Performance, Leon, Norwegen, 4 - 9 June 1978

/5/ P.Hofmann
Influence of iodine on the strain and rupture behavior of Zry-4 tubing at high temperatures.
4th International Conference on ZIRCONIUM IN THE NUCLEAR INDUSTRY, Stratford-upon-Avon, England, 26 - 29 June 1978

/6/ P.Hofmann, C.Politis
Chemical interactions between UO_2 and Zry-4 in the temperature range between 900 and 1500°C
ibid

/7/ P.Hofmann, C.Politis
UO_2/Zry-Reaktionen bei Power Cooling Mismatch-Störfällen
Reaktortagung 1978, Hannover, S. 541 - 544

/8/ P.Hofmann, C.Petersen, G.Schanz
In-pile experiments on the LWR fuel rod behavior during a hypothetical LOCA. Reports about the nuclear experiments A2.1, A2.3, and B1.2
Results of destructive post-irradiation esaminations (metallography).

/9/ P.Hofmann
Einfluss des Spaltproduktelementes Jod auf das Verformungs- und Bruchverhalten von Zry-4-Hüllrohren bei Temperaturen zwischen 600 und 1000°C

/10/ H.Schneider
Einsatz der Auger-Elektronenspektroskopie zur Bestimmung von Sauerstoff-Konzentrationsprofilen in Zry-Oberflächenschichten und deren Quantifizierung.
9. Colloquium on Metallurgical Analysis, Vienne, Austria, Oct. 1978

10. Degree of Availability

/1/ - /7/, /9/, /10/ unrestricted distribution
/8/ restrected distribution; internal reports

Berichtszeitraum Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.1978-31.12.1978	11.1	06.01.06 (PNS 4235.4)
Vorhaben/Project Title Berstversuche an Zircaloy-Hüllrohren unter kombinierter mechanisch-chemischer Beanspruchung Burst-tests of Zircaloy Cladding Tubes under Mechanical and Chemical Load		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe Projekt Nukleare Sicherheit / IMF
Arbeitsbeginn/Initiated 1977	Arbeitsende/Completed 1980/81	Leiter des Vorhabens/Project Leader L. Schmidt, S. Leistikow
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Cladding Material Behaviour under Accident Conditions.

2. Particular Objectives

Investigation of the High Temperature Behavior of Zircaloy 4 Cladding Tubes under LOCA typical conditions.

3. Research Program

Burst-tests of indirectly heated Zircaloy Tubes under Mechanical and Chemical Load.

4. Experimental Facilities

Facility for Burst Testing under Fuel Interaction, Steam Oxidation, and LOCA Conditions (FABIOLA).

5. Progress to Date

All necessary components of the steam generation system were completely mounted and the assembling was started. The installation of the electrical control and measure devices and the power supply units was almost finished.
A special computer program for recording the important test data, including the on-line measurement of creep deformation, was set up. For the graphic plotting of these test data the available computer will be supplemented by a plotting system, which was specified and ordered.
The function test of the X-ray cinematographic device, measuring the tube expansion continiously, could be performed. An additional lead screen for shielding this device was designed and given into fabrication.
An on-line TV-width analyzer, with computer interface, for digital storage of the cladding tube expansion values, was specified and its purchase was pre-

-2-

pared. This system allowes to plot the creep curve immediately after an experiment.
For the production of fuel rod simulators a series of Zircaloy tubes one-sided closed by welding and some internal heaters were manufactured. So it is possible to assemble and weld the complete fuel rod simulators within a short time. An experimental program was set up and a matrix was achieved to begin after start-up and testing of the basic functions of the FABIOLA-test facility with REBEKA-single rod reference tests.
A preliminary test section for the design of the electrically heated fuel rod simulator was constructed and within the fuel rod simulators two types of internal heaters (tube and spiral) were tested. The experiments were used also for confirmation that the already existing power supply was sufficient.

6. Results

Using the prototyp heaters different ramp tests were performed during which at a rod power of 50 W/cm a heating rate of 10 K/s were reached. The effective wall temperature of the shroud was about 500-750°C, the steam temperature at the entrance was about 200°C. The number of ramps performed was 60 in case of the spiral heater, 40 for the tube heater.
Post-test evaluation of the used heaters show that the heaters are intact and could be removed from the pellets and the Zircaloy test tube without major problems to be used for further experiments. It could be seen that the existing power was sufficient for the supply of both types of heaters. During a heat-up experiment of the shroud up to 760°C it could be proved that the mechanical stability of the whole arrangement is sufficiently high.

7. Next steps

- Complete assembling of the FABIOLA-test facility.
- Start-up and testing of the X-ray cinematography including all necessary licensing.
- Pretests to measure the azimuthal temperature distribution under original conditions.
- REBEKA-single rod reference experiments.
- Post-test analysis of the ruptured Zircaloy tubes.
- Order of the TV-width analyzer.
- Testing of the plotting system.

8. Relations to other programs

All others of KfK/PNS, KWU und USNRC.

9. References

-

10. Degree of Availibility of the Reports

-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	11.1	GKSS-214

Vorhaben/Project Title	Land/Country
Kriechverhalten von Hüllrohren unter Bestrahlung und innerem/ äußerem Uberdruck Creep behaviour of fuel canning under irradiation and internal/external overpressure	FRG
	Fördernde Institution/Sponsor BMFT
	Auftragnehmer/Contractor GKSS, Geesthacht Institut für Werkstofftechnologie und Chemie

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1977	1983	H. Wilhelm
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	December 1978	

1. General

Designing of fuel rods for LWRs is mainly influenced by the creep resistance of the used zircaloy-tubes and their ductility. At the beginning of life the tubes have to withstand the external overpressure of the cooling water. Therefore their diameter will decrease by creeping during operation of the reactor. In the same measure, as the local burnup increases, the swelling of the UO_2-pellets results in an internal contact pressure which will be enhanced by thermal-induced growing of the pellets during load-change-operation of the reactor. By this mechanical pellet-clad-interaction and especially with increased power fuel rods may fail releasing a lot of fission products into the colant from where they may come to the open air by leakages. Moreover, above all high radioactivity of the coolant means higher irradiation level for the operators during maintenance, service-inspection and refuelling. Therefore general aim is to investigate the longtime creep behaviour of zirconium alloy tubes under reactor conditions.

2. Particular objectives

The mechanical data of the inpile behaviour of the canning material, especially the inpile creep strength, are not yet known. The creep data known from the literature have been obtained mainly by uniaxial creep experiments. Due to the texture of the canning tubes in connection with the remarkable anisotropy of the α-Zirconium-structure mechanical properties which strongly depend on the direction of stress are resulting. Today there are only few predictions for a hexagonal system like Zirconium concerning the influence of irradiation on creep strength and ultimate elongation of tubes with a clearly distinguished texture and hoop stress conditions represented by a stress rate factor of $\sigma_t/\sigma_a = 2 : 1$.

For LOCA-transient calculations of fuel rods data are needed. Especially data concerning the probable geometry of the canning at any point of operation time have to be obtained as well as a better knowledge of the available ductility remaining after creep under irradiation.

Therefore, the influence of neutron irradiation on the creep behaviour of commercial canning tubes and of those being developed (i.e. Zy4 with different heat treatment and the Zr-alloy ZrNb3Sn1) will be investigated in close operation with KWU. The creep loci which will describe the creep behaviour of canning tubes with various stress conditions has to be determined by using specimens uni- and biaxial loaded as well.

3. Research Program

After a review of the test program the field of test parameters is given in the following table.

Biaxial stress

Set of specimen	Nr.	3		5	6	4		1	2
Test temperature	[°C]	300		350		375		400	
Tangential tension	[MPa]	70 and 100							
Internal overpressure of the canning		x		x		x		x	
External overpressure of the canning					x				x
Planned test duration	[h x 750]	8		8	8	4		4	8

Uniaxial stress

Set of specimen	Nr.			2	4			1	3
Test temperature	[°C]	300		350		375		400	
Axial tension	[MPa]			100	70 or 120			100	70 or 120
Test duration	[h x 750]			8	8			4	4

Additional investigations with regard to texture and precipitations will be carried out to obtain an objective valuation of the resulting creep data.

4. Experimental Facilities, Computer Codes

The tests will be carried out in the GKSS-developed irradiation capsules K1 (uniaxial) and K3 (biaxial) and in the test reactor FRG 2. Each irradiation period will be followed by Hot-Cell examinations.

5. Progress to Date

In consequence of the shut down of the FRG 2 in spring 1978 only one irradiation cycle has been carried out during the year 1978. While results of the last period are going to be evaluated and interpretated, preparation of the next run have been started.

6. Results

It seems to be obvious, that an internal overpressure leads to a little higher creeprate than an external overpressure under equivalent stresses.

7. Next steps

After restart of the test reactor FRG 2 irradiation cycles and investigation after each run will be continued.

8. Relation to other projects

-

9. References

-

10. Degree of availability

-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 – 31.12.78	11.1	GKSS-213

Vorhaben/Project Title	
Untersuchung des Betriebsverhaltens defekter Brennstäbe Investigation of the behavior under operation condition of defected LWR fuel rods	Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor GKSS, Geesthacht Institut für Werkstofftechnologie und Chemie

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Loader
1975	probably 1983	Dr.J.Ahlf
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General aim

Investigation of the contribution of defect fuel rods to the primary circuit radioactivity of light water reactors by means of model experiments.

2. Particular objectives

Enlargement of knowledge on the spread out of a defect in a LWR fuel rod after occurence of a cladding perforation. Producing data on fission product release after the occurence of a fuel rod perforation until scheduled shutdown of the reactor.

3. Research program

3.1 Development and construction of an irradiation devise for irradiation LWR fuel rods under PWR conditions.

3.2 Irradiation experiments on fuel rod segments with different burnup, which are perforated before or during irradiation. Continuous measurement of fission product release.

3.3 Post irradiation examination of irradiated test fuel rods.

3.4 Modelling fission product release of defect LWR fuel rods.

4. Experimental Facilities, Computer Codes

4.1 For the experiments an irradiation rig with designation Z1 is being developed, which will be operated in FRG-2.

4.2 The postirradiation examination will be performed in the hot cells.

5. Progress to Date

5.1. A test rig built in the foregoing year equipped with an electric heater simulating the test fuel rod has been operated. At the beginning different failures occured with the circulating pump for the primary water, mainly at the bearings. Several design modifications have lead to a pump which has been operated more than 3000 h without trouble.
With the test rig a number of experiments to demonstrate the safety of the device have been carried out
- performance at pump failure
- heat transfer with pump out of operation
- loss of pressure experiments

A test facility for the primary pump was designed and built.

5.2. Design modifications have been made at the inpile-section in order to improve the reloadability in the hot cell.

5.3. Construction and testing of components for the irradiation device have been continued.
The safety report was completed.

5.4. Details of the investigation program have been discussed with KWU as regards water chemistry control during the irradiation tests, fission product release measurement (on-line and off-line), design of the test fuel rods.

6. Results

Not to be expected before 1980

7. Next steps

Further operation of test rig and pump test facility to demonstrate longtime reliability of components and instrumentation.
Completion of the irradiation device including licensing procedure.
Preparatory work for the defect fuel irradiation experiments.
Further development of computer programs to model the operational history of the test fuel rods (GELS-FREX, ORIGEN) and fission product release.

8. Relation to other projects

-

9. References

-

10. Degree of availability

-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	11.1	RS 203

Vorhaben/Project Title	
KFA/KWU-Leistungsrampen Testbrennstabbestrahlungen 1976/1977 KFA/KWU-Power Ramps Fuel Rod Irradiation Tests 1976/1977	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik B 21, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 4. 76	28. 2. 78	H. Knaab
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	31. 12. 78	

1. General Aim

Power ramp experiments in HFR Petten were carried out with PWR standard fuel rods of different production parameters, in order to investigate the reason for fuel rod defects, due to fuel-clad interaction (PCI).

2. Particular Objectives

The mechanism causing fuel rod defects by positive power ramps is not yet well known. In practice, fuel rod defects have been discovered after local power increases in power reactors and in power ramp tests in Halden, Studsvik and Risø .

In order to study this problem, the irradiation and load change behaviour of PWR fuel rods with oxide fuel (UO_2) were investigated in HFR Petten. The fuel rods, assembled to segmented rods, have been preirradiated in the Obrigheim power station up to 7000 - 25000 MWd/t(U) burnup.

3. Research Program

3.1 Unloading from reactor and sectioning of segmented PWR fuel rods, intermediate inspection of the fuel rod segments/test fuel rods.

3.2 Neutronradiographic investigations on the test fuel rods.

3.3 Installation of the test fuel rods into the irradiation capsules and performance of power ramp experiments in HFR Petten.

3.4 Post-irradiation examination of the test fuel rods.

3.5 Evaluation of the power ramp experiments and report of the results.

4. Experimental Facilities

The test fuel rods have been preirradiated in the Obrigheim power station for 1-2 reactor cycles. After preirradiation the test fuel rods were inserted into test capsules and ramped under representative LWR conditions in the pool-side-facility of HFR Petten. The fuel rods were ramped to linear heat generation rates of $q' \leq$ 680 W/cm with ramp rates of $\dot{q}' \leq$ 100 W/cm min and heat generation rate increases of $\Delta q' \leq$ 350 W/cm. The tests were performed as a joint project of KFA Jülich and Kraftwerk Union.

The following parameters were varied:

fuel:	UO_2 of different structures, density, pellet geometry
cladding tube:	production parameters (e.g. graphite on inner surfaces)
fuel rod:	pressurization, burnup.

5. Progress to Date

To 3.2
- Pre-irradiation and intermediate examination of additional segments (2-cycle rods with fuel of reduced density and 1-cycle rods with modified pellet geometry).

To 3.4
- performance of ISM ramps on 6 fuel rods
- Gamma scanning on 5 fuel rods
- post-irradiation examination of 6 fuel rods in the HFR pool

- visual inspection after irradiation on fuel rods from a total of 9 experiments
- measurement of gamma-spectra of deposits observed at defective locations on two fuel rods, deposits collected to be taken to hot cell examination
- computational verification of all transients with revised evaluation program
- determination of the fission gas content of 7 ramped segments and 1 reference segment
- He-leakage test on 2 ramped segments
- preparatory work covering 5 additional cross sections and two cladding tube internal examinations.

To 3.5 - continuation and completion of the test evaluation

6. Results

To 3.3 To date the failure thresholds of standard fuel rods during full power reactor operation were determined for both the In-situ (IS) and Start-up (SU) experiments. In all these experiments the rod power was initially increased to a stable power level of approximately 300 W/cm.
In the SU-ramps the power was then immediately increased to approx. 550 W/cm applying different power ramp rates in a range of 0.3 to 95 W/cm·min. This high power level was then maintained for 72 hrs. At the IS-ramps a 400 hour irradiation phase at a rod power of about 300 W/cm was performed prior to the ramp. In order to simulate power reduction caused by either partial load operation of the reactor or by Xenon-oscillation, rod power was then reduced to 150 W/cm for 1 hr; subsequently it was increased again to various ramp power (420 to 680 W/cm) at a power increase rate of 100 W/cm min and held constant for 48 Hrs.
The results showed that the failure thresholds below which no ramp failures have to be expected significantly depend on the fuel rod burnup. The failure threshold of 1-cycle fuel rods with burnups of approx. 12 GWd/t(U) is approx. 480 W/cm, while the failure threshold of 2-cycle

fuel rods with burnups of approx. 23 GWd/t(U) seems to be approx. 420 W/cm.

In the first quarter of 1978 the ISM-series was continued with two 2-cycle segments (fuel with reduced density) and four 1-cycle segments.

With these modified IS-ramp experiments the 1-cycle fuel rods, after being exposed to a 400 hr irradiation and subsequent temporary power reduction, were raised in power to 480 W/cm at a ramp rate of 100 W/cm·min; after a 2 hr hold time, it was increased to 550 W/cm at a reduced rate. The corresponding power values for 2-cycle fuel rods were 420 and 490 W/cm.

This modified IS (ISM)-ramp basically corresponded to the ramp types which actually occur in the power reactor. These ramp types show high power ramp rates in the lower power range, but slow down with increasing rod power. The upper boundary value in advanced PWRs is set by the reactor power limitation system.

Current ISM ramps with a reduced power ramp rate < 5 W/cm·min in the upper power region did not reveal any activity release, i.e. cladding tube perforations have not been observed, either with 1-cycle or with 2-cycle fuel rods. Eddy current testing on the 2-cycle fuel rods, however, showed a relatively strong ridge formation.

To 3.4 The two ISM tests with burnups of approx. 26 GWd/t(U) showed that ramp powers of 490 W/cm which most probably lead to fuel rod failures of IS-ramping can be reached without rod defects using the ISM-method. The ISM tests within a burnup range of about 12 GWd/t(U) also showed that ramp power exceeding the failure threshold power of 480 W/cm which resulted in failures with IS-experiments at times did not cause any 1-cycle rod failure.

The destructive post-irradiation examination were continued. The effect of the discontinuous dish closure, particulary noticeable in the neutron radiography of a fuel rod, could be verified in the predicted location by a longitudinal micrograph.

7. Next Steps

Project completed.

8. Relation with Other Projects

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9. References

M. Gärtner, F. Sontheimer, W. Vogl
"KFA/KWU - Power Ramp Experiments - Irradiation of Test Fuel Rods 1976/77" - Final Report on the BMFT funded project RS 203, October 1978

10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	11.1	RS 203 A
Vorhaben/Project Title KFA/KWU-Leistungsrampen Testbrennstabbestrahlungen 1976/77 KFA/KWU-Power Ramps Fuel Rod Irradiation Tests 1976/77		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik B 21, Erlangen
Arbeitsbeginn/Initiated 1. 3. 78	Arbeitsende/Completed 31. 12. 80	Leiter des Vorhabens/Project Leader H. Knaab
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating 31. 12. 78	Bewilligte Mittel/Funds

1. General Aim

During the restarts of power reactors after refueling and by power changes during reactor operation and rapid control assembly movements, fuel rods can be damaged due to the resulting local rod power changes. The aim of the power ramp tests in the HFR in Petten is to determine the causes of such fuel rod defects, particularly the pellet/cladding interaction (PCI), and to ascertain the defect thresholds for PWR and BWR fuel rods of standard design with selectively varied manufacturing parameters.

2. Particular Objectives

Information is still lacking on the mechanisms involved in the occurrence of fuel rod defects under changing fuel rod loads by positive power ramps. This problem of fuel rod design has become particularly topical due to fuel damages during local power changes in power reactors and during power ramp experiments at Halden, Studsvik and Risø.

In the case of rapid load changes of fuel assemblies with burnups of > 5,000 MWd/t(U), fuel rod defects may occur particularly in the power reactor recommissioning phase after refueling, during rapid movement of control assemblies and reactor load changes. These damages are traced largely to the considerable increase of fuel rod power exceeding the peak power released up to that time and explained by the resulting mechanical/chemical interaction of the UO_2 pellets with the cladding tube. In order to solve this problem, the irradiation and load change behaviour of test fuel rods with oxide fuels (UO_2) shall be examined in the scope of power ramp experiments with segmented PWR and BWR fuel rods in the HFR Petten after pre-irradiation in Obrigheim (KWO) or Würgassen (KWW) Nuclear Power Plants. A large number of segmented fuel rods has already been inserted into the power reactors and pre-irradiated to burnups of up to 5,000 to 30,000 MWd/t(U).

3. Research Program

3.1 Removal of segmented PWR and BWR fuel rods from the power reactors and intermediate inspections

3.2 Neutron-radiographic analyses of the test fuel rods/segments

3.3 Installation of the test fuel rods in irradiation capsules and performance of power ramp experiments in the HFR Petten

3.4 Post-irradiation examinations of the test fuel rods

3.5 Intermediate evaluation of power ramp experiments and reporting.

4. Experimental Facilities

After pre-irradiation over 1 to 3 reactor cycles in the Obrigheim or Würgassen power reactor the test fuel rods are being exposed to power ramps ($q' \leq 620$ W/cm, $\dot{q}' \leq 100$ W/cm min, $\Delta q' \leq 350$ W/cm) in the pool-side-facility of the HFR in Petten/Netherlands. The test capsules shall be instrumented for future tests with inductive displacement transducers to measure the changes in fuel rod length. The tests are being performed as a joint project of KFA Jülich and KWU.

The following parameters of the test fuel rods/segments are varied:

Fuel:	UO_2 Density, pellet geometry
Cladding Tube:	Manufacturing parameters, graphitisation of internal surface of some segments
Fuel rod:	Prepressurization, burnup

5. Progress to Date

To 3.2 Inspection of suitable, pre-irradiated segments prior to ramping (1-cycle and 2-cycle standard segments as well as segments with graphitised internal cladding tube surfaces and with ovalities locally induced during manufacture; furthermore segments with fuel pellets of "long taper" shape and rods with fuel of reduced density, with varied pellet geometry, with unground pellets etc).

To 3.4

- Performance of 8 IS ramps
- Performance of 15 ISM ramps
- Post-irradiation examination on 23 fuel rods in the reactor pool
- γ-scans of 23 rods
- Test of a computer program for analyzing the γ-scan profiles
- Transport of 4 PWR and 15 BWR rods from the Karlstein hot cells into the hot cells in Petten and of 19 rods in the reverse direction
- Fission gas analyses, metallography and examination of the inner clad surface in Karlstein
- Evaluation of isotope γ-scans on ramped fuel rods and continuation of general test evaluation
- Determination of the fission gas content of 2 rods
- Visual inspection of 10 rods prior to irradiation
- Assembly of one bundle of 10 segments and of the bundle of 14 segments
- Preparation of tests with BWR segments (capsules, accessories, reconstruction of BWFC systems).

- Performance and evaluation of diameter measurements on 12 segments
- Performance of eddy current examinations on 18 segments

6. Results

To 3.4
- The ISM ramp series on five 2-cycle segments of standard design with a cold gap in the range of the lower tolerance limit (140 µm) did not lead to any defects even with ramps of 490 W/cm rod power at 5 W/cm·min (≙ 60 %/h related to 490 W/cm).
- It has been shown with six 1-cycle segments of PWR standard design (two of them having local ovalities of up to 100 µm induced artificially during manufacture) with cold gaps in the range of the lower tolerance limit (140 µm) that 550 W/cm rod power at 5 W/cm·min (= approx. 50 %/h related to 550 W/cm) can be achieved during ISM ramping without any defect of the fuel rod.
- The post-irradiation examinations (PIE) in the pool (eddy current examination and neutron-radiography). confirmed the good results of the 1-cycle and 2-cycle rods.
- Rapid (100 W/cm·min) IS ramps to 490 W/cm were performed with two 2-cycle segments with graphitised internal cladding tube surfaces and pellets of 18 ° shoulder slopes without any defects.
- A rapid IS ramp (100 W/cm·min) to 480 W/cm with a 2-cycle segment with fuel pellets of "long taper" shape resulted in a rod defect within 26 minutes.
- Analogously to the general release of fission gases, I and Cs are released at a fuel rod power above approx. 400 W/cm proportionally to the power.
- Further diameter measurements on ramped segments confirm the statements already made after the first diameter measurements: no measurable change of average rod diameter by the ramps, but occurrence of ridges at fuel pellet interfaces. A clear correlation between ridge height and defect occurrence of a segment could not be found up to now.

7. Next Steps

- Continuation of the IS and ISM ramp experiments on PWR segments with varied manufacturing parameters. The parameters are: cold gap size, local ovality, "long taper" pellets, internal cladding tube graphitisation, helium fill gas pressure of 1 bar, IS ramps with 3-cycle segments.
- Ramping of BWR fuel rods.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

	CLASSIFICATION: 11.1
TITLE (ORIGINAL LANGUAGE): Pålidelighed af brændselselementer	COUNTRY: Denmark SPONSOR: Risø National Laboratory
TITLE (ENGLISH LANGUAGE): Fuel Reliability	ORGANISATION: Risø National Laboratory PROJECT LEADER: Ib. Misfeldt

INITIATED: 1975	COMPLETED:	SCIENTISTS:
STATUS: In progress	LAST UPDATING:	

1. General Aim:

To develop and verify a computer model for the statistical evaluation of LWR fuel performance.

2. Particular Objectives

The computer code FRP, Fuel Reliability Predictor, calculates the distributions for parameters characterizing the fuel performance and failure probability. The statistical methods employed are either Monte Carlo simulations or a low order Taylor approximation. Included in the computer system is a deterministic fuel performance code, FFRS. The distributions for all material data utilized in the fuel simulations are estimations from the best available information in the literature. For the failure prediction, a stress corrosion failure criterion has been derived. The failure criterion is based on data from out-of-reactor stress corrosion experiments performed on unirradiated and irradiated zircaloy with iodine present.

3. Experimental Facilities and Programme

Danish ramp testing programme.

4. Project status

4.1. Progress to date: The program FRP is fully developed.

4.2. Essential results: The deterministic fuel performance code and the failure criteria have been verified by comparison with a large number of ramp experiments.

Next steps

a) Further verification of the deterministic model by comparison with ramp experiments
b) Improvements to the statistical models for the material properties.

6. Relation with Other Projects

Utilizes data from the Danish ramp testing programme for verification. Utilize the Danish fuel model WAFER for "calibration" where no data are avaiable.

7. Reference Documents

1. Misfeldt, I. The Reliability of Nuclear Fuel. In: Probabilistic Analysis of Nuclear Reactor Safety. Topical Meeting May 8-10, 1978, Los Angeles, California. Vol. 2 (American Nuclear Society, Hinsdale, Ill., 1978), paper 3.2, 1-12.

2. Misfeldt, Ib (1978). FFRS: A Computer Program for the Thermal and Mechanical Analysis of Fuel Rods. Risø Report No. 373,53 pp.

3. Misfeldt, Ib (1978), Probabilistic Assessment of Light Water Reactor Fuel Performance. Risø Report No. 390, 63 pp.

4. Misfeldt, Ib. Failure criteria for the probabilistic fuel performance code FRP. To be presented at the 5th SMIRT conference, Berlin 13-17 August 1979.

Classification: 11.1	
Title 1 Ramp Testing of UO_2-Zr Fuel	COUNTRY: Denmark SPONSOR: Risø National Laboratory ORGANIZATION: Risø National Laboratory
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

Not generally available.

	CLASSIFICATION: II-1
TITLE (ORIGINAL LANGUAGE): Fuel Modelling	COUNTRY: Denmark SPONSOR: Risø National Lab.
TITLE (ENGLISH LANGUAGE): Fuel Modelling	ORGANISATION: Risø National Laboratory PROJECT LEADER: N. Kjær-Pedersen
INITIATED: 1972 | COMPLETED:	SCIENTISTS:
STATUS: Progressing | LAST UPDATING:	

1. General Aim

To develop and maintain an up-to-date computer model of fuel pin performance under nominally normal operating conditions.

2. Particular Objectives

The code considers one pellet-size segment of a fuel rod. A finite difference model of the pellet is coupled with a thin shell model of the cladding. Pellet cracking is directly accounted for, based on certain assumptions relating to crack patterns. Secondary and primary creep of fuel and clad are represented. The model aims at an analysis of local stresses and strains in the cladding, including ridge formation, as well as fission gas release, to form a basis for a failure probability estimation.

3. Experimental Facilities and Programme

Danish ramp testing programme.

4. Project Status

4.1. Progress to date: The third version of the model is in full production. Permits 20 axial nodes.

4.2. Essential results: Proven capability of predicting measured ridge-heights within a factor of two (ref. 2.). Generalized failure criterion based on contact pressure between fuel and clad.

5. Next Steps

Maintain present version.

6. Relation with other Projects

Utilizes data from Danish overpower test programme for verification.

7. Reference Documents

1. N. Kjær-Pedersen, "A New Version of the LWR Fuel Performance Model WAFER", 4th International Conference on Structural Mechanics in Reactor Technology, San Francisco, U.S.A., 15-19 August 1977, Paper No. D 1/3.

2. N. Kjær-Pedersen, "WAFER-2. A Code for Thermal and Mechanical LWR Fuel Performance Modelling", IAEA Specialists Meeting on Fuel Element Performance Computer Modelling, Blackpool, UK, 13-17 March 1978.

3. N. Kjær-Pedersen, "WAFER-3. An Extended Version for High-Speed Analysis of Rods with an Axial Power Profile," 5th International Conf. on Structural Mechanics in Reactor Technology Berlin. W. Germany 13-17 August 1979, Paper No. D 1/5

140.1.06		11-1
TITRE SYNTHESE DES RESULTATS DES IRRADIATIONS EXPERIMENTALES PORTANT SUR L'INTERACTION COMBUSTIBLE-GAINE, EN VUE DE L'ETABLISSEMENT D'UN CRITERE DE RUPTURE.		Pays FRANCE
		Organisme Directeur CEA/Dg CS et EDF
TITLE (Anglais) SYNTHESIS OF PCI EXPERIMENTAL RESULTS FOR DERIVING A CLADDING RUPTURE CRITERIUM.		Organisme exécuteur CEA/DMECN/DMG
		Responsable
Date de démarrage 1/1/79	Etat actuel A LANCER	Scientifiques
Date d'achèvement 31/12/79	Dernière mise à jour 20/12/78 (nouvelle fiche)	

1. OBJECTIF GENERAL :

Faire la synthèse des résultats des irradiations expérimentales réalisées au CEA ou avec sa participation dans lesquelles des ruptures de gaines ont été observées.

2. OBJECTIFS PARTICULIERS :

1. Faire la synthèse de l'ensemble des irradiations expérimentales en cause.
2. Effectuer un bilan des ruptures de gaines observées.
3. Déterminer leurs causes et les circonstances dans lesquelles elles se sont produites, variation de puissance notamment.
4. Evaluer leurs conséquences.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME :

Sans objet : étude bibliographique.

4. ETAT DE L'ETUDE :

1. Avancement à ce jour :

Sans objet : étude à lancer.

2. Résultats essentiels :

Sans objet : étude à lancer.

5. PROCHAINES ETAPES :

Sans objet : étude ponctuelle.

6. RELATION AVEC D'AUTRES ETUDES :

Cette étude ayant pour objectif final d'évaluer le risque de ruptures de gaine à prendre en compte dans les réacteurs, est directement en support de l'analyse de sûreté. Elle devrait, par ailleurs, contribuer à permettre d'orienter dans l'avenir les programmes EDITH, CYPHON, CRUSIFON et DEPTO-JET.

7. DOCUMENTS DE REFERENCE :

Sans objet : étude à lancer.

8. DEGRE DE DISPONIBILITE DES DOCUMENTS :

Sans objet.

TITRE
DEMETER : DEVELOPPEMENT D'UN CODE DE CALCUL DE SURETE POUR LES ELEMENTS COMBUSTIBLES DES REACTEURS A EAU ORDINAIRE EN REGIME ACCIDENTEL.

Pays
FRANCE

Organisme Directeur
CEA/DgCS

TITLE (Anglais)
DEMETER : DEVELOPMENT OF A SAFETY CODE FOR PRESSURIZED WATER REACTOR FUEL ELEMENTS UNDER ACCIDENTAL TRANSIENT CONDITIONS.

Organisme exécuteur
CEA/DMECN/D.TECH

Responsable

Date de démarrage
1/1/77

Etat actuel
EN COURS

Scientifiques

Date d'achèvement
31/12/80

Dernière mise à jour
MISE A JOUR N° 2 AU 20.12.78

1. OBJECTIF GENERAL :

Disposer d'un code de calcul des éléments combustibles des réacteurs à eau ordinaire sous pression rendant compte de leur comportement mécanique et thermique ainsi que de l'oxydation de leur gaine au cours d'un accident de perte du réfrigérant primaire.

2. OBJECTIFS PARTICULIERS :

1. Juger de l'importance relative des différents phénomènes mis en jeu et faire la liste de ceux à prendre en compte dans le code.
2. Rechercher les modèles les plus aptes à les représenter. S'ils font défaut, les établir à partir de l'expérience acquise par le CEA.
3. En particulier, participer au développement et au perfectionnement du modèle de déformation de gaine, à partir du programme EDGAR.
4. Fournir aux responsables du programme POSEIDON, et en accord avec eux, les modèles qui leur sont nécessaire.
5. Fournir à la CISI les éléments nécessaires au développement des modules et du code DEMETER. Coordonner cette action.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME :

Sans objet.

4. ETAT DE L'ETUDE

1. Avancement à ce jour :

   Un code relatif à l'accident de dépressurisation dans le cas particulier du programme PHEBUS, CUPIDON a été écrit. Ce code est opérationnel dans sa version définitive.

   Un modèle simplifié de déformation de gaine a été établi à partir de résultats du programme EDGAR.

   Ce code et ce modèle doivent servir de point de départ au code DEMETER.

   Un modèle destiné au transfert de données en provenance du code RESTA pour définir l'état de référence du combustible a été développé.

2. Résultats essentiels :

   Le code CUPIDON permet de calculer l'évolution des principaux paramètres (profil thermique du crayon, contraintes dans la gaine, déformation, épaisseur oxydée) au cours d'un accident, dans le cas simple d'un crayon non irradié, en utilisant les résultats du programme EDGAR.

5. PROCHAINES ETAPES :

1. Etablir la liste des modules à inclure dans les codes DEMETER et POSEIDON
2. Développer un nouveau modèle de déformation de gaine à partir du modèle existant dans CUPIDON et lancer un programme complémentaire dans EDGAR pour obtenir les données nécessaires.
3. Développer l'ensemble des modèles nécessaires à POSEIDON et DEMETER. Coordonner l'élaboration des modules destinés à DEMETER.
4. Qualifier le code DEMETER.

6. RELATION AVEC D'AUTRES ETUDES :

Ce programme est en relation, d'une part, avec le programme EDGAR-ZY qui fournit les données nécessaires à l'établissement d'un modèle de déformation de la gaine, et d'autre part, avec le programme PHEBUS. Le code CUPIDON a été développé en premier lieu pour effectuer les calculs de projet de ce programme. Les résultats de celui-ci seront utilisés pour qualifier le code DEMETER.

7. DOCUMENTS DE REFERENCE :

Code CUPIDON :
Note SETSSR/T/77.509 du 6 mai 1977. Proposition d'élaboration d'un code de comportement du combustible pour le calcul prévisionnel des essais PHEBUS.
Model of behaviour of PWR fuel element during a loss of pressure accident. IAEA Specialists meeting on fuel element performance computer modelling, Blackpool, UK, 13-17 mars 1978.

Rapport interne SPE/CIS/AS/78.011. Notice d'utilisation du code CUPIDON : code décrivant le comportement thermique et mécanique d'un crayon combustible de réacteur à eau sous pression au cours d'un accident de dépressurisation.

Code DEMETER :
Note D.Tech. SECS/SEECRE 78.123 du 18 avril 1978. Code DEMETER : note de présentation.

8. <u>DEGRE DE DISPONIBILITE DES DOCUMENTS</u> :

Notes internes, à l'exception de la présentation de CUPIDON à Blackpool.

145.2.02		11-1
TITRE EDGAR-ZY : COMPORTEMENT DES GAINES EN ZIRCALOY AU COURS D'UN ACCIDENT DE DEPRESSURISATION.		Pays FRANCE Organisme Directeur CEA/DgCS, EDF et FRAMATOME
TITLE (Anglais) EDGAR-ZY : ZIRCALOY CLADDING BEHAVIOUR UNDER LOSS OF COOLANT ACCIDENTAL CONDITIONS.		Organisme exécuteur CEA/DMECN/D.TECH Responsable
Date de démarrage 1/1/74	Etat actuel EN COURS	Scientifiques
Date d'achèvement 31/12/80	Dernière mise à jour MISE A JOUR N° 2 AU 19.12.78	

1. OBJECTIF GENERAL :

Déterminer les paramètres d'éclatement, la vitesse de déformation et l'épaisseur de la couche oxydée des gaines en zircaloy des éléments combustibles des réacteurs à eau sous pression, lorsqu'elles sont soumises à des variations transitoires de température et de pression représentatives d'un accident de perte de caloporteur.

2. OBJECTIFS PARTICULIERS :

1. Vérifier que le comportement des gaines permet le respect des critères de température et d'épaisseur oxydée maximales imposées par les autorités de sûreté.

2. Elaborer un modèle de déformation de gaine utilisable dans les codes CUPIDON et DEMETER.

3. Contribuer à la qualification de ces codes.

4. Contribuer à qualifier l'instrumentation prévue pour les essais PHEBUS et réaliser des essais d'accompagnement hors pile de ce programme.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME :

1. Pour les essais réalisés jusqu'à présent, des gaines neuves ont été utilisées exclusivement, dans une installation du D.TECH-SRMA à Saclay. Cette installation comporte un chauffage direct du tube par effet Joule et une pressurisation interne programmables de manière à reproduire

l'évolution de la température et de la contrainte caractéristique d'un accident. Les essais peuvent être réalisés sous vide, ou sous une atmosphère externe-inerte, ou bien de vapeur d'eau. Cette dernière possibilité constitue le cas nominal.

2. Un dispositif analogue, monté dans une cellule chaude du D.Tech-SECS (LECI) à Saclay, permettra après ses essais de démarrage, de réaliser l'étude complémentaire du comportement de gaines préirradiées. En particulier, la supposition de la restauration rapide du zircaloy sera vérifiée.

4. ETAT DE L'ETUDE :

1. Avancement à ce jour :

1. La première phase du programme ayant pour objet de vérifier que les caractéristiques de déformation à pression et vitesse de chauffe constantes sont en bon accord avec les résultats publiés à l'étranger est terminée.

2. Des essais à vitesses de pressurisation et de chauffage programmées selon des calculs types d'accident de FESSENHEIM 1 et sous vapeur d'eau ont été repris après qualification des mesures de température sans bouclage entre le calcul et l'expérience.

3. Un modèle de déformation de gaine utilisable dans le code CUPIDON a été élaboré.

2. Résultats essentiels :

Les résultats obtenus ont été généralisés sous la forme :

1. D'abaques donnant le temps de fluage nécessaire pour atteindre une déformation donnée, dans des conditions de pression initiale et de vitesse de chauffage données.

2. De courbes maitresses de déformation en fonction de la contrainte pour des températures et vitesses de chauffage données.

3. D'une relation donnant la vitesse instantanée de déformation en fonction de la contrainte, de la température et de la vitesse de variation de la température (modèle utilisé dans le code CUPIDON).

5. PROCHAINES ETAPES :

1. Reprise éventuelle, en accord avec les autres participants au programme, d'essais représentatifs d'accidents dans une centrale si nécessaire avec bouclage entre le calcul et l'expérience.

2. Elaboration d'un nouveau modèle de déformation de gaine destiné au code DEMETER ; utilisation du dispositif EDGAR comme l'un des moyens expérimentaux de qualification des codes CUPIDON et DEMETER.

3. Réalisation d'essais d'assistance au programme PHEBUS, notamment pour contribuer à la qualification de l'instrumentation envisagée.

4. Ultérieurement, transformation du dispositif pour vérifier les modèles

de cinétique d'oxydation existants et, éventuellement pour en élaborer de nouveaux tenant compte de la déformation, de la contrainte et de la vitesse de variation de la température.

5. Vérification de l'influence de l'irradiation sur le comportement des gaines et, éventuellement, vérification de l'influence de la présence de pastilles dans la gaine. Etude de l'influence de l'iode sur la déformation à rupture. En 1979, le dispositif en cellule chaude doit être mis en route, et le programme expérimental doit débuter.

6. RELATION AVEC D'AUTRES ETUDES :

Cette étude est en relation directe, d'une part, avec l'analyse de sûreté des réacteurs à eau sous pression (vérification du respect des critères de sûreté) et d'autre part, avec l'élaboration de modèles de déformation de gaine pour les codes de calcul (CUPIDON, DEMETER) et avec la qualification de ces derniers. Elle doit, d'autre part, apporter une contribution à l'exploitation des résultats du programme PHEBUS.

7. DOCUMENTS DE REFERENCE :

Les principaux résultats obtenus jusqu'à présent ont été publiés à la réunion de spécialistes du CSNI à SPATIND (Norvège) en septembre 1976. Comportement du gainage en zircaloy des éléments combustibles des réacteurs à eau sous pression pendant un accident de refroidissement-programme EDGAR (note D.TECH-RMA/76.710 de septembre 1976).

Note D.TECH-SRMA/73.533 d'avril 1973 : comportement des gaines en zircaloy pendant un accident de refroidissement d'un réacteur à eau ordinaire analyse bibliographique - Proposition pour les études CEA.

Note D.TECH-SRMA/GMM-76-2883 d'avril 1976 : comportement des gaines en zircaloy pendant un accident de refroidissement. Point des essais au 15 février 1976.

Note technique D.TECH-RMA (78)821 - mai 1978 : étude de la déformation des gaines en zircaloy pendant un accident de refroidissement.

8. DEGRE DE DISPONIBILITE DES DOCUMENTS :

Notes internes, à l'exception de la première citée.

TITRE	Pays
EOLE : ETUDE DE LA PERTE DE CHARGE DANS LE JEU ENTRE LE COMBUSTIBLE ET LA GAINE LORS DU TRANSFERT DE GAZ DU VOLUME LIBRE AU BALLONNEMENT.	FRANCE
	Organisme Directeur
	CEA/Dg.CS
TITLE (Anglais)	**Organisme exécuteur**
EOLE : PRESSURE DROP EVALUATION IN THE FUEL-CLADDING GAP DURING GAS TRANSFER FROM PLENUM TO BALLOONED SECTION.	CEA/DMECN/D.TECH
	Responsable

Date de démarrage	Etat actuel	Scientifiques
1/1/79	A LANCER.	
Date d'achèvement	**Dernière mise à jour**	
31/12/79	20/12/78 (nouvelle fiche)	

1. <u>OBJECTIF GENERAL</u> :

Evaluer la perte de charge à prendre en compte dans un crayon pendant son ballonnement au cours d'un accident de perte du réfrigérant primaire, en fonction du jeu, de la distance entre le ballonnement et le volume libre qui l'alimente, et de la viscosité du gaz.

2. <u>OBJECTIFS PARTICULIERS</u> :

   1. Réaliser un dispositif propre à effectuer ces mesures.
   2. En fonction des résultats obtenus, déterminer s'il est nécessaire d'introduire un modèle de perte de charge dans les codes DEMETER et POSEIDON.
   3. Eventuellement, modéliser les résultats obtenus.

3. <u>INSTALLATIONS EXPERIMENTALES ET PROGRAMME</u> :

Le dispositif expérimental, à définir et à construire, pourrait comporter un tube contenant des pastilles d'oxyde d'uranium, un réservoir de gaz à l'une de ses extrêmités, un piston se déplaçant dans un cylindre à une vitesse prédéterminée à l'autre extrêmité, et un dispositif de mesure de l'écart des pressions entre les deux extrêmités. A cet effet, la mesure de l'effort exercé sur la colonne de pastilles pourrait être utilisée. Le dispositif est à prévoir de telle manière que la longueur du tube, le jeu entre les pastilles et le tube et la nature du gaz puissent être des paramètres, variables, de l'étude.

4. <u>ETAT DE L'ETUDE</u> :

   1. <u>Avancement à ce jour</u> :

Sans objet : à lancer.

2. Résultats essentiels :

Sans objet.

5. PROCHAINES ETAPES :

Sans objet : étude ponctuelle.

6. RELATION AVEC D'AUTRES ETUDES :

Etude complémentaire du programme DEMETER, en vue de l'évaluation de la contrainte exercée sur la gaine pendant le ballonnement.

		CLASSIFICATION: 11.1
TITLE (ORIGINAL LANGUAGE): Creep dello Zircaloy sotto irraggiamento		COUNTRY: Italy SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Irradiation creep of Zircaloy		ORGANISATION: AMN/NUCLITAL PROJECT LEADER: U.V. Rocca
INITIATED: 1977	COMPLETED:	SCIENTISTS: F. Raggi
STATUS: In progress	LAST UPDATING: June 1979	

General Aim:

Some BWR cladding specimens have been tested in Siloé reactor. Creep-down and creep-out in biaxial stress condition and growth are considered, several differential pressures on the cladding are applied at constant temperature and clad deformations as well as cladding ovality are examined versus neutron fluence.

Particular Objectives:

- To obtain further information on in pile creep deformation of a Zircaloy cladding and to perform an in pile creep law analytical development.

Experimental Facilities and Programme:

The specimens are stacked in stainless steel carriers, two for creep out and growth tests and one for creep down tests. Each carrier is introduced into a thermostatic furnace and then into the pool reactor. The furnace used is the standard "Chouca" type O for structural material irradiation in pool reactor (Siloé, Grenoble).
A Nak eutectic alloy, contained in the inner tube of the furnace under Helium atmosphere, ensures uniform temperatures. Cromel-Allumel thermocouples that elastically contact the specimen are used (three for each specimen for creep out and growth device and four for each specimen for creep down device). Six and five fast fluence detectors are used respectively for creep out and creep down tests.

Project Status:

1. Progress to Date:

   Experimental data up to 5000 hours of irradiation time are available. Specimens irradiation and post-irradiation measurements have been completed. A preliminary experimental result evaluation has been performed.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Irradiation creep of Zircaloy	11.1

2. Essential Results:

For the fast flux of $10^{13} \div 10^{14}$ n/cm^2 sec and for the stresses of $70 \div 124$ MN/cm^2 the creep correlation could be of the form $\varepsilon = K(T) \cdot \sigma^{n} \cdot (\phi \cdot t)^{m}$ with n = 1 and m < 1. Creep-out and creep-down behaviour seem differ each other.

Next Steps:

Burst tests will be carried out on both irradiated and unirradiated specimens.

Reference Documents:

1. "Zircaloy-2 in Pile Creep Predicted - Measured Values Comparison" by P.L. Ficara, U. Pazzi, F. Raggi, U.V. Rocca. Paper presented at the enlarged Halden Group Meeting, Loen, Norway, 5th - 9th May 1978.
2. "Comparison of Predicted-to-Measured Values of Zircaloy-2 in Pile Creep" by P.L. Ficara, V. Pazzi, F. Raggi, U.V. Rocca. Energia Nucleare, 10, ott. 1978.

Degree of Availability

This work is carried out in the frame of an agreement between CNEN (Roma) and AMN (Genova) and the experimental results are not open without CNEN and AMN permission. General information is available. (Nuclital, V. G. D'Annunzio 113, I-16121, Genova).

	CLASSIFICATION: 11.1
TITLE (ORIGINAL LANGUAGE): Creep delle guaine di Zircaloy ad alta temperatura fuori pila	COUNTRY: Italy
	SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Out of pile high temperature Zircaloy cladding creep	ORGANISATION: AMN/NUCLITAL/ CNEN/CISE
	PROJECT LEADER: U.V. Rocca
INITIATED: 1977 / COMPLETED:	SCIENTISTS: S. Omarini (CNEN) R. Sozzi (CISE)
STATUS: In progress / LAST UPDATING: June 1979	

General Aim:

Some experimental programs were performed in order to study the BWR Zircaloy-2 cladding behaviour in LOCA conditions. The specimens used for these programs have been tested in steam environment at high temperature and with an applied differential pressure.

Particular Objectives:

The purpose of these experimental programs is to investigate the oxidizing cladding deformation and ballooning under LOCA conditions and to obtain a creep correlation with a plastic instability criterion.

Experimental Facilities and Programme:

The tests are performed using specimens representative of BWR fuel cladding (CAORSO reactor first fuel load and BWR/6 retrofit). The experimental equipments used for these tests can subject the specimens to different temperatures and pressures. A differential pressure is applied on the cladding using an inert-gas. Inside pressure can reach values from 2 to 60 atm; outside pressure is the atmospherical pressure. The heating is obtained by Joule effect. The temperatures reach values from 850 °C to 1200 °C. Steam is introduced at atmospheric pressure and it flows on the specimen outside surface. The strain values are obtained by measurement of cold maximum hoop strain and for some tests also by photografic technique.

Project Status:

1. Progress to Date:
   120 tests were performed at isothermal and isobaric conditions. The experimental results of these tests are available and a preliminary data evaluation has been performed.
   Other 60 tests were performed at transient temperature conditions. The temperature transients for these tests are obtained by constant heating rate ramps (2 and 12° C/s).

45.897

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Out of pile high temperature Zircaloy cladding creep.	11.1

The experimental results are already available. Their evaluation is in progress. 125 samples have been tested in a steam environment at transient temperature and pressure conditions. Some samples have been tested without stress in order to study the oxidizing mechanism in transient conditions.

2. Essential Results.

The obtained results show that the strains in steam environment are lower than those reported in literature for vacuum tests under the same temperature and stress conditions. The experimental data seem confirm that the oxide layer induces a cladding strength hardening and a particular rupture mechanism. Moreover clad deformations have been observed without any differential pressure applied.

Next Steps:

CNEN and CISE are developing a new device to obtain clad temperature variation by internal heating.

Reference Documents:

1. Nuclital NT 77/001, M.S. Lavitola, G. Lippolis, G. Maraniello, R. Potì, V. Pazzi, U.V. Rocca, "Analisi delle correlazioni di creep dello Zircaloy ad elevata temperatura utilizzate in codici di progetto".
2. Nuclital NT 76/007, V. Pazzi, "Programmi sperimentali di interesse Nuclital sul comportamento meccanico dello Zircaloy ad alta temperatura".
3. CISE Divisione Tecnologie, Novembre 1977, "Esperienze fuori pila di creep ad alta temperatura ed in vapore d'acqua su tubi di guaina BWR" (lavoro nell'ambito di un contratto di ricerca NUCLITAL/CISE).
4. Nuclital RT 78/004, M.S. Lavitola, V. Pazzi, U.V. Rocca, "Problematiche dell'ossidazione dello Zircaloy durante l'incidente di LOCA".
5. CISE-78.076, "Esperienze di creep ad alta temperatura su tubi di guaina BWR: qualificazione e certificazione delle prove".

Degree of Availability:

This work is carried out in the frame of an agreement between CNEN (Roma) and AMN (Genova) and the experimental results are not open without CNEN and AMN permission - General information is available. (Nuclital, Via G. D'Annunzio 113, I-16121, Genova).

		CLASSIFICATION: 11.1
TITLE (ORIGINAL LANGUALE): Studio della conduttanza dell'intercapedine combustibile/guaina in barre di combustibile BWR		COUNTRY: ITALY
		SPONSOR: CNEN
TITLE (ENGLISH LANGUALE): Gap conductance in BWR fuel rods		ORGANISATION: CNEN
		PROJECT LEADER: NOBILI
INITIATED: 1968	COMPLETED:	SCIENTISTS:
STATUS: in progress	LAST UPDATING: June '79	

1. General aim

The aim of the experimental program is to study the effect of the different parameters (linear heat rating, cold gap, power cycling, etc.) on the gap conductance.

2. Experimental facilities

The irradiation has been performed utilizing instrumented rigs capable to measure directly the power generated in the fuel as well as the fuel and cladding temperatures (1).
Reactor utilized: Avogadro, Siloè, Essor.

3. The second series of irradiation experiments to study the cold gap effect on gap conductance have been planned in the Siloè and Essor Reactors.
The part of program in Siloè Reactor has been finished, and irradiations in Essor are still in progress.

4. Next step

Extension of the experiments to the study of other significant parameters.

5. Relation to other project

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(10)75.

TITLE (ENGLISH LANGUALE):	CLASSIFICATION:
Gap conductance in BWR fuel rods	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 conducibilitàe conduttanza dell'intercapedine tra combustibile e guaina", CNEN.

6.4. G. Cannata (ENEL), G. Cosoli (CNEN, COMB), "Gap conductance analysis for BWR rods relocation model" specialists' Meeting on "Fuel Element Performance Computer Modelling", Springfields, U.K., March. '78.

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.

(A. Nobili, CNEN, CSN Casaccia, CP 2400, I-00100 Roma)

	CLASSIFICATION: 11.1
TITLE (ORIGINAL LANGUAGE): Comportamento meccanico e chimico di guaine di Zr ad alta temperatura	COUNTRY: ITALY SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): High temperature mechanical and chemical behaviour of cladding tubes	ORGANISATION: CNEN PROJECT LEADER: S. OMARINI

INITIATED: 1975	COMPLETED:	SCIENTISTS:
STATUS: In progress	LAST UPDATING: July 1979	

1. General Aim

   Development of stress strain and oxide growth rate correlations at high temperature for Zircaloy fuel cladding.

2. Particular Objectives

   Stress-strain correlation in function of heating-rate and temperature.
   Axial extent of ballooning.
   Cinetic of oxidation in function of heating-rate and temperature.

3. 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 2-40 ata.

   b) Radiant heat facility for the high temperature Zr oxidation studies.

   c) Internal heater with the same performances of the facility a) and with controlled heated external sheath.

4. Project Status

   About 100 tests carried out on CIRENE and CAORSO LWR. Other tests on PWR are in progress.

   b) All the planned tests are carried out.

5. Next Steps

   Carrying out the facility c). Analytical elaboration of the achieved data. Analytical model on axial ballooning..

6. Relation to Other Projects and Codes

   High temperature tests on irradiated clads (National Laboratories -Risø -Danmark).
   Single pin in pile tests (EOLO) - (CCR Euratom-Ispra).

9.507

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
High temperature mechanical and chemical behaviour of cladding tubes	11.1

7. Reference Documents

"Attività inerente la ricerca sulla cinetica di ossidazione di guaine di Zircaloy ad alta temperatura" - RTI(CECAT-3), Gennaio 1977.

"Valutazione dei coefficienti, programmazione e "fitting" di una relazione di deformazione guaine tratta dal metodo di calcolo CANSWEL" RTI(CECAT-4), Febbraio 1977.

"Valutazioni analitiche e sperimentali sul comportamento ad alta temperatura di guaine di geometria CIRENE" - CECAT/TERM-RAP(78)2, Febbraio 1978.

"Procedure di Prova - Esperienze alta temperatura su guaine BWR" RAP(78)-02/CECAT, Febbraio 1978.

"Prove Preoperazionali - Esperienze ad alta temperatura su guaine BWR" RAP(78)03-CECAT, Marzo 1978.

8. Degree of availability

To a limited extent.

S. Omarini, CNEN, CSN Casaccia, CP 2400, I-00100 Roma.

	CLASSIFICATION: 11.1
TITLE (ORIGINAL LANGUAGE): Analisi del comportamento termomeccanico delle barre di combustibile in condizioni di normale funzionamento e di incidente	COUNTRY: ITALY SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Thermomechanical analysis of fuel rods during normal operations and accident conditions	ORGANISATION: SOPREN PROJECT LEADER: G.P. Pozzi

INITIATED: 1978	COMPLETED: 1982	SCIENTISTS: E. Mascellani, M.S. Lavitola
STATUS: in progress	LAST UPDATING: September 1979	

1. General aim

Development of two computer codes, the first regarding the steady state behaviour of LWR fuel rods during irradiation, and the second regarding the thermomechanical behaviour of the same rods during accident conditions.

2. Particular objectives

Verification (using the developed codes) of experiences performed by Italian CNEN and other international organizations. The aim of this work is the validation of the models contained in the developed codes.

3. Experimental facilities

In cooperation with CNEN an experimental facility for out of pile rod bursting and ballooning of PWR rods has been designed; its realization will be completed in 1979.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Thermomechanical analysis of fuel rods during normal operations and accident conditions	11.1

4. Project status

Using the information available the definition of the various models to be inserted in the two above codes is in progress.
The models cover the following aspects:

a) for the steady state irradiation program: fission gas release; fuel densification; swelling of the Uranium dioxide and of the Zircalloy cladding; pellet-cladding interaction; stress and temperature distribution in the pellet and in the cladding.

b) for the transient accident program: cladding ballooning and bursting during a LOCA; cladding collapsing associated with fuel densification; cladding oxidation and embrittlement; transient DNB, cladding temperature increase and degradation of the cladding mechanical properties; fuel central melting.

In 1979 some separated models were verified against experimental results (in particular the rod failure tests performed in the frame of Studsvik OVER-RAMP Program).

5. Next steps

Are the following ones:

a) codification of the different models in various separated subroutines; debugging of the same

b) formulation and codification of the two digital programs for the steady state and transient fuel rod analysis

c) validation of the above programs against experimental results.

6. Degree of availability

To a limited extent

7. Budget, personnel involved

2 engineers and 2 analysts for 5 years

	CLASSIFICATION: 11.1
TITLE (ORIGINAL LANGUAGE): DEFORMATION OF FUEL ROD CLADDING	COUNTRY: UK
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: NPDL SPRINGFIELDS
	PROJECT LEADER: MR C A MANN
INITIATED: 1971 / COMPLETED:	SCIENTISTS:
STATUS: / LAST UPDATING: MAY 1979	

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 in Zircalloy-clad fuel rods causes deformation which could obstruct further cooling. If this is so, to define the conditions under which it could occur, and to investigate remedial measures.

3. EXPERIMENTAL FACILITIES AND PROGRAMME

SGHWR fuel bundles up to 2m length have been heated in a furnace through defined temperature and pressure sequences. The bundles were then dismantled and the distortions of the fuel rods were measured.

Short lengths of LWR fuel rod are taken through defined sequences of temperature and internal pressure, and their deformations measured.

4. PROJECT STATUS

The deformation of LWR fuel rod cladding in steam atmosphere has been studied over a range of internal pressures and over the range of temperatures calculated to occur after a loss of coolant accident. In some combinations of pressure and temperature extended deformation is seen with strains of 30% and more. Investigation of the mechanisms of deformation is proceeding. One of the controlling parameters is the heat transfer at the surface of the cladding. A programme of tests has been carried out on single rods in flowing steam. This will be extended to multi-rod assemblies to study interactive effects.

REFERENCE DOCUMENTS

SNI Reports SNI/1/14 October 1973
Mann C A The Behaviour of SGHWR Fuel Elements Under Accident Conditions (CSNI Specialists' Meeting, Spåtind, Norway, 13-16 September 1976)
Rose K M, Mann C A and Hindle E D The Axial Distribution of Deformation in the Cladding of PWR Fuel Rods in a Loss-of-Coolant Accident (BNES/ANS Topical Meeting on Nuclear Power Reactor Safety, Brussels, October 1978).

	CLASSIFICATION: 11.1
TITLE (ORIGINAL LANGUAGE): MODELLING OF FUEL ELEMENT BEHAVIOUR DURING TRANSIETNS	COUNTRY: UK
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA
	PROJECT LEADER: J H GITTUS
INITIATED: 1976 / COMPLETED:	SCIENTISTS: P M JONES, T J HASTE
STATUS: CONTINUING / LAST UPDATING:	

BACKGROUND

During a loss-of-coolant accident or other transient the combination of a decrease in clad strength and an increase in differential gas pressure may cause the cladding to swell and partially block the coolant channel. This job is concerned primarily with the theoretical modelling of the factors controlling the extent of such a blockage.

OBJECTIVE

Transient codes will be developed and compared with other existing codes to give improved models of clad deformations. Particular attention will be given to the prediction of strain distribution during clad ballooning, rod/rod interaction and gap conductance modelling. The codes will be assessed against available irradiation experiments and laboratory experiments.

PRESENT POSITION

A one-dimensional clad deformation model CANSWEL 1 has been developed and integrated into the MABEL 1 rod model. CANSWEL 1 contains models for the creep of Zircaloy in the alpha, beta and mixed phase regions and includes the effect of oxidation. Current development concerns the extension to two dimensions to account for azimuthal temperature variations and the inclusion of a model of the mechanical restraint due to rod/rod contract. The resulting CANSWEL 2 will be the deformation package in the MABEL 2 two-dimensional node code.

	CLASSIFICATION: 11.1
TITLE (ORIGINAL LANGUAGE): Fuel-cladding interface conductance (FS 6.1)	COUNTRY: UNITED KINGDOM SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Harwell) PROJECT LEADER: D O Pickman (SNL) J B Sayers (Harwell)

INITIATED:	COMPLETED:	SCIENTISTS:
March 1978		
STATUS: In progress	LAST UPDATING: March 1979	

Background

Fuel rod behaviour in LOCA is sensitive to the fuel temperature and rod internal pressure just prior to the event. Current code predictions suffer from the difficulty in estimating the pellet to cladding conductance.

Objectives

To modify the MINIPAT/SLEUTH code and verify by reference to instrumented pin experimental irradiations carried out in the DIDO reactor at Harwell.

Programme

The first experiment (IE 1124/1) has been completed. Analysis of the results will examine the effective gap width (the term $G + Y_{min}$ in the conductance equation) for the case with nominal contact at full power and a range of helium - fission gas mixtures. The second (IE 1124/2) will make similar temperature measurements with a range of fuel - cladding gaps and gas composition.

Facilities

The rig SD 1124/2 rig in the DIDO reactor at AERE

Reference Documents

	CLASSIFICATION: 11.1
TITLE (ORIGINAL LANGUAGE): Centre UO_2 temperature measurement (FS 7.1)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Harwell)
	PROJECT LEADER: D O Pickman (SNL) J B Sayers (Harwell)
INITIATED: March 1978 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: March 1979	

Background

Fuel centre temperatures in water reactor fuel pins may change substantially in the early stages of irradiation as a result of the balance between densification, pellet fragment relocation, and fission gas bubble swelling. The magnitude of the effect depends on fuel rating, UO_2 density and pore geometry, pellet design and fabrication route.

Objectives

1. To determine the effect of fuel fabrication route, initial density and fuel-clad gap size on UO_2 centre fuel temperatures.

2. To use MINIPAT/SLEUTH codes to rationalise the experimental data with current code predictions.

Programme

Three experiments completed by April 1979. Four experiment follow.

Facilities

High pressure water loop in DIDO Harwell.

Reference Documents

		Classification 1.2, 11.1
Title 1	Thermohydraulic Safety Studies Heater Rod Cluster Rig	Country U.K.
Title 2	Effects of Fuel Pin Ballooning on Reflood Heat Transfer	Sponsor Organisation CEGB
Initiated 1.6.78 Status Continuing	Completed Last updating	Project Leaders S.J. Board S.A. Fairbairn

Berichtszeitraum/Period 1.1.78 - 31.12.78	Klassifikation/Classification 11.2	Kennzeichen/Project Number RS 245
Vorhaben/Project Title Großbehälter - Phase 1 Full Size Vessel - Phase 1		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Staatliche Material-prüfungsanstalt (MPA) Stuttgart
Arbeitsbeginn/Initiated 1.10.1976	Arbeitsende/Completed 31.12.1979	Leiter des Vorhabens/Project Leader Dr. Sturm/Doll
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 31.12.1978	Bewilligte Mittel/Funds 16.530.922,--

1. General Aim

Elaboration of a rough specification for the research programme "Full Size Vessel" as well as purchase and transport of a boiling water reactor pressure vessel. Fabrication of a part of a shell made of material in boundary condition.

2. Particular Objectives

The rough specification shall contain the following urgent examinations to be performed on the full size vessel:

- nondestructive examinations (flaw detection probability) detection of the zero-condition
- removal of trepan to verify probable defects
- internal pressure test

Construction and montage of BWR-pressure vessel:

- inserting of defects in the site weld seams
- construction and insertion of a weakened section from material in boundary condition. Welding of circular and length seam welds with man-made and natural defects
- non-destructive testing

3. Research programme

-

4. Experimental Facilities

To perform automatical non-destructive testings a central mast manipulator must be developped and constructed.

5. Progress to Date

Rough specification has been peformed in collaboration with the Institut fuer zerstörungsfreie Pruefungen (Izfp), Saarbruecken. Completion of the testing cell.

Completion of the frame, transport to MPA Stuttgart and montage of the two parts in the testing cell. (Fig. 1)

Completion of the vessel sections at Breda S-p.A., Milan (acceptance in May 1978) Transport of the full size vessel (Fig. 2,3,4) from Milan to the MPA Stuttgart (June - July 1978)

The site weld seams M1 and M2 are ready except the final annealing.

The site weld seams M3 is intended as a narrow split weld (Fig. 5). For the determination of the welding parameters a preliminary test weld has been performed.

Weakened section of the vessel:
All circular and length weld seams as well as the nozzles R and S have been welded (Fig. 6)

In the circular weld seams RN 4/1 as well as in the length weld seams LN 4/3, 4/4 and 4/7 natural and man-made defect's have been inserted.

6. Results

-

7. Next Steps

- Completion of section 4 at Kloeckner - AG, Osnabrueck and transport to MPA (march 79)
- Completion of circular site weld seams M3 and M4
- final annealing of the site weld seams
- grinding of the cladding

8. Relation with other Projects

RS 304 research programme for integrity of components

9. References

-

10. Degree of Availability

-

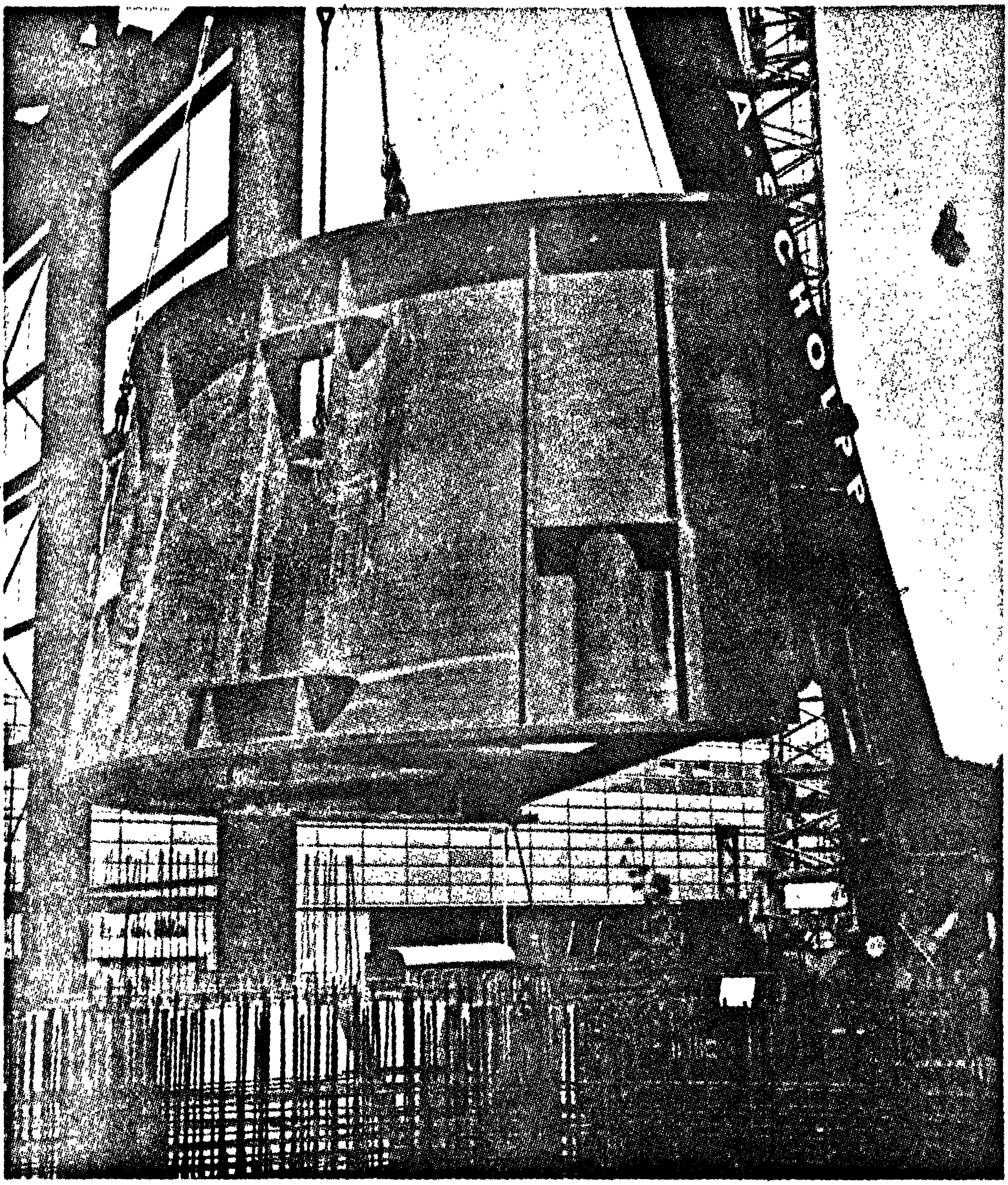

Fig. 1: Montáge of the frame (two parts)

Fig. 2: Sections of the Full Size Vessel at the transshipment in the port of Heilbronn

Fig. 3: Arrival of the Full Size Vessel in Stuttgart (section 2)

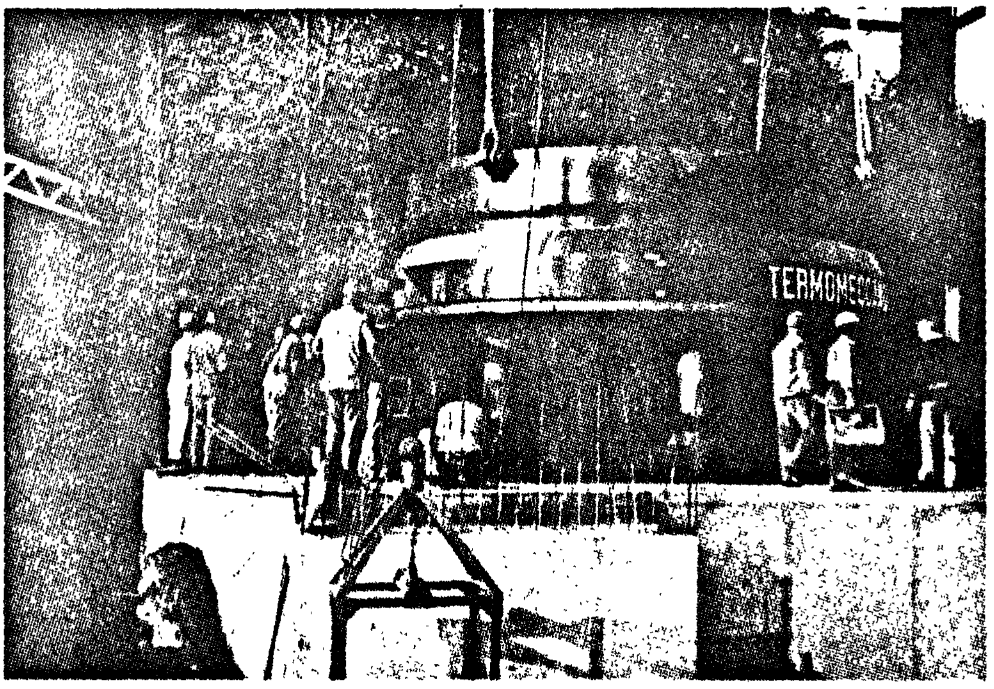

Fig. 4: Bottom head assembly (section 1) lowerment in the testing cell

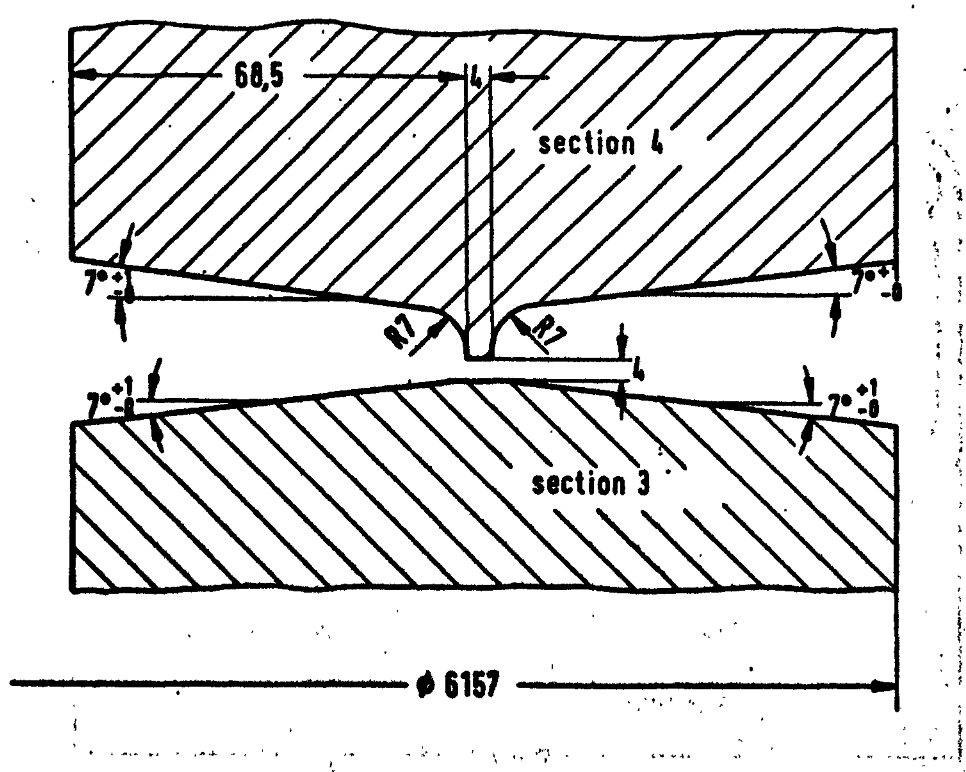

Fig. 5: Site weld seam M3 (narrow split weld seam)

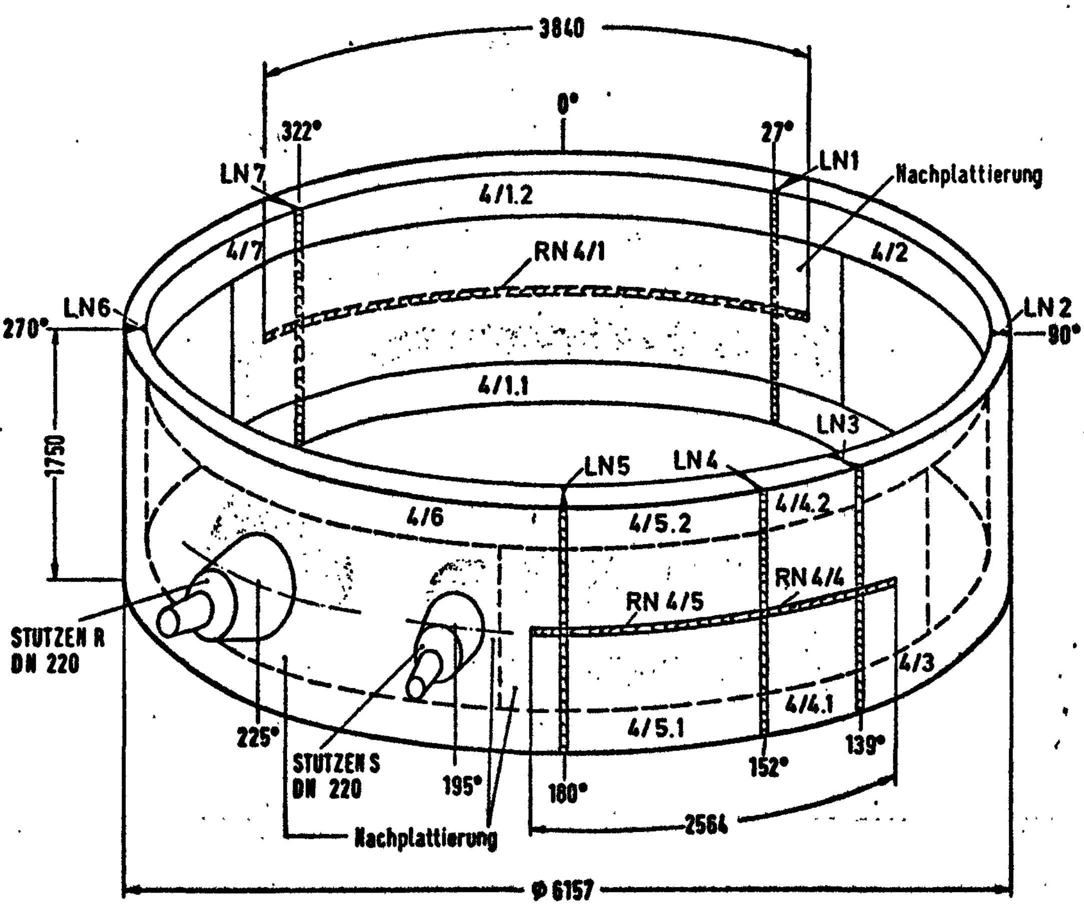

Fig. 6: Weakened section of the Full Size Vessel (section 4)

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.1978-31.12.1978	11.2	RS 213

Vorhaben/Project Title	Land/Country
Entwicklung und Erprobung von Verfahren zum mechanisierten Schweißen von kugelförmigen Reaktorsicherheitsbehältern. Development and Testing of Mechanized Welding Methods for Spherical Reactor Containment.	FRG **Fördernde Institution/Sponsor** BMFT **Auftragnehmer/Contractor** Salzgitter AG

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.02.1978	31.10.1981	Deipenau
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

The commissioning of nuclear power stations is to be expedited using mechanized welding processes rather than conventional methods.

2. Particular Objectives

This project aims to develop mechanized welding processes in order to achieve an improvement of the technological quality coefficient of welds, increase their resistance to porosity and cracking and create a process independent of the human factor.

2.1 Pre-assembly

As part of the pre-assembly of reactor containment shells, three segment plates are to be simultaneously welded together to form a triple plate unit.

2.2 Welding of circumferential seams

Welding of containment circumferential seams is to be tested using tandem submerged-arc welders.

2.3 Welding of vertical seams

For welding of vertical seams, the use of electroslag and electrogas welders is to be tested and modified.

2.4 Welding of upper and lower sphere zones

For welding of the upper and lower sphere zones, inert-gas shielded arc welders with automatic weld centerline guidance are to be tested and modified.

3. Research Program

3.1 Pre-assembly

3.2 Welding of circumferential seams

3.3 Welding of vertical seams

3.4 Welding of upper and lower sphere zones.

4. Experimental Facilities

re 3.1 Test stand with tandem submerged-arc welder including additional gas-shielded-arc welding head for root welding. The equipment is controlled from a central control unit with speed regulation and synchronization of two travelling gears. There is a tilting table with vacuum holding-down appliance for receiving the three plate segments and control of the inclination adjustment for site tests.

re 3.2 The test stand includes a rail for guidance of the welder, a flux retaining strip, a tandem submerged-arc welder with follow-up device and controlled travelling gear speed.

re 3.3 & 3.4 The test stand will be installed after completion of equipment survey and ordering (scheduled for July 1979).

5. Progress to Date

re 3.1 The laboratory welders are now available and are being modified. Design work for the tilting table control is in progress. Laboratory welding tests on boiler plates have been completed. Laboratory welding tests on 15 MnNi 63 material have been initiated.

re 3.2 Design work for flux retainer and guide rails is in its final stage. The laboratory welding tests on 15 MnNi 63 material have been initiated.

re 3.3 & 3.4 Survey of equipment available in Japan has been completed. Survey of European equipment is in its final stage.

6. Results

re 3.1 & 3.2 Noteworthy results are not yet available.

re 3.3 & 3.4 Representative results of the equipment survey are not yet available.

7. Next Steps

re 3.1 Welding and testing of 15 MnNi 63 test plates. Manufacture of tilting table. Modification of welding and control equipment.

re 3.2 Welding and testing of 15 MnNi 63 test plates. Completion of the design work for flux retainer and rails. Modification of the welding and control equipment.

re 3.3 & 3.4 Preparation and completion of equipment survey in the 1st quarter of 1979.

8. Relation with other Projects

-

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1978 - 31.12.1978	11.2	RS 304 A

Vorhaben/Project Title	
Forschungsvorhaben Komponentensicherheit, Programmjahr 1978 Research Programme Integrity of Components	Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Staatliche MPA Universität Stuttgart Pfaffenwaldring 32 7000 Stuttgart 80

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.1978	31.12.1978	Dr. Issler/ Dr. Föhl
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

The research programme integrity of components (FKS) is concerned with the reactor pressure vessel of nuclear light water reactors, with the focus point of the long time behavior under special consideration of unfavorable material flaw and stress state.

The programme year 1978, which followed the phase of specification sept.-dec. 1977 had the aim to purchase the specified materials and to prepare them for the subsequent tests. In the frame of the specified aim of the FKS it was necessary to specify and realize the materials in the various toughness levels including the "low shelf"-behavior as well as flaw conditions relevant for the safety aspects. In first investigations at these materials the structure- and flaw state and the basic material properties were characterized.

The aim of the theoretical tasks was the critical judges of the calculation and test procedures being applied and the creation of transferability criteria specimen/component.

2. Particular Objectives

The particular objectives of the programme year 1978 and the investigation under chapter 3 derived herefrom are arranged as follows

2.1 Provision of RPV-materials in real ingot sizes with various toughness levels and flaw conditions.

2.2 Modifications of these materials by suitable choice of production and processing procedures.

2.3 Modification of these materials by in-service conditions.

2.4 Description of material state and material behavior under the influence of production- and in-service parameters.

2.5 Creation of transferability criteria specimen/component.

3. Research Programme

Corresponding to the particular objectives the following means were performed

3.1 Melting of 200 t ingot of material 22 NiMoCr 37 (A 508 B Cl 1) and 22 MnMoNi 55 (A 533 Cl 2) by conventional and MHKW-procedure. (Electro slag remelting).

3.2 Reforging of the ingots with various degrees of deformation and directions. Welding- and heat treatment of plates with production like and "lower-bound" parameters.

3.3
- Long time aging of specimens and vessel parts
- Irradiation experiments in power reactors
- Preparation of irradiation experiments in research reactors
- Preparation of tests in static autoclaves for examination of environmental influences

3.4
- Nondestructive examinations of base material (slaps, plates, specimens) and weld joints.
- Metallographic investigations of base material and weld joints.
- Destructive testing to determine the acceptance data with small scale specimens.
- Preparation of large scale specimens and intermediate size vessel tests

3.5
- Instrumented Charpy impact tests and drop weight tests conducted in different laboratories (Round-Robin-Test)
- Finite element calculations for large scale specimens, nozzle corners, fracture mechanics specimens and determination of residual stresses of weldings at model test pieces.
- Advancement of test and measuring techniques for heat affected zone simulation, dynamic fracture phenomenon and elastic-plastic fracture behavior.
- Calculation of damage functions and neutron spectra.

4. Experimental Facilities, Computer Programme
see annual report 1977

5. Progress to Date
The means mentioned under chapter 3 for procurement and processing of the materials mainly have been finished, furthermore a characterization of the materials by nondestructive, metallogical and mechanical testing has been conducted.
Finite element- and transferability calculations as well as the preparation and verification of test and measuring techniques to be applied have been executed.
The first two irradiation capsules have been completed and will be inserted in the power reactor in January 1979.

6. Results
By nondestructive examination of the base materials of present forgings (flange ring, steam-generator plate) the flaw state necessary for the aim of the programme was determined as specified.
The location of the specimens to be tested(large scale specimens, welded in pieces for intermediate size vessels) was chosen in accordance with the flaw state. One heat specified as "above limit" regarding toughness and crack formation has shown low toughness but no significant cracks.
In consideration of the results of a test ingot (20 t) with natural flaws a new ingot with changed chemical analysis has been produced and is now being examined. By welding with unfavorable parameters (preheat temperature, build up sequence, geometry a.o.) welded joints in full component thickness with partly considerable cracks could be produced, the behavior of which will be investigated in large scale specimens.
Based on the first results of the FE-calculations nozzle geometries for intermediate size vessels and notch shapes for large scale specimens were established. One dimensional transport calculations for determination of neutron spectra will be used for design and dimensioning (shield arrangement and material) of irradiation facilities in research reactors.

7. Next Steps
Near the end of the programme year a considerable delay in the processing of some materials was ascertained, so that the research programme planned for 1978 could not be settled completely. These tasks will be continued in the first few months of the next year parallel to the activities planned for 1979.

In the programme year 1979 the experimental work will be supplemented by theoretical considerations furthermore extensive tests with the first large scale specimens will be conducted.
The intermediate size vessels will be completed for testing in 1980. The irradiation experiments initiated in 1978 in research and power reactors as well as corrosion tests in autoclaves will be continued during the programme year 1979.

8. Relation with other Projects

see annual report 1977

9. References

Annual reports related to the particular materials and special fields are in progress.

10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.7.78 - 31.12.78	11.2	GKSS 241

Vorhaben/Project Title	
Bestrahlungen im Rahmen des Forschungsprogramms Komponentensicherheit von Reaktordruckbehälterstählen Irradiation Tests in the Framework of the Research Program Component Safety	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: GKSS, Geesthacht Institut für Werkstofftechnologie und Chemie sowie Institut für Physik

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.7.78	31.12.83	Dr. Schmitt/Dr. Wille
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General aim

The research program is emphasized on sounding out safety margins of the primary pressure components taking into account postulated states of material on extreme conditions, unfavourable states of flaws and stress as well as influences of operation.

2. Particular objectives

Pressure vessels of light water reactors will change their material properties by neutron irradiation. Ductility properties required for safety will be deteriorated especially. Those depend essentially on material composition, structural condition and state of stress and are influenced by neutron fluence, neutron spectrum and temperature during irradiation as well. There are two particular objectives:

2.1 Irradiation of selected pressure vessel steel melts and welding material in advance on operational conditions of light water reactors and subsequent tests by various methods in order to determine changes of material properties quantitively.

2.2 Investigation and physical interpretation of grating defects in iron-alloys caused by irradiation aiming at a clarification of the influence of alloy composition, heat treatment, mechanical load and irradiation at which an exact understanding of the microscopic effects should be achieved.

3. Research program

3.1 Development, design and construction of large sized irradiation capsules, operating and supply equipment, devices for measuring and accessories.

3.2 Investigation on the following material properties: strength, irradiation embrittlement behaviour and ductility dependent on material, state of material and the parameters neutron fluence and spectrum as well as irradiation temperature.

Furthermore, corrosion attack by the primary demineralized water, irradiation being present, will be investigated in order to simulate flaws in a welding plate.
The following types of specimen will be applied: Tensile strength notched-bar impact bending, impact-testing according to Pellini and fracture mechanics. Three states of material will be used: Basic material, weld metal and heat influenced weld zones of melts consisting of the steels 22 Ni Mo Cr 37 and 20 Mn Mo Ni 55.

Parameters indicating influence, especially neutron fluence and neutron spectrum of the research reactor FRG 2 will be determined experimentally and theoretically for transferring the results to light water power reactor conditions.

3.3 Investigation on the microstructure of steel specimens of which the following will be used for experim ental analysis:

- Pressure vessel steel specimens.
- Alloy composed of pure iron and copper.
- Pure alloy of iron and other additional material.

3.4 Material technological investigations on calculating stress of fracture mechanical specimens, on yield fracture mechanics, on effects of the microstructure, temoerature and surface during fracturing and on the mechanism of damage by irradiation embrittlement in metal gratings.

Main part of the program will be carried out in context of the research program "Komponentensicherheit" (FKS) and the second IAEO-program.

4. Experimental Facilities, Computer Codes

- Research reactor FRG 1 and 2
- Hot Cells (Concrete and lead cells)

- Irradiation capsules of the type A5, W3 and CK (large container capsule), supply equipment and measuring devices included
- Tensile strength testing-machines up to 2500 KN
- Instrumented impact testing-machine according to Pellini
- Hardness testing-machines
- Electron scan microscope
- Metallographical laboratory
- neutron beam small scattering angle facility at the research reactor FRG1*
- Special evaluation methods

* An accurately collimated, monochromated neutron beam is focussed on a specimen being in a magnetic field directed transverse to the beam. There are two detectors reproducing minute local details located in a plane some meters far away which take up two profiles of the scattering intensity parallel and perpendicular to the direction of the magnetic field.

5. Progress to Date

Ref. to 3.1 A special large container capsule being able to take up large fracture mechanic as well as impact Pellini besides conventional small tensile strength and notched-bar impact specimens has been developed. A prototyp capsule were being operated and tested sufficiently. Some of a set of container capsules and their auxiliary systems are going to be manufactured.

Ref. to 3.2 Charpy - v - notch impact specimens of the steels 22 Ni Mo Cr 37 and 20 Mn Mo Ni 55 (basic material and welded zone) were neutron irradiated in the research reactor FRG 2 at temperatures 290 °C up to a fluence of $2 \cdot 10^{19}$ n/cm^2, E > 1 MeV in order to study the influence of the irradiation temperature.

The use of an iron shielding should give an answer to the influence of the neutron spectrum on the impact toughness properties.

Charpy - v - notch impact specimens were also irradiated in the power reactor of Stade (PWR) for comparison and to find out a possible

influence of the different neutron spectrum.
The test material was machined from a block of the HSST-Plate 03, A 533 B. The fluences were 2,0 · 10^{19} and 1,6 · 10^{19} n/cm^2, respectively. The irradiation temperature was 290 °C.

Charpy - v - notch specimens of the welded zone (22 Ni Mo Cr 37) which were irradiated to a fluence of 9 · 10^{18} n/cm^2 in the Power Reactor of Stade (PWR), were tested in the Hot Cells in Geesthacht. These tests are a part of the Reactor Pressure Vessel surveillance program.

Ref. to 3.3 Work was concentrated on evaluation of neutron beam small scattering angle measurements parallel and perpendicular to a strong magnetic field so that nuclear and magnetic scattering were yielding a more complete interpretation.

Ref. to 3.4 Further development of a computer code for experimental fracture mechanics as well as considerations on a theoretical determination of fracture and irradiation influenced occurrences in the microstructure will be continued.

6. Results

Ref. to 3.1 After development, construction and a prototype test the first large container capsules including auxiliary systems have been prepared.

Ref. to 3.2 As expected the shift of the NDT-Temperature for both steels is smaller at the higher irradiation temperature of 320 °C than the shift at the lower temperature. On the other hand results up to now show influence of an additional shielding on the NDT-temperature is negligible.

As to the comparison between power and research reactor the impact-temperature curves of the two irradiation tests show that there are no greater differences in the NDT-temperature at fluences up to 2 · $10^{19}$$n/cm^2$ in the two reactors, although the irradition time was 6014 hours in the Power Reactor and 350 hours in the research reactor FRG 2.

The impact-temperature-diagramm of the PWR-Test shows a decrease of the upper shelf energy which has not been observed in the research reactor tests.

The evaluation of the Charpy - impact specimens with regard to the surveillance program which had been irradiated in the power reactor of Stade (PWR) showed the expected shift of the NDT-temperature.

Ref. to 3.3 and 3.4 Results of investigations on irradiation defects in iron alloys achieved by microstructure measurements in the neutron beam small scattering angle facility are as follows:

A. Pure alloy of iron and copper

The evaluation shows that the greater part of copper of the specimen is present in a still fine-dispersive state. There is a small amount of separations of copper particles larger than 20 Å in specimens being cooled down slowly from the melt (up to 25 p.c. of copper ingredients). Separations in specimens being cooled down fast have not been proved. On the other hand, specimens being cooled down slowly and irradiated by a fast neutron fluence of $2 \cdot 10^{19}$ n/cm^2 and temperature of 290 ^{o}C showed separation of about 80 p.c. to 100 p.c. of the present copper in a state of an agglomerated cake with a diameter in the order of 50 Å. On an average, smaller clusters having a smaller total volume of copper were originated in specimens being cooled down fast.

Furthermore, it was proved that the separation effect activated at the temperature of 290 ^{o}C is only taking place to a less extent. This means the limit of solubility of copper in an iron alloy is close to nil at a temperature of 290 ^{o}C, since a significant effect has still been proved at a copper content of 0.1 p.c., too. Separation is accelerated by irradiation obviously.

Processes of nucleus formation play a part eventually, too. Separation of copper during irradiation is the most obvious effect. Besides there are clear symptoms pointing at the formation of dislocation loops, and that especially in the specimens being cooled down fast.

Both effects can be distinguished by the different azimuthal dependence of the scattering, however in principle by applying different strong magnetic fields, too.

B. Specimens of steel

Pressure vessel steel is already containing many separated ingredients resulting in a strong small scattering angle, but mostly meeting an angle range below 2^{o} at a wave length of about 5 Å. Thus, a clear effect in irradiation damaged specimens can be seen in the case of greater angles, and that every time after experience up to now, when mechanical tests have shown an essential embrittlement effect. Only steel specimens with a content of copper above 0.1 p.c. indicate essential effects at an irradiation level of $2 \cdot 10^{19}$ fast neutron per cm^2. However, scattering is stronger than indicated in the pure iron specimen with the same content of copper. Consequently, other ingredients of the steel alloy are supposed to play their part, although copper has a triggering function.

7. Next steps

- Irradiation runs of notched-bar impact bending specimens at temperature between 260^{o} and $270^{o}C$ will be prepared.
- Specimens having been irradiated in large container capsules (tensile strength-, notched-bar impact bending-, Pellini- and fracture mechanics-) will be taken from in the Hot Cells and investigated subsequently.
- Preparations for the Research Program "Komponentensicherheit" will be continued. The Round-Robin-Test for Pellini specimens will be set about.
- Further development of dosimetrie measuring and evaluation methods.
- Continuation of measurements by neutron optical methods for the investigation on the microstructure of iron specimen differently pretreated and those of pressure vessel steel.
- Continuation of theoretical work on fracture mechanical problems

8. Relation to other projects

RS 304 A

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	11.2	RS 263/8

Vorhaben/Project Title	Land/Country; Fördernde Institution/Sponsor; Auftragnehmer/Contractor
Analytische Tätigkeiten der GRS im Rahmen des Reaktorsicherheitsforschungsprogramms des BMFT Behälterversagen Analytical activities of the GRS in the frame of the BMFT research program on reactor safety Pressure Vessel Failure	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Gesellschaft für Reaktorsicherheit (GRS) mbH

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

1. General Aim

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

2. Particular Objectives

Calculation of fluid structure interaction.

3. Research Program

During revision of the working program the research goals have been checked and redefined in some points. In addition the pressure vessel failure tests, which are carried out by the Materialprüfungsanstalt (MPA), Stuttgart, will be taken into consideration in the future. These tests will be assisted by calculations concerning with special problems.

3.1 Design calculations of the pressure vessel failure tests of the MPA.

3.2 Further development of the DAISY Code for application in the pressure failure tests.

3.3 Verification of DAISY using pressure vessel failure tests of the BASF.

3.4 Evaluation and post-test calculations of the MPA tests.

3.5 Comparison with other codes treating fluid structure interaction.

4. Experimental Facilities, Computer Codes

Fluiddynamics: DAPSY - Simulation of multidimensional pressure wave propagation.

Structure dynamics: SAP VI, ASKA - Finite Codes which are used according to their special advantages. KBOS - Code for radial symmetric problems.
Fluid structure Interaction: DAISY - Coupled code for calculations with fluid structure interaction. DAISY consists of DAPSY, some coupling modules and a structure dynamic module, which treats the mass and stiffness matrices from the structure dynamic code.
Containment: COFLOW /1/ - Calculation of pressure distribution in the containment during a LOCA.

5. Progress to Date

The DAISY subroutines for data exchange between the fluid and the structure dynamic calculations have been checked and improved, so that the core barrel wall and the annulus of the BASF tests are represented correctly. Furthermore, several modifications for the multidimensional simulation by means of a network have been taken from DAPSY and implemented in DAISY.
In order to prepare the MPA pressure vessel failure tests, calculations concerning some special problems have been conducted. For the design of a force measuring equipment for very high loadings structure dynamic calculations with SAP IV were necessary. The required number of force cells, plate thickness (to avoid non-linear effect) and the expected error band were estimated by these calculations.
Other design calculations were carried out to determine the maximal discharge rate and the highest containment pressure during the tests. The loading function on the pipes of the test loops during the blowdown following the vessel failure were also calculated.

6. Results

The accompanying work for the pressure vessel failure tests and the calculation results are documented in several technical notices /2, 3/, which were delivered to the MPA.

8. Relation to other Projects

The technique which is used for calculations of fluid structure interaction, has been checked by post-test calculations of a RS 16 test and will also be used for HDR blowdown tests.

9. References

/1/ G. Hellings, F. Mansfeld: COFLOW - Ein Rechenmodell zur Ermittlung des instationären Druckaufbaus in Volldrucksicherheitsbehältern wassergekühlter Kernreaktoren, Programmbeschreibung. GRS-A-254, Dezember 1978

/2/ T. Grillenberger, G. Hellings: Abschätzung der Belastungen beim Behälterbersten. Technische Notiz, 9.8.1978

/3/ B. Riegel, T. Grillenberger: Abschätzung der Belastungen in den Rohrleitungen beim Behälterbersten. Technische Notiz 20.10.1978

10. Degree of Availability of the Reports

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

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	11.2.	RS 279
Vorhaben/Project Title Phänomenologische Behälterberstversuche Phenomenological Pressure Vessel Burst Experiments		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Materialprüfungsanstalt Stuttgart (MPA)
Arbeitsbeginn/Initiated 1.9.77	Arbeitsende/Completed 31.8.80	Leiter des Vorhabens/Project Leader Kussmaul/Sturm
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Determination and quantitative conceiving of the effects during fast failure of unprotected pressure vessels under service conditions (pressure up to 170 bar, temperature up to 350°C).

2. Particular Objectives

30 pressure vessel burst experiments are carried out with vessels (outer diameter 790 mm, wall thickness 45 mm, length 2500/5000 mm) with artificial notches under service conditions, figure 1. Some of the vessels are of a material (20 MnMoNi 55) with a high upper shelf (> 150 J/cm², 300°C) and some of a material (similar to 22 NiMoCr 37) with a low upper shelf (< 50 J/cm², 300°C) toughness. The artificial notches are milled circumferential, longitudinal and crosswise with different depths and lengths.

The main points of the investigation are:

- burst behavior and fracture shape
- crack velocity and arrest
- pressure
- temperature
- strain

$\left.\begin{array}{l}\text{- pressure}\\ \text{- temperature}\\ \text{- strain}\end{array}\right\}$ distribution and progress

3. Research Program

3.1. Material investigations

Performance of mechanical-technological and fracture mechanical investigations

(Specimen thickness up to component thickness).
Selection of materials with a high upper shelf (> 150 J/cm^2, 300°C) respectively low upper shelf (< 50 J/cm^2, 300°C) toughness.

3.2. Stress analysis

Precomputing to determinate the instrumentation for experimental stress analysis.
Computing of strains, displacements and stresses with finite element programs.
Analysis of the results from burst tests with support of fracture mechanics concepts.

3.3. Failure analysis

Performance of burst tests with pipes and cylindric vessels with artificial notches (circumferential, longitudinal and crosswise) and different material toughness.
Burst experiments under service conditions.
Recording the transient response of pressure, temperature, strain, acceleration, crack opening displacement, crack velocity.
Investigation of burst behavior with the aid of high speed photography.

3.4. Analysing the measuring methods

Performance of pretests with pipes and specimens under dynamic loading and high temperature (320°C).
Calibration of the measuring system.

4. Experimental Facilities, Computer Codes

4.1. Pretests

The laboratory equipment faciliates burst tests with pipes (outer diameter x wall thickness = 88,9 x 8,8 ; 101,6 x 10 ; 139,7 x 12,5 mm) and spherical vessels (1000 x 55 mm) up to 2000 bar.
The temperature range is between -180°C and 400°C and the maximum heat power amounts 78 kW.

4.2. Main tests

The test circuit is been built up at the Großkraftwerk Mannheim, where the burst experiments with the vessels shall be performed, figure 2. The circuit has the following data:

- rate of temperature rise 50°C/h
- rate of pressure rise 200 bar/h
- flow rate 275 m^3/h
- maximum pressure 250 bar
- maximum temperature 350°C

4.3. Computer Codes

Computations are performed:

- stress in the notched vessel wall
- strain behavior, displacement of the test vessel
- axial and radial pressure distribution in the test vessel after failure occured
- response forces of the vessel
- pressure in the testing building

To compute the fluid dynamic process the code DAPSY from GRS is used.

Stress, strain and stress intensity factor in the wall of the vessel is computed by FE-codes.

4.4. Plotting methods

Special codes for analysing the measuring results are assembled

5. Progress to Date

Top 3.1. The material 20 MnMoNi 55 was selected for the vessels with high upper shelf toughness. Special heat treatments with the same material should produce also a low upper shelf toughness, but should be without influence on yield stress and tensile strength. Therefore the following heat treatment tests with specimens of 20 MnMoNi 55 were performed:

annealing	1050°C/0,5 h	
tempering	630°C/1,5 h	air cooling
	640°C/1,5 h	
	680°C/1,5 h	
annealing	920°C/0,75 h	
tempering	620°C/1,5 h	air cooling
	660°C/1,5 h	

Top 3.2. The stress intensity factor was computed for circumferential notches in pipes (outer diameter x wall thickness = 88,9 x 8,8; 298 x 22,2; 790 x 45 mm) in relationship to the ratio of crack depth and wall thickness.

Top 3.4. The adaption of the measuring equipment to the special request of the experiments is still in work. Piezo electrical manometers with different constructions, high temperature strain gauges and thermocouples were tested under service conditions. Therefore a water-filled spherical vessel with an outer diameter about 1000 mm and a wall thickness about 50 mm was heated up to 300°C and pressurized at 170 bar. The fast depressurizing took place through a blow nozzle, which was opened by ignition of explosives. The datas were recorded by PCM and simultaneous some important channels by FM. So it was possible, to verify the scan frequency of the PCM recording. The method for measuring crack velocity was tested with burst experiments on pipes with outer diameters about 300 and 400 mm, length 2000 to 6000 mm. The medium was air and the temperature about 20°C.

Top 4.2. The conditions for the test loop and the test room were fixed and the planning performed. At present the construction work is being carried out.

Top 4.3. With the computer code DAPSY from GRS the pressure progress in the testing room after failure of the vessel and the response forces on the piping system was computed.

6. Results

Top 3.1. The result of the tests was, that under no conditions of heat treatment the present charge of material 20 MnMoNi 55 the toughness in the upper shelf is below 85 J/cm^2 (300°C). But a material could be found on the base of 22 NiMoCr 37 with modified chemical analysis, of which the toughness was lower than 50 J at a temperature of 300°C and the ultimate stress values were between the waranty of the material 22 NiMoCr 37. This material is now used for the fabrication of the test vessels with low upper shelf toughness.

Top 3.2. The results of the computation of the stress intensity factor (pipes with circumferential notches) are in hand.

Top 3.4. The method of measuring crack velocity is shown in figure 3 and 4. The splitting wires are bearing a strain about 8% . This value is sufficient for the provided tests. Air pressurized burst experiments with pipes about 300 and 400 mm diameter have shown, that crack velocity (in this case about 200 m/s) can be certain measured.

Top 4.2. It was computed that maximum average pressure in the testing room will rise about 0,22 bar. It is possible that there are oscillations with $\pm$ 0,06 bar about the average pressure in the severest burst tests. The response forces on the piping system will be up to 10^5 N. This datas were taken into consideration in constructing work.

7. Next Steps

Another pretests were carried out to prove the transducers measuring system, the measuring codes as well as the testing loop. Material constants with test specimens from the original vessel material were investigated. The dimensions of the longitudinal notches were computed. Burst experiments with the testing vessels (material high upper shelf toughness, longitudinal notches) were performed. Computations of the stress intensity factor of longitudinal and crosswise notches were carried out.

8. Relations with other projects –

9. References –

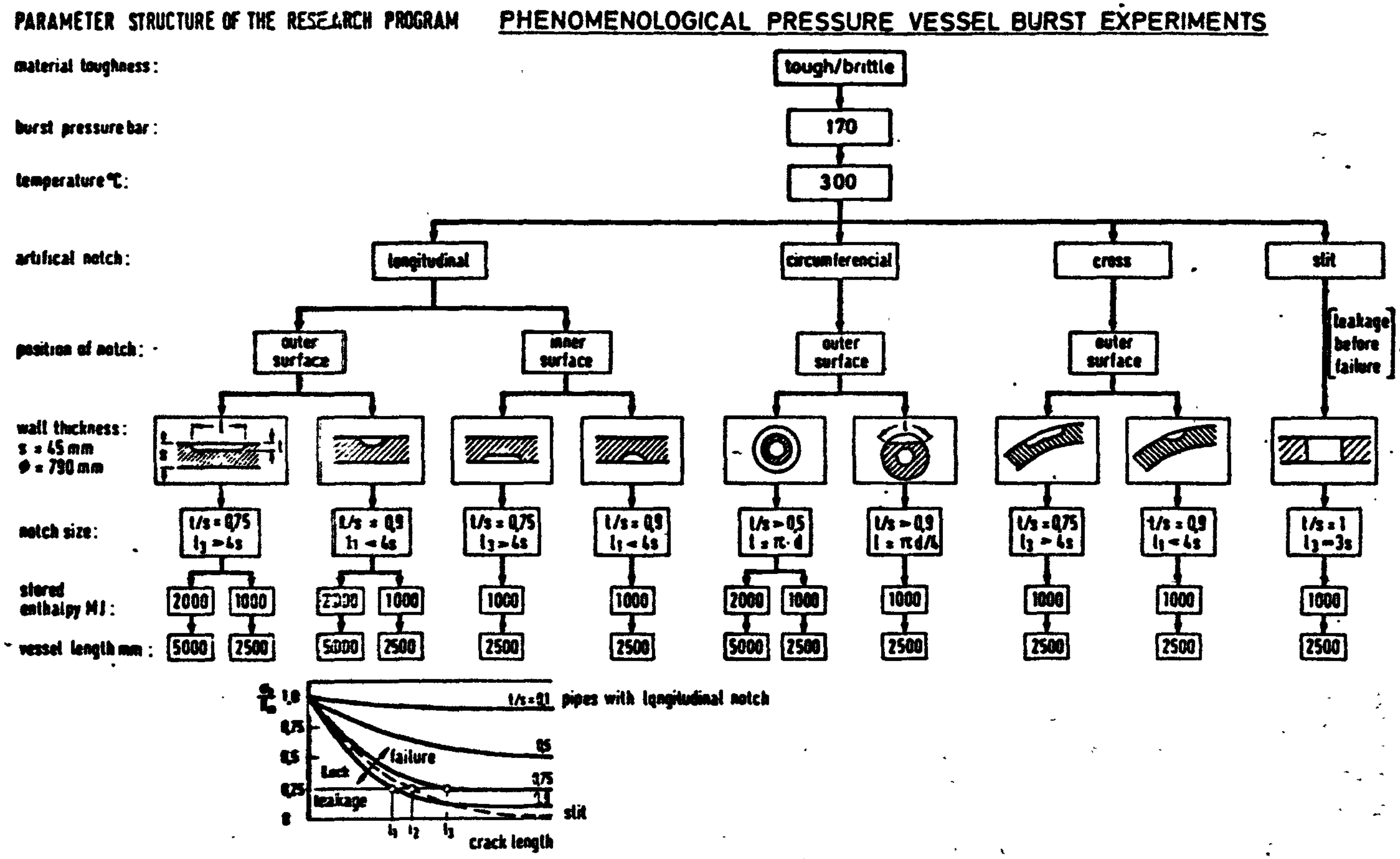
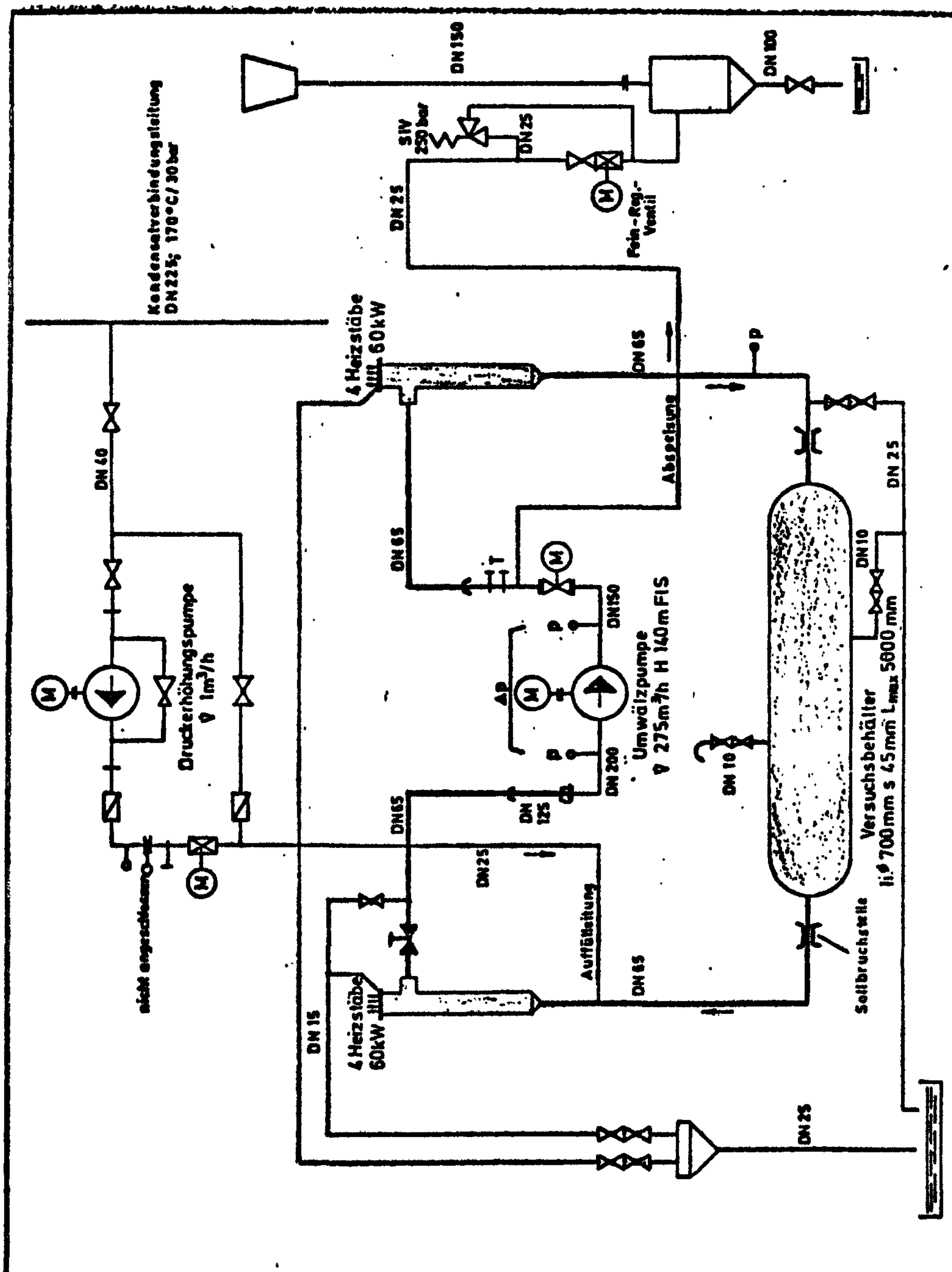

PARAMETER STRUCTURE OF THE RESEARCH PROGRAM
PHENOMENOLOGICAL PRESSURE VESSEL BURST EXPERIMENTS
material toughness:
tough/brittle
burst pressure bar:
170
temperature °C:
300
artifical notch:
longitudinal
circumferencial
cross
slit
position of notch:
outer surface
inner surface
outer surface
outer surface
leakage before failure
wall thickness:
s = 45 mm
Ø = 730 mm
notch size:
t/s = 0,75
l_3 > 4s
t/s = 0,9
l_1 < 4s
t/s = 0,75
l_3 > 4s
t/s = 0,9
l_1 < 4s
t/s > 0,5
l = π · d
t/s > 0,9
l = π d/4
t/s = 0,75
l_3 > 4s
t/s = 0,9
l_1 < 4s
t/s = 1
l_3 = 3s
stored enthalpy MJ:
2000
1000
2000
1000
1000
1000
2000
1000
1000
1000
1000
1000
vessel length mm:
5000
2500
5000
2500
2500
2500
5000
2500
2500
2500
2500
2500
1,0
0,75
0,5
0,25
0
t/s = 0,1
pipes with longitudinal notch
0,5
0,75
1,0
failure
leakage
slit
l_1
l_2
l_3
crack length

Figure 1



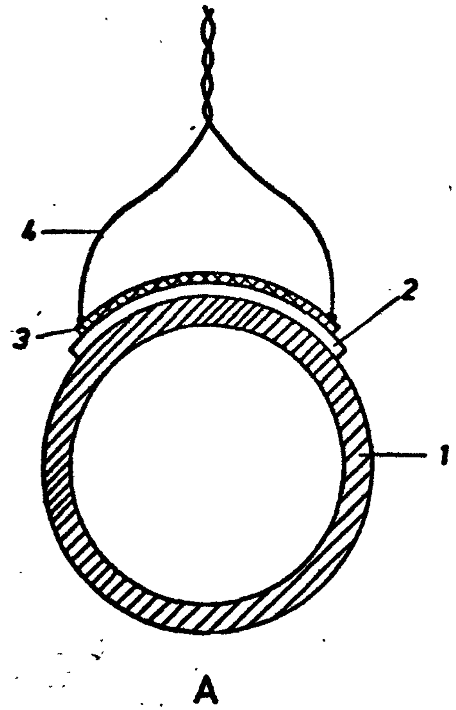
Werk Fritz

Stand vom 10.8.78

Werkstoff:	Gezeichn.: 4 8 78 Bauer	Großkraftwerk Mannheim A.G. Mannheim-Neckarau
	Geprüft:	

Maßstab		Zeichnung Nr.
x	Berstversuche der MPA-Stuttgart Versuchskreislauf	OM 11.345/A

Figure 2



1 pipe
2 ceramic insulator layer (~0,1 mm thick)
3 copper layer (~0,1 mm thick)
4 connecting wire

Figure 3

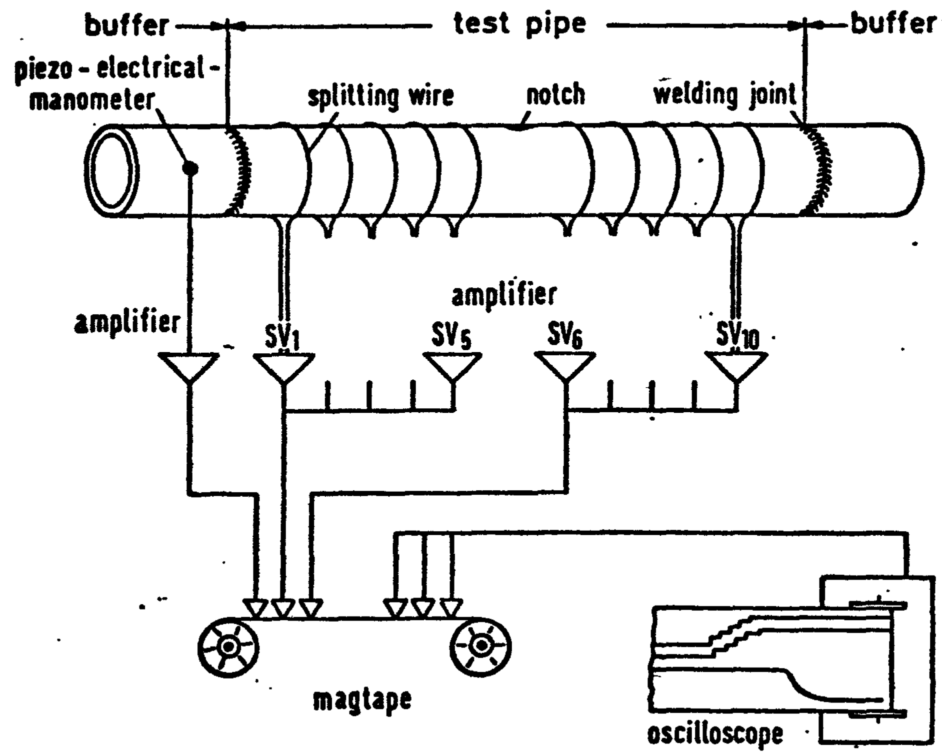


Figure 4

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.10.78 - 31.12.1978	11.2	RS 353
Vorhaben/Project Title High Speed Tensile Tests on Large Specimens Stage I: Model Tests and Constructional Preliminary Work Schnellzerreißversuche mit Großproben Stufe 1: Modellversuche und konstruktive Vorarbeiten		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Staatliche Material-prüfungsanstalt (MPA)
Arbeitsbeginn/Initiated 1.10.1978	Arbeitsende/Completed 31.12.1979	Leiter des Vorhabens/Project Leader Dr. Sturm/Dipl.-Ing. Küri
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Quantification of Safety of Reactor Pressure Vessels

2. Particular Objectives

Determining the behavior of strength and deformation of unwelded and welded large tensile specimens made of ferritic and austenitic steels conducted under high loading and deformation velocity.

Stage I: Testing series performed with the 1,2 MN high speed tensile testing machine (intermediate size) and constructional preliminary work concerning the 12 MN testing unit.

3. Research Programme

3.1 Testing of the 1,2 MN high speed tensile testing machine with regard to its working order

3.2 Testing series on austenitic specimens

3.3 Testing series on ferritic specimens

3.4 Allocation of constructional work regarding the 12 MN large testing unit.

4. Experimental Facilities, Computer Programmes

1,2 MN high speed tensile testing machine

14 Channel-magnetic tape recorder (2 items) with attachments

Extensometer

High-speed camera

5. Progress to Date

Assembling and installing the 1,2 MN high speed tensile testing machine

Strain gauges instrumentation

Preliminary test on austenitic specimen, cross-section to be tested is 30 percent of the nominal size

6. Results

Test readings are being evaluated

7. Next Steps

Performance of further preliminary tests

Checking of the machine

Performance of necessary alterations and assembly

Tests to be continued on specimens of 50 percent testing cross-section

8. Relation with other Projects

High speed tensile testing within the scope of the SNR-300 programme, project Kalkar

9. References

-

10. Degree of Availability

-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	11.2	RS 320

Vorhaben/Project Title	
Rißstopverhalten Crack Arrest Behaviour	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 213, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 8. 78	31. 1. 81	W. Schmitt
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 78	

1. General Aim

Experimental and theoretical work shall be performed for validating of the pre-rupture leak criterion and for demonstrating the crack arrest of unstable, axial, wall-penetrating cracks in order to prove the bursting resistance of pressure-containing nuclear power plant components.

2. Particular Objectives

The experiments provided in the present research project are designed, so that it will be possible to improve present and to establish a basis for fracture mechanics and thermodynamic computing models. These computing models shall be used eventually to assess failure of pressure-containing components and piping used in the primary and secondary systems.

3. Research Program

3.1 Experimental Work

a) A test cylinder provided with and axial part-through crack of defined size fails at a precalculated internal pressure load (pressure reservoir) because of the occurrence of a stable leak.

b) A test cylinder provided with an axial part-through crack of defined size fails at a precalculated internal pressure load with simultaneous formation of an unstable, wall-penetrating axial crack, which is arrested in the basic material.

c) Like test above but with a pressure reservoir.
d) Like test 3 but at a different crack initiation rate, if necessary, with designed crack arrester. Tubes made of 20MnMoNi55, approx. 3 m long, internal diameter = 300, wall thickness s = 20 mm will be used as test cylinders.
e) must be determined in accordance with the results of a) - c)

3.2 Theoretical Work

a) Thermodynamic analyses
b) Fracture mechanics analyses
c) Transferability of results obtained by the research project of other geometries and thermodynamic conditions.

4. Test Facilities

4.1 Experimental set-up

Test tube without succeeding reservoir

The test tube which should exhibit a high degree of dimensional accuracy, in particular with respect to wall thickness and roundness, is provided with a theoretically defined part-through crack (milling of a longitudinal groove).

After this, the tube is closed at both ends with covers. A heating rod is installed in once cover. After the tube has been instrumented, it will be fastened to the floor inside a bunker using a special support structure.

Test tube with succeeding reservoir

Here, after introducing the part-through crack, the test tube is closed only on one side with a cover, in which the heater rod has also been fitted. After it has been instrumented, the tube is welded together with the reservoir - this is a tube procured within the scope of the government-sponsored project RS 104 of 800 mm internal diameter and 4500 mm length. The reservoir is also fitted with heater rods and also instrumented to a small extent. The set-up is fitted in a bunker.

4.2 Instrumentation

The instrumentation of the test tubes should allow statements to be made on the following processes:

- pressure and temperature on tube failure, pressure drop in test tube after tube failure
- flow momentum of the fluid escaping suddenly when the tube fails
- distortion of the test tube
- cracking speed
- leak surface as a function of time
- crack arrest length

4.3 Measuring set-up

The measuring set-up provided for the approx. 40 measuring signals, made up of signal conditioning (measuring and supply voltage amplifiers) and analog recording and display equipment (a total of 3 analog magnetic tape machines) corresponds in principle with the set-up used for the tests for RS 104. It is installed in a separate protected room located at a distance of approx. 15 m from the test area in the bunker.

5. Progress to Date

To 3.1 The test set-up was finalized for the first "stable leak" experiment in a test tube with pressure reservoir, the most important measuring data will be: leak surface and pressure drop as functions of time and also indirect measurement of the jet force.

A preliminary test was conceptually designed for testing the quick-response pressure pickups required for the experiment.

To 3.2 Discussions have been held on the details of the concepts of thermohydraulics and structural dynamics.

6. Results

None

7. Next Steps

To 3.1 Assembly of test tube 1

Tests of measuring pickups

To 3.2 Start of finite-element calculations

Start of thermohydraulics calculations

8. Relation with Other Projects

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9. References

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10. Degree of Availability

141-1-02 4111-10		11-2
TITRE EVALUATION DE LA PROBABILITE DE RUPTURE D'UNE CUVE DE REACTEUR A EAU PRESSURISEE.		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) FAILURE PROBABILITY CALCULATION OF A PWR PRESSURE VESSEL.		Organisme exécuteur CEA + EURATOM + FRAMATOME Responsable.
Date de démarrage 1975	Etat actuel en cours	Scientifiques
Date d'achèvement 6/1979	Dernière mise à jour 12/1978	

1. OBJECTIF GENERAL

Evaluation de la probabilité de rupture d'une cuve de réacteur à eau pressurisée par une méthode probabiliste basée sur les lois de la mécanique des ruptures. Cette méthode devrait permettre d'aboutir à :

- une comparaison entre la probabilité de rupture de la cuve de réacteur et celle des canalisations du circuit primaire.

- définir l'importance relative des différents paramètres intervenant dans le mécanisme de la rupture.

- une comparaison entre les différentes méthodes de construction et d'exloitation.

2. OBJECTIFS PARTICULIERS.

Toutes les données sont introduites sous forme d'histogrammes. Le projet général se subdivise en 7 sous programmes.

- Revue et discussion des principales méthodes statistiques et probabilistes utilisées.

- Collecte des défauts recensés dans les différents cuves existantes. 3 contrats ont été signés avec des fabriquants européens pour recueillir les données sous un formulaire standard.

.../...

- Recueil des données relatives aux propriétés mécaniques de l'acier A 508 Cl 2 sous forme probabiliste. La plupart de ces données sont obtenues à partir de L'US-HSST programme.

- Validation des formules de calcul du coefficient d'intensité de contrainte.

- Validation des lois de propagation des fissures par vérification des données expérimentales obtenues sur des éprouvettes CT, et sur des plaques ou des tubes comportant des défauts elliptiques artificiels.

- Analyse du système de contrôle du réacteur afin de définir les conditions de fonctionnement programmées et accidentelles ainsi que leur probabilité d'occurence.

- Mise au point d'un programme de calcul faisant entrer, sous forme d'histogramme, le défaut initial, la propagation pendant le fonctionnement, la résistance du matériau et les critères de rupture.

Cette étude comporte également 3 programmes expérimentaux :

- Croissance en fatigue des défauts naturels

- Croissance des défauts sous eau de composition anormale

- Etude de la propagation des défauts en mode mixte.

3. INSTALLATION EXPERIMENTALES ET PROGRAMME :

Travaux effectués avec le concours d'EURATOM et de FRAMATOME ainsi que, par contrats avec d'autres industries et universités ; Universités de Technologie de Compiègne - Université de Metz - Creusot Loire (Saint-Etienne) ; CETIM (Senlis), - STCAN (Paris).

Les calculs et la mise en oeuvre des codes sont effectués au Centre Commun de Recherche d'ISPRA.

4. ETAT DE L'ETUDE.

Avancement à ce jour :

1°) La pemière partie est terminée : il en a été déduit que le nombre de cuve est réellement trop insuffisant pour en tirer une valeur probabiliste de rupture.

2°) Recueil des données sur les dimensions des défauts observés sur des cuves de réacteur après fabrication. Les données de constructeurs ont été fournies à ISPRA qui en effectue le traitement statistique. D'autre part une méthode a été mise au point pour calculer la distribution de la largeur des défauts (valeur qui n'est pas fournie par les constructeurs).

3°) Recueil des données sur les propriétés de l'acier A 508 C13 - terminé.

4°) Validation des formules de calcul des coefficients d'intensité des contraintes - terminée.

.../...

5°) Validation des critères de rupture en situation fragile et ductile - le critère de DOWLING et TOWNLEY a été retenu et validé sur les résultats publiés.

6°) Situations de fonctionnement. Les 24 situations de fonctionnement normal sont totalement définis. Les situations accidentelles sont en cours (LOCA-ATWS- surpression)

7°) Programme général de calcul. L'organigramme du programme est terminé et la transcription en FORTRAN est en cours.

Etat d'avancement des programmes experimentaux.

1°) Etude de la croissance des défauts naturels en fatigue. 5 éprouvettes ont été fabriquées par Creusot Loire et sont en cours d'essais au STCAN.

2°) Etude de la propagation de fissures en fatigue sous eau. L'étude bibliographique des incidents sur la chimie de l'eau des PWR est terminée. La composition de l'eau a été déterminée selon ces résultats et les essais sont commencés.

3°) Etude de la propagation de fissures en mode mixte. Les éprouvettes sont mise au point et les moyens de calculs sont en cours de développement.

Résultats essentiels.

Ils sont relatifs aux viroles du coeur. La croissance des défauts au cours des transitoires normaux est assez faible.
La probabilité finale apparaît très sensible à la méthode utilisée pour évaluer la dispersion de la ténacité KIC.
La probabilité pour qu'un défaut de 6mm de large et placé à 9mm de la paroi interne, entraine la rupture de la cuve au cours d'un LOCA est de 6.10^{-5}. Cette probabilité est inférieure à 10^{-6} si le défaut est placé à 25mm de la paroi interne de la cuve.

5. PROCHAINES ETAPES.

- Mise au point du code ce calcul.
- Etude de sensibilité des paramètres.

6. RELATIONS AVEC D'AUTRES ETUDES.

7. DOCUMENTS DE REFERENCE

- Etude probabiliste de la rupture de cuve des chaudières nucléaires à eau ordinaire : J. DUFRESNE

Premier rapport d'avancement	DSN 116
Second rapport d'avancement	DSN 145
Troisième rapport d'avancement	DSN 177
Quatrième rapport d'avancement	DSN 216

- "The CEA sponsored research programm on PWR reactor vessel failure probability calculation"

  J. DUFRESNE ET AUTRES. ANS Los Angeles mai 1978

- "Evaluation de la probabilité de rupture d'une cuve de réacteur à eau préssurisée"

  J. DUFRESNE ET AUTRES. NUCLEX 78 Bale octobre 78

- "A model for estimation of failure probability of pressurised water reactor vessle"

  J. DUFRESNE ET AUTRES. ANS Bruxelles octobre 78

8. DEGRE DE DISPONIBILITE DES DOCUMENTS.

141.1.07/4112.02		II-2
TITRE ETUDE DES PROBLEMES DE CORROSION LIES AU REVETEMENT DES CUVES DE REACTEURS.		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) CORROSION ASPECTS OF NUCLEAR PRESSURE VESSEL STEELS.		Organisme exécuteur CEA/DCA/SECE/SECAE Responsable
Date de démarrage 1.11.78	Etat actuel EN COURS	Scientifiques
Date d'achèvement 31.12.80	Dernière mise à jour 1.1.79	

1. OBJECTIF GENERAL :

Les cuves des réacteurs à eau de type PWR sont revêtues d'une couche d'acier austénitique. En cas de défaut ou de fissuration de ce revêtement, il convient d'examiner du point de vue de la corrosion les conséquences sur l'acier de cuve.

2. OBJECTIFS PARTICULIERS :

Les essais de corrosion sont effectués dans les conditions PWR sur des échantillons revêtus d'acier inoxydable présentant différents défauts superficiels avec ou sans contrainte mécanique.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME :

4. ETAT DE L'ETUDE :

1. Avancement à ce jour :

- Fabrication d'éprouvettes à âmes ferritiques entièrement revêtues.
- Réalisation des défauts superficiels.

2. Résultats essentiels :

5. PROCHAINES ETAPES :

Expérimentation de longue durée en milieu PWR.

4. ETAT DE L'ETUDE :

Les résultats 1978 concernent essentiellement la qualification de l'acier FRAMATOME (1) et de la soudure hors irradiation avant découpage et expédition aux laboratoires étrangers, participant au programme. Les premières irradiations sont également en préparation.

5. PROCHAINES ETAPES :

Voir tableau 3.

6. RELATIONS AVEC D'AUTRES ETUDES :

Sans objet.

7. DOCUMENTS DE REFERENCE :

(1) SA 508 classe 3 - Programme AIEA. Description de la fabrication TE/M.DC03-48 du 30.03.78.

8. DEGRE DE DISPONIBILITE :

Disponible.

TITLE (ORIGINAL LANGUAGE): Sull'inizio di propagazione di una rottura in un recipiente in pressione costruito con materiale duttile		CLASSIFICATION: 11.2
		COUNTRY: ITALY
		SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Elasto-plastic analysis of crack-initiation in thin-walled low carbon steeel pipes.		ORGANISATION: University of Pisa
		PROJECT LEADER: Costantino CARMIGNANI
INITIATED: 1976	COMPLETED:	SCIENTISTS: Sergio REALE
STATUS: completed	LAST UPDATING: June 1979	

1. General aim
   General aim of the research is to utilize EPFM parameters in order to forecast the behaviour of pipes considered as simple structures.

2. Particular objectives
   Particular objective is an experimental J and COD analysis of crack initiation in through-thickness cracked pipes.

3. Experimental facilities and programme
   Facilities: pressurization system for burst tests and a frame for bending tests. A data acquisition system is also available.
   Programme: tests are run on low carbon steel pipes at room temperature in two basic conditions:
   - burst tests under internal pressure on longitudinally cracked pipes;
   - four point bending tests on circumferentially cracked pipes.

4. Project status
   4.1 Progress to date: it is developed a suitable technique for processing results. Reliable J and COD curves are obtained.
   4.2 Essential results: the experimental approach gives a reliable technique to evaluate the "critical" value of fracture mechanics parameters to detect some resistance correlated curve to forecast fracture behaviour of structures.

5. Reference documents

1- C. CARMIGNANI, S. REALE, G. TOMASSETTI
"Fracture Mechanics Research Activities of the "Istituto di Impianti Nucleari" of the University of Pisa in cooperation with the CNEN"
Presented at "Specialist's meeting on elasto-plastic fracture mechanics, Daresbury, May 1978.

2- C. CARMIGNANI, S. REALE
"Elasto-plastic Analysis of Fracture Behaviour of Cracked Ductile Pipes"
VII Congress on Material Testing, Budapest 1978.

45.897

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Elasto-plastic analysis of crack-initiation in thin-walled low carbon steel pipes.	11.2

3- C. CARMIGNANI, S. REALE, G. TOMASSETTI
"Experimental J and COD Analysis of Crack Initially in Though Thickness Cracked Pipes"
To be presented to V° SMIRT, Berlino, August 1979.

6. Degree of availibility
Open, please contact: Sergio REALE, Istituto di Impianti Nucleari, Via Diotisalvi, 2, Pisa.

	CLASSIFICATION: 11.2
TITLE (ORIGINAL LANGUAGE): Fatica ad alta temperatura di elementi strutturali	COUNTRY: ITALY
	SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): High temperature fatigue on structural elements	ORGANISATION: University of Pisa
	PROJECT LEADER: Enrico MANFREDI

INITIATED: 1976	COMPLETED: 1980	SCIENTISTS: Aldo DEL PUGLIA, Emilio VITALE
STATUS: in progress	LAST UPDATING: June 1979	

1. General aim
This program is aimed at getting information about the low cycle, high temperature (creep-fatigue) behaviour of welded structures and components manufactured in austenic stainless steels (type AISI 304 and 316).

2. Particular objectives
To measure experimentally the life to rupture of simple structural specimens and to correlate these data with reliable prediction approaches, with the aid of further experimental information obtained either during these tests either on smooth small size specimens.

3. Experimental facilities and programme
20 ton dynamic test rig, with temperature and cycle control system, transducers and data logging instrumentation, data processing computer.
The programmed tests refer to controlled axial cycling of AISI 304 and 316 welded tubular specimens 60,3 mm outer diameter, 500 mm long.
The tests are performed at 650 °C temperature; triangular and up to 30 minutes hold times waveforms are used.
About 40 tests both in AISI 304 and 316, using different welding and heat treatment parameters are programmed. Smooth specimen testing to characterize visco-elasto-plastic properties and creep-fatigue resistance of base and weld material is programmed, as well as metallographic and fractographic analysis.

4. Project status
   1. Progress to date: the programmed tests on AISI 304 has been completed. Non linear code analysis of visco-elasto-plastic behaviour has been performed
   2. Essential results: it has been detected a marked reduction of the fatigue life of the weldments, which is due to influence of defects, to less favourable metallurgical structure and to strain concentration phenomena in the weld zone.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
High temperature fatigue on structural elements	11.2

5. Next steps

The series of tests on AISI 316 specimens, manufactured on strict quality assurance requirements will be performed next, together with characterization tests on smooth specimens.

6. Reference documents

1- A. DEL PUGLIA, E. MANFREDI, R. MATERA, G. PIATTI
"High temperature Low Cycle Fatigue Damage Analysis of Welded Stainless Steel Tubular Elements"
5th SMiRT Conference, Berlin 1979, Paper L12/4.

2- C. CARMIGNANI, A. DE PAULIS, E. VITALE
"Influenza delle saldature in componenti sollecitate ciclicamente ad alta temperatura"
VII Convegno AIAS, Cagliari 1979.

3- E. MANFREDI
"Valutazione approssimata di un fattore di concentrazione delle deformazioni in regime elasto-plastico"
Atti Istituto di Impianti Nucleari dell'Università di Pisa, RP 339(79)

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

	CLASSIFICATION: 11.2
TITLE (ORIGINAL LANGUAGE): Ricerca sperimentale sull'accumulo delle deformazioni e sulle rotture per fatica di tubazioni soggette a carichi variabili ripetuti.	COUNTRY: ITALY SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Experimental research on incremental collapse and plastic fatigue of piping components	ORGANISATION: Istituto di Impianti Nucleari-Università di PISA PROJECT LEADER: Giovanni NERLI
INITIATED: 1976 | COMPLETED: 1981 STATUS: in progress | LAST UPDATING: June 1979	SCIENTISTS: Paolo CITTI, Sergio REALE, Paolo RISSONE

1. General aim
   Study of incremental collapse phenomena with progressively increasing deflections and plastic fatigue phenomena in stainless steel piping components subjected to variable repeated loads.

2. Particular objectives
   Relating to stainless steel T-branched pipes models fixed at their flanged ends and loaded in two sections, by variable repeated loads, tests are carried out to determine:
   a- strain hardening behaviour
   b- shakedown load conditions
   c- plastic fatigue fracture loads.
   An "ad hoc" computer code is also developed.

3. Experimental facilities
   All tests are carried out at Laboratories of Istituto di Ingegneria Meccanica of Florence University, where load apparatus, automatic data acquisition system and POP 11/40 computer are available.

4. Project status
   1. Progress to date: Test method has been developed and results have been obtained on strain hardening behaviour of tested models. Mathematical model has also been developed.
   2. Essential results: Results are related in reference document n. 4.

5. Next steps
   Other tests on several models and the development of computer code will be performed.

6. Reference documents
   1- P. CITTI, A. DEL PUGLIA, G. NERLI
   "Experimental study on the effect of variable repeated loads on steel piping components"
   3rd SMIRT, London 1975.

45.897

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Experimental research on incremental collapse and plastic fatigue of piping components.	11.2

2- P. CITTI, A. DEL PUGLIA, G. NERLI
"Esperienze su tubazioni soggette a carichi ripetuti"
3° AIAS, Bologna 1975.

3- P. CITTI, P. RISSONE
"Calcolo e verifica sperimentale della deformata di una travatura in campo elastico incrudente"
4° AIAS, Roma 1976.

4- P. CITTI, G. NERLI, S. REALE, P. RISSONE
"Experimental analysis on elasto-plastic behaviour of T-branched stainless steeel pipe"
5th SMIRT, Berlin 1979.

7. Degree of Availability
Open, please contact: Sergio REALE, Istituto di Ingegneria Meccanica, Facoltà di Ingegneria, Via S. Marta 2, Firenze.

Rotterdam Dockyard and others -127A-	CLASSIFICATION: 11.2
TITLE: Breukanalyse Onderzoek aan Stompen (BROS)	COUNTRY: THE 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: May 1979	SCIENTISTS: M.J.C. Broekhoven H.J.M. van Rongen A. de Sterke/~~M.J.A. Koning~~ W. VAN PAAREN
STATUS: in progress | COMPLETED: End 1979	

General aim

Crack extension behaviour in heavy section nuclear steel pressure vessels in areas of complicated geometry.

Particular objectives

In particular attention will be paid to crack occuring in areas of complicated geometry and stress distribution, notably nozzle corner regions. The program covers the following items:

- early detections of defects
- detailed surveillance of the growth of defects
- establishment of prediction for further growth of defects by fatigue and fracture.

Experimental facilities and program

The main activities are:

- theoretical research directed towards the development of efficient computation methods to calculate the crack extension behaviour for complicated configurations, such as nozzle corner cracks, using advanced analytical and finite element techniques,
- experimental investigations of crack extension behaviour in uni-axial and bi-axial loaded cracked nozzle-on-flat-plate models, manufactured mainly from nuclear grade pressure vessel steel (ASTM A508 Cl.2),
- determination of fatigue and fracture (elastic and elastic-plastic) related material parameters for A508 material,
- research on the applicability of Acoustic Emission techniques to detect, locate and characterize crack extension in the material of concern,
- the main test facility to be used is a bi-axial loading bench, suitable for plate dimensions 700x700x15 mm, capacity 200 tonnes static force, 200 tonnes dynamic loading, frequency 1Hz.

Project status

- A series of nozzle-on-flat-plate models has been uni-axially fatigue loaded and after sufficient crack growth, fractured by overload,
- Computer programs have been completed,
- Accoustic Emission testing has been completed,
- Fatigue and fracture (J_{IC}-) tests on standard specimens have been completed,
- The bi-axial loading bench has been constructed and tested,
- The testing of flat plate test pieces using the biaxial loading bench has been finished.
- The publishing of the final technical reports (in English language) is in progress.

Next steps The next step will be

the publishing of the final technical report concerning the ~~above~~ afore mentioned tests.

Relation with other projects

Results of this project will be used within the "EPOSS" project. Furthermore a follow-up program, BROS II, is anticipated.

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 Minist~~e~~ry of Economic Affairs, Laan van Nieuw-Oost Indië 123, The Hague. Requests for obtaining copies should be sent to this address.

Budget

$f.\ 6.10^6$.

Personnel

46 manyears.

Rhine Schelde Verolme and others	CLASSIFICATION: 11.2
TITLE: Onderzoek naar het gedrag van scheuren in dikwandige stalen constructies	COUNTRY: THE NETHERLANDS
	SPONSOR: Ministry of Economic Affairs
	ORGANIZATION: Rhine Schelde Verolme and others
TITLE (ENGLISH LANGUAGE): Fracture analysis research on nozzle intersections (BROS II)	PROJECTLEADER: C.J. Drijver
INITIATED: July 1979 — LAST UPDATING: -	SCIENTISTS: A. Bakker / R.M. van Kuij, W. van Maaren / J. Poort, H.J.M. van Rongen
STATUS: Starting — COMPLETED: 1982	

General aim

Crack initiation and extension behaviour in heavy section steel structures under elasto-plastic conditions.

Particular objectives

In particular attention will be paid to the crack initiation and extension behaviour in steel components under contained yielding conditions. The program will be directed at establishing fracture parameters by numerical and experimental methods for complicated structures. Beyond this an elastic analysis of a cracked nozzle corner under thermal loading conditions will be carried out.

Experimental facilities and program

The main activities will be:

- theoretical research regarding crack extension under thermal loading conditions for nozzle corner cracks using linear elastic finite element methods
- theoretical research concerning elastic-plastic computation procedures for three dimensional configurations; finite element methods will be applied
- experimental research on 2D configurations under biaxial load to evaluate the merits of various elastic-plastic fracture parameters under practical conditions
- research on parameters which describe stable crackgrowth upto instability, R-curve approach
- a preliminary study on the merits of testing a large pressure vessel to verify the applicability of the current safety analysis methods for nuclear reactors.

Project status

A detailed proposal for the program has been issued and it is anticipated that execution will start in july 1979.

Particular objectives; Experimental facilities and programme; Project status; Next steps; Relation to other projects; Reference documents; Degree of availability; Budget; Personnel.

Rhine Schelde Verolme and others	CLASSIFICATION : 11.2

Relation with other projects

BROS II is a direct follow-up program of the BROS and EPOSS project.

Reference documents

Future documents will be prepared in the English language.

Degree of availability

Future documents will be submitted to the Ministry of Economic Affairs, Laan van Nieuw-Oost Indië 123, The Hague. Requests for obtaining copies should be sent to this address.

Budget

Hfl. 3.000.000,--

Personnel

20 manyears.

	CLASSIFICATION: 11.2
TITLE (ORIGINAL LANGUAGE): Pressure vessel integrity under fault conditions (M 7.4)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Risley)
	PROJECT LEADER: B Watkins (Risley)

INITIATED: March 1978	COMPLETED:	SCIENTISTS:
STATUS: In progress	LAST UPDATING: March 1979	

Background

Avoidance of fast fracture under emergency/fault conditions is a primary consideration in any safety assessment of pressure vessel integrity. The thermal shock which accompanies a major fault condition (steam line break or loss of coolant accident) produces through thickness thermal gradients with attendant gradients in stresses and fracture toughness. Further work is needed to assess the influence of the thermal stress gradients on fracture initiation. Current ASME flaw evaluation procedures rely on the concept of crack arrest to justify pressure vessel integrity under fault conditions. The validity of a crack arrest philosophy is not fully substantiated and considerable work is in progress on this topic, particularly in the US. Critical experiments to demonstrate the relevance of crack arrest have still to be made and test methods for measuring "crack arrest toughness" are not standardised. The work proposed will rely heavily on the US research and will be concerned with understanding the principles of crack arrest using laboratory scale tests.

Objectives

To develop a laboratory test method to investigate the relevance of the crack arrest concept applied in ASME XI.

Programme

1. Assess feasibility of experimental techniques for applying simple thermal shock loading to pre-stressed fracture mechanics test pieces and demonstrate whether cracks will arrest in an increasing toughness gradient.

2. Develop more sophisticated methods of applying a quantifiable thermal shock using information from Instruction 1 and the evaluation of existing work.

Facilities

Reference Documents

		Classification	11.2
Title 1	Development of failure and fracture analysis methods.	Country	U.K.
		Sponsor	CEGB
Title 2		Organisation	CEGB Berkeley Nuclear Labs, Berkeley, Glos.
Initiated	Completed	Project Leaders	
Status	Last updating	B J L Darlaston	

1. General Aim

To develop and validate fracture and failure analysis methods applicable to Water Reactor systems.

2. Particular Objectives

To develop and validate the two criteria approach known as the CEGB Integrity Assessment Route (R6). The associated development of the BERSAFE finite element suite of programs, evaluation of modelling techniques for determination of collapse loads of structures, methods for thermal and residual stresses.

3. Project Status

J_θ for thermal stress has been evolved. A method for thermal and residual stresses has been formulated for introduction into R6. Validation work continues on growth of fatigue cracks and ligament failure in structures associated with leak before break.

4. Next Steps

Further work on thermal and residual stress analysis and experimental validation of all aspects.

Reference Documents

CEGB Reports (available from Berkeley Nuclear Laboratories).

	CLASSIFICATION : 11.2.1.
TITLE : Metallurgy-Dosimetry Research and Development Program for Improved LWR Pressure Vessel Surveillance and In-Vessel Neutron Embrittlement Characterization of a Reference Commercial Steel.	COUNTRY : Belgium
	SPONSOR : CEN-SCK
	ORGANISATION : CEN-SCK/S.A.COCKERILL
	PROJECT LEADER : Mrs. A. Fabry and J. Debrue/ Mr. R. Salkin

INITIATED : June 1977	COMPLETED :	SCIENTISTS : G. and S. De Leeuw, P. Gubel, G. Minsart, H.Tourwé,Ph. Van Asbroeck/ A. Scailteur, J. Widart
STATUS : continuing	LAST UPDATING :	

1. General Aim :

The objective of the program is to improve LWR pressure vessel surveillance practices and to validate projected in-vessel neutron embrittlement behaviour, more specifically for a reference national steel type A 508 cl. 3.

2. Particular Objectives :

The program encompasses three interrelated main tasks :

a) Validation of transport theory methods for prediction of surveillance and in-vessel neutron flux spectra and fluences

b) Improvement, validation and calibration of surveillance dosimetry methods

c) Characterization of a reference Belgian commercial steel (type A 508 cl.3) in terms of the neutron flux level, flux spectrum and irradiation temperature conditions of PWR pressure vessels.

3. Experimental Facilities and Program :

For task a :
- The one-dimensional spherical geometry Iron Shell reference benchmark neutron field at CEN-SCK
- the LWR low flux pressure vessel mock up at the Pool Critical Assembly (PCA) at ORNL
- test regions in the BR3 reactor at CEN-SCK.

For task b :
- the uranium-235 fission spectrum standard neutron fields at CEN-SCK and NBS
- test regions in the BR3 reactor at CEN-SCK and in the FRJ-2 reactor at KFA, Jülich.

For task c (Charpy-V, tensile, compression and TEM steel specimens only at present) :
- the LWR high flux pressure vessel mock up at the Pool Side Facility (PSF) of the Oak Ridge Research Reactor (ORR)
- test regions in the BR3 reactor at CEN-SCK
- in-core and reflector rigs in the FRJ-2 and FRJ-1 reactors at KFA, Jülich
- in-core thimble facility under design at the high flux materials testing reactor BR2 at CEN-SCK.

4. Project Status :

1. Progress to date : dosimetry and metallurgy irradiations in progress or about to start in all facilities delineated under tasks b and c (except BR2, planned for fall 1980); extensive $^{6}Li(n,\alpha)$ and proton recoil neutron spectrometry, fission chamber, radiometric and silicon damage rate measurements in progress or completed in PCA and Iron Shell benchmarks, task a; extensive discrete-ordinates transport theory analysis in progress or planned for all considered neutron fields.

2. Essential results : see reports and forthcoming publications at the Third ASTM-EURATOM Reactor Dosimetry Symposium, Ispra, October 1979.

5. Next Steps :

Continue experiments and analysis. Contribute to the design, implementation and initial use of a mètallurgical reference facility in the NBS reactor. Participate in the preparation of new ASTM recommended practices for LWR pressure vessel surveillance. Participate in the organization of a LWR surveillance "transport theory blind test". Evaluate feasibility to irradiate small fracture toughness specimens in BR2.

6. Relation to Other Projects :

Coordinated with the US Nuclear Regulatory Commission "LWR Pressure Vessel Irradiation Surveillance Improvement" program at HEDL, ORNL and NBS and with similar research at the Electric Power Research Institute, at KFA, Jülich, Germany and at Rolls-Royce Associate, England.

7. Reference Documents :

- CEN-SCK Annual Scientific Report 1977, BLG 524, and 1978, BLG 527.
- LWR Pressure Vessel Irradiation Surveillance Dosimetry Quarterly Progress Report, July-September, 1977, NUREG/CR-0038.

8. Degree of Availability : no restriction.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	11.2.1	RS 207

Vorhaben/Project Title	
Erweiterung des Einsatzbereiches kobaltfreier Werkstoffe für Reib- und Verschleißbeanspruchung im NDES von Leichtwasserreaktoren Extension of the Use of Cobalt-Free Materials for Resistance of Friction, Wear and Corrosion in the Nuclear Steam Supply System (NSSS) of LWR'S	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 41, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 5. 76	30. 9. 79	L. Stieding
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 78	

1. General Aim

Reduction of contamination of the NSSS by products of corrosion and wear, which together with the coolant pass through the reactor core and thereby become activated.

2. Particular Objectives

a) Selection and (eventually in cooperation with the manufacturers) development of Co-free plating materials, which can also be exposed to impact stress conditions and which can be worked more easily than Everit 55. Investigations of the behaviour of these materials and conditions of sliding, impact and corrosion.

b) Selection and evaluation of plasma-sprayed materials with satisfactory corrosion and wear properties under PWR and BWR conditions.

c) Selection and evaluation of stainless steels which are expected to show better friction and wear behaviour than material No. 1.4550.

3. Research Program

3.1 Selection of test materials:
pre-selection, acquisition, fabrication of samples, adhesion tests, friction and wear tests at 20 °C.

3.2 Corrosion tests:
fabrication of samples, corrosion tests at 320 °C, tests for resistance to integranular corrosion.

3.3 Friction and wear tests:
fabrication of samples, tests at 20 °C and 300 °C.

3.4 Mechanical tests:
fabrication of samples, adhesion tests, brittleness test

3.5 Evaluation of test results:

4. Test Facilities

The required test facilities have either been used in previous tests or have recently been developed (hydraulic apparatus for RT tests; stand for high-temperature wear tests at 300 °C with simultaneous influence of corrosion; static and refreshing autoclaves for long term corrosion tests) and are available for the above described tests.

5. Progress to Date

To 3.1 Ten friction tests were performed at 20 °C in water with the second series of the VEW test alloys (weld cladding).
One friction test with Pantanax 2510 Mo-G at 280 °C in water.
Friction tests at 20 °C in water with 3 test alloys (2nd series) at enhanced surface pressures (up to 112 N/mm^2).

To 3.2 Corrosion test on pressure gun and plasma sprayed layers at 350 °C in primary water.

To 3.3 Structural tests of Nitronic 60 construction material.

6. Results

To 3.1 The friction tests performed with the test alloys (2nd series) did not reveal any indication of adhesive wear, even with high surface pressure (112 N/mm^2). Thus it was proved that special austenitic steels can also be applied under extreme load conditions.

The corrosion tests with Al_2O_3 layers led to a complete chipping-off of the claddings whether applied with the pressure gun or in the forms of plasma sprayed layers.

The friction test using Pantanax 25/O Mo-G (both friction surfaces) which was performed at 280 °C in primary water showed a good behaviour of this combination within the above temperature range.

To 3.2 The Mo-cladded corrosion samples showed chipping-off here and there even prior to test, presumably caused by contamination.

To 3.3 It is anticipated that the Nitronic 60 examination, to be performed by irradiation tests, will not show any structural changes of this steel at primary circuit temperatures.

7. Next Steps

To 3.1 Friction tests at 300 °C will be performed with those plate weldings and construction materials that had shown favourable results in primary water at room temperature-friction tests and/or corrosion tests.

To 3.2 Continuation of corrosion tests with sprayed layer.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period 1.1.78 - 31.12.78	Klassifikation/Classification 11.2.1 - 11.2.2	Kennzeichen/Project Number RS 101A
Vorhaben/Project Title Bau einer Prüfhalle der Staatlichen Materialprüfungsanstalt Stuttgart zur Durchführung des Forschungsprogramms Reaktorsicherheit Construction of a Testing Hall by the MPA in order to conduct the Research Programme Reactor Safety		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor OFD-Stuttgart Materialprüfungsanstalt Stuttgart (MPA)
Arbeitsbeginn/Initiated 1.7.76	Arbeitsende/Completed 31.12.78	Leiter des Vorhabens/Project Leader Wagner (OFD) Doll (MPA)
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Quantification of Safety of Reaktor Pressure Vessels.

2. Particular Objectives

Construction of the Testing hall with the Testing cell and fundaments for the full size vessel (RS 245) and the 10 000 Mp-Tensile Testing Machine (RS 101)

3. Research Programme

-

4. Experimental Facilities

-

5. Progress to date

Stage of construction:

The testing cell including the fundament for the full size vessel has been finished (Fig. 1-2) and the steel structure above the testing cell (second stage) mounted and roofed (Fig. 3). The 126 Mp-hangar crane functions over the total length of the hall (Fig. 4). The works for installations such as heating, cooling, plumbing and electrical fittings have nearly been finished. Panelling of the complete hall (internal shell) has been finished (Fig.5).

6. Results

-

7. Next Steps

The installation and external panelling including heat insulation have still to be carried out.

8. Relation with other Projects

RS 245 Research Programme Full Size Vessel
RS 304 Research Programme of Integrity of Components

9. References

-

10. Degree of Availability

Upon request at BMFT

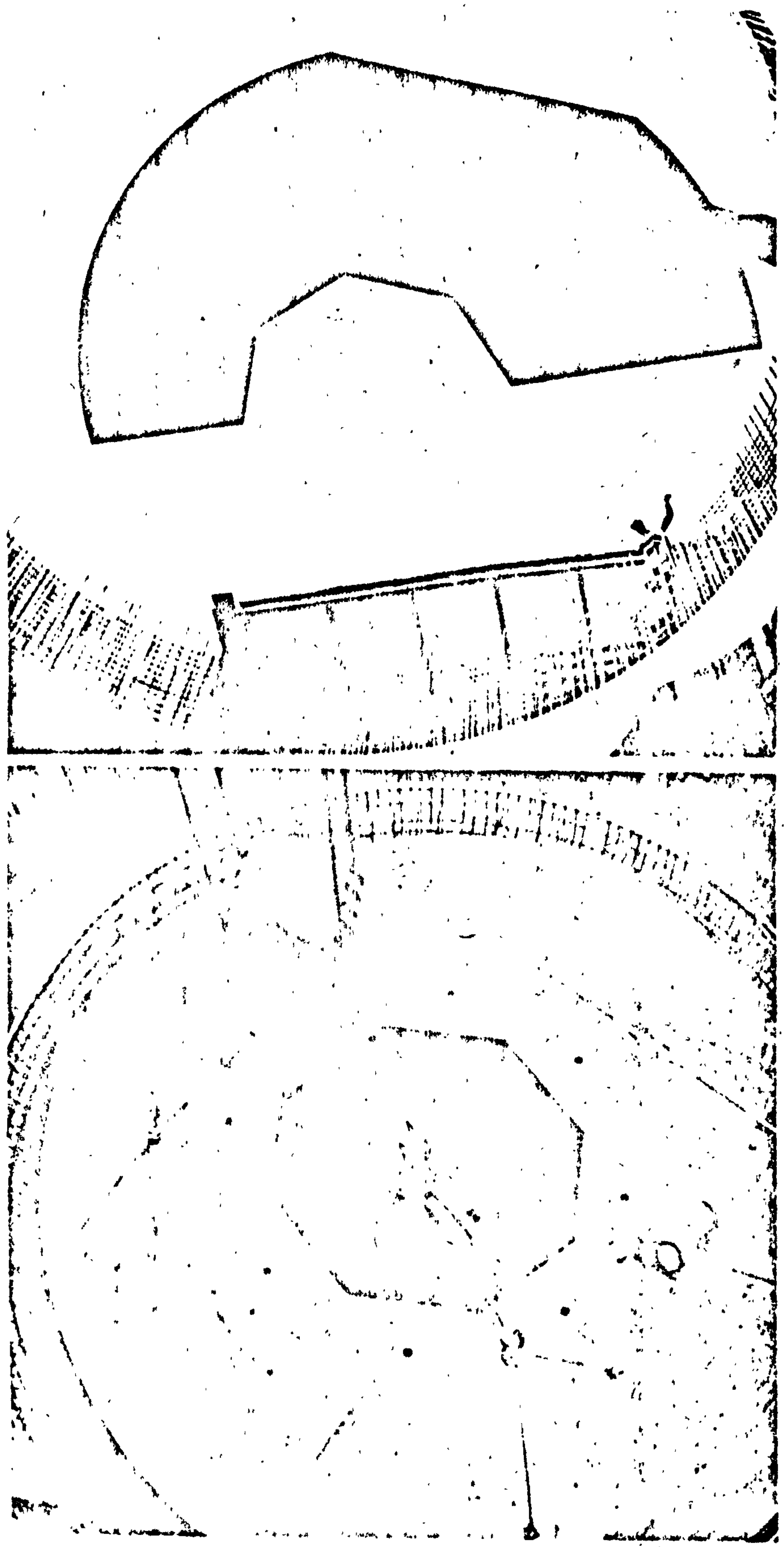


Fig. 1 - 2: Mounting the upper fundament plate (steel 50 mm, two parts) with mobile crane

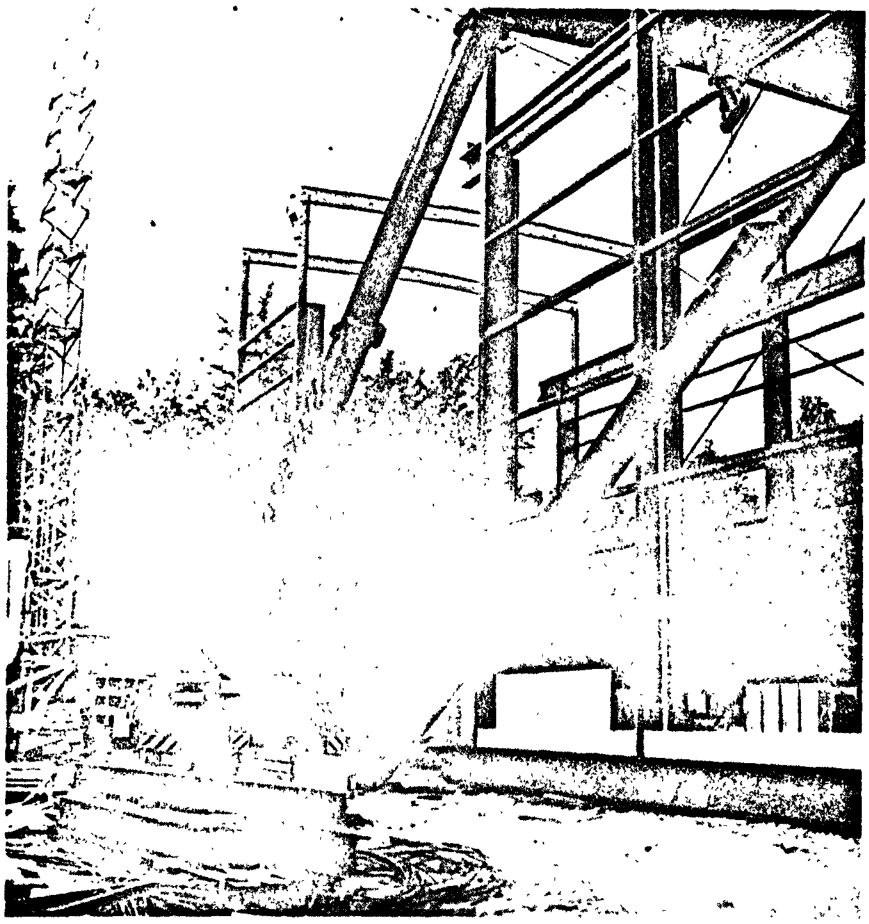


Fig. 3: Erection of steel construction with mobile crane

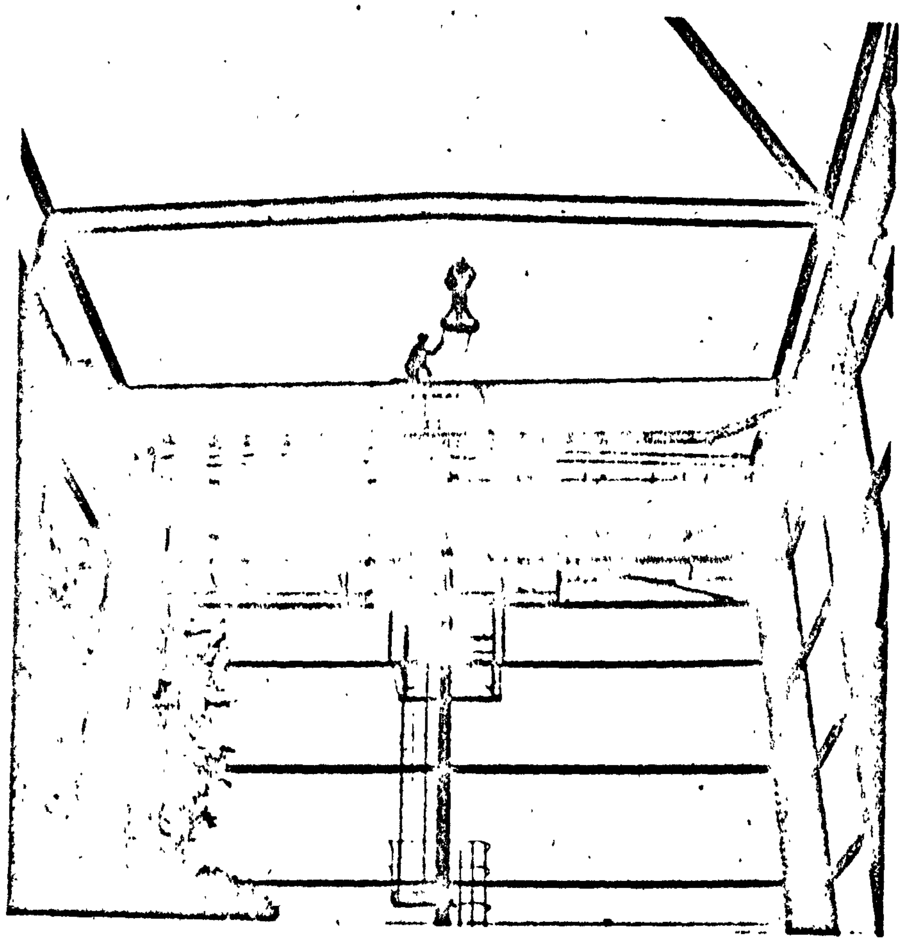


Fig.4: Mounting the 126 Mp-hangar crane

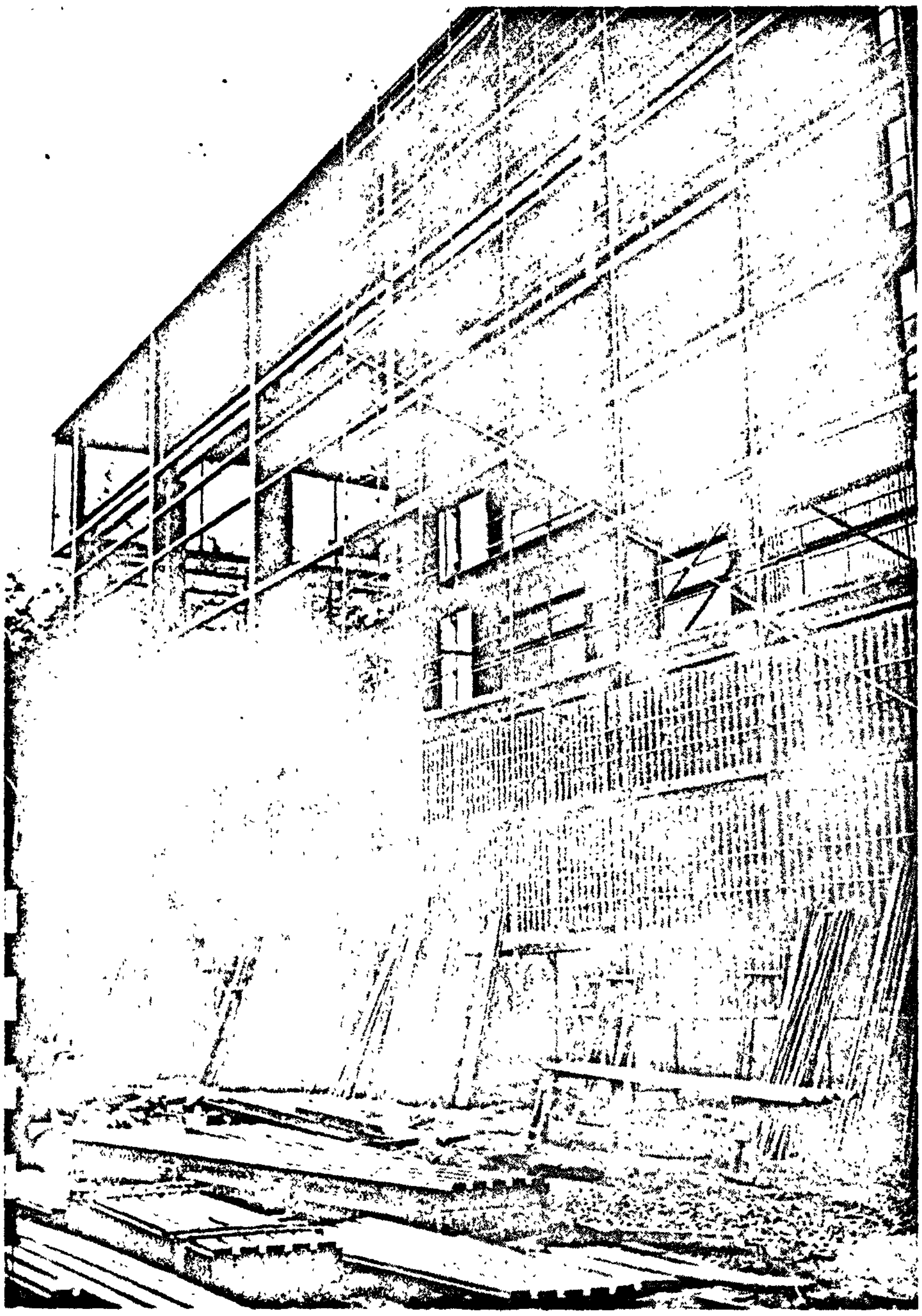


Fig. 5: Testing Hall: internal shell of panelling under construction

	CLASSIFICATION: 11.2.1.
TITLE (ORIGINAL LANGUAGE): Dynamisk brudmekanik	COUNTRY: Denmark
	SPONSOR: Risø National Lab.
TITLE (ENGLISH LANGUAGE): Dynamic Fracture Mechanics	ORGANISATION: Risø National Lab.
	PROJECT LEADER: C. P. Debel
INITIATED: 1973 / COMPLETED:	SCIENTISTS: C. P. Debel
STATUS: In progress / LAST UPDATING: May 1978	

1. General aim:

To provide materials data and theoretical background for the assessment of arrest of fast running cracks in structures.

2. Particular objectives:

To measure the (minimum) Dynamic Fracture Toughness K_{ID} as well as Crack Arrest Fracture Toughness K_I of steels.

3. Experimental facilities and programme:

500 kN closed-up servo controlled tensile testing machine as well as electronic equipment capable of measuring the rate of fast crack extention using the conducting grid method.

4. Project status:

The K_{ID} versus crack velocity relationship of a heattreated structural steel has been obtained at various temperatures, and K_{ID} has been compared to static toughness values.

5. Next steps:

To modify specimen geometry and loading condition in order to suppress crack branching and improve load train stiffness.

6. Relationship with other projects: Comparison of dynamic and static data.

7. Reference documents:

"Dynamic Fracture Toughness Testing of Structural Steels", presented at the OECD-CSNI specialists meeting at Daresbury, UK, 22-24 May 1978; and "Experimental Evaluation of Brittle Crack Propagation Velocity - on Improved Technique", Eng.Frac.Mech., Vol 11, No 2 pp. 423-430, 1979.

8. Degree of availabity:

No limitation.

45.897

	CLASSIFICATION: 11.2.1.
TITLE (ORIGINAL LANGUAGE): Bestrålingsskader i reaktorstål	COUNTRY: Denmark SPONSOR: Risø National Lab.
TITLE (ENGLISH LANGUAGE): Analysis of the Behaviour of advanced Reactor Pressure Vessel Steels under Neutron Irradiation.	ORGANISATION: Risø National Lab. PROJECT LEADER: C. P. Debel
INITIATED: 1977 / COMPLETED:	SCIENTISTS: C. P. Debel, P. E. Becher, J. Westermann
STATUS: In progress / LAST UPDATING: February 1979	

1. General aim:

To compare the fracture toughness of steel ASTM A 533 B before and after neutron irradiation.

2. Particular objections:

Specimens cut from base plates and weldments procured from Japan, West-Germany and France will be irradiated and tested.

3. Experimental facilities and programme:

Precracked Charpy V-notch specimens will be irradiated to a doze of $2x10^{19} n/cm^2$ at 290°C. During testing at various temperatures the specimens will be loaded in slow bending and J-integral versus crack extension curves will be obtained, using the multispecimen method.

4. Project status:

Procurement of steel and machining of specimens.

5. Next step:

Irradiation.

6. Relationship with other projects:

This project is a part of a collaborate research carried out under the auspicies of The International Atomic Energy Agency.

7. Referance documents:

IAEA-176, Vienna 1975
IAEA-IWG RRPC-78/1, Vienna 1978
Progress Report on Proj. 2090/CF, Risø National Lab. February 1979.

8. Degree of availability:

No limitation.

45.897

	CLASSIFICATION: 11.2.1.
TITLE (ORIGINAL LANGUAGE): Elastisk - Plastiske Brudmekaniske Materialeprøvningemetoder	COUNTRY: Denmark
	SPONSOR: EEC-European Coal and Steel Community
TITLE (ENGLISH LANGUAGE): Elasto - Plastic Fracture Mechanics Assessment Methods	ORGANISATION: Risø National Lab.
	PROJECT LEADER: C. P. Debel

INITIATED: 1979	COMPLETED:	SCIENTISTS: C. P. Debel
STATUS: In progress	LAST UPDATING:	

1. General aim:

To evaluate the Fracture Toughness of a ductile steel employing a variety of test-piece geometries and the various analytical assessment methods available.

2. Particular abjectives:

To test Compact Tension specimens at three temperatures in the transition region under various machine control modes and to evaluate the results as indicated above.

3. Experimental facilities and programme:

500 kN closed-loop servocontrolled hydraulic testing machine and constant current equipment for DC-monitoring of initiation of stable crack growth.

4. Project status:

Machining of specimens under-way.

5. Next step:

Testing of specimens.

6. Relationship with other projects:

This project has been planned as a part of a collaborative effort in which a total of 13 laboratories throughout the EEC will participate.

7. Reference documents:

Progress report issued by the Welding Institute, UK.

8. Degree of availability:

As decided by the sponsor, The European Coal and Steel Community.

	Classification 11.2.1 11.2.2
Title 1 Strukturel pålidelighed	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 Structural Reliability	Project leader: P.E. Becher
Initiated: 1970 Completed: Status: In progress Last updating:	Scientists: P.E. Becher H.E. Kongsø S. Weber

1. General aim: To develop methods for evaluation of the reliability of structural components.

2. Particular objectives: Development of computer codes, based on probabilistic methods, for evaluation of the reliability of primary components in light water reactors, specifically the steel pressure vessel.

3. Experimental facilities:

4. Project status

As a supplement to the computer code PEP 706 for calculation (by Monte Carlo with Importance Sampling) of the failure probability of a steel pressure vessel an analytical program ANPEP was developed. ANPEP makes a numerical integration of the failure integral by means of discreetizing of all the parameters in the failure criteria. ANPEP has proved to be much faster and easier to work with than PEP 706 and has even been able to take into account correlated variables without making exessive demands on computer memory or time.

The computercode PFM 690 for calculation (by Monte Carlo) of the statistical crack growth based on Paris's formula has proved the mathematical unstability of this formula when using the most recent experimental data on crack growth caracteristics.

5. Next steps:

Development of statistical models for timedependent phenomenons like

a) Crack growth
b) Inservice Inspection
c) Neutron Embrittlement
d) Updating of distribution functions for appropriate parameters when the components in question has survived a major transient

Furthermore other areas/other components (steel) and other failure modes/failure criterias should be included.

6. Relation with other projects:

Closely related with system reliability and fuel reliability

7. Reference documents:

Risø-M-1650: Application of statistical linear elastic fracture mechanics to pressure vessel reliability analysis, September 1973. P.E. Becher, Arne Pedersen.

Risø-M-1918: ANPEP/V2, A computer program for calculation of the probability of failure of structures, March 1977, H.E. Kongsø, K.E. Petersen.

8. Availability The project information is freely available apart from cases, where commercial interests may be violated.

141-1-03/4111-65		11.2.1
TITRE COMPORTEMENT A L'IRRADIATION DES ACIERS CREUSOT-LOIRE POUR CUVES PWR.		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) RESEARCH PROGRAMME ON IRRADIATION EMBRITTLEMENT OF PRESSURE VESSEL STEELS.		Organisme exécuteur CEA/DTECH Responsable
Date de démarrage 1/01/76	Etat actuel en cours	Scientifiques
Date d'achèvement 31/12/79	Dernière mise à jour 1/1/79	

1. OBJECTIF GENERAL

Le programme a pour but de déterminer le comportement sous irradiation neutronique de l'acier Creusot-Loire (type A508 CL3) utilisé pour la fabrication des cuves PWR du programme nucléaire français.

2. OBJECTIFS PARTICULIERS.

Le programme comporte essentiellement :

- Les mesures de KCV et NDT
- Les mesures de K_1d sur petites éprouvettes
- Les mesures de J_1c
- Les mesures de K_1c sur grosses éprouvettes
- L'étude de la restauration

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME.

Les irradiations sont effectuées dans le réacteur expérimental TRITON (CEN - Fontenay-aux-Roses)

.../...

4. ETAT DE L'ETUDE.

Les résultats essentiels ayant été acquis précedemment, le programme comportait principalement quelques études spécifiques sur les effets de l'irradiation sur les aciers de cuve à savoir :

- détermination de K_{Ic} en fonction de l'irradiation. La maquette a été mise au point et testée dans le réacteur TRITON et la cellule chaude correspondante est opérationnelle.
- détermination de l'influence du flux neutronique sur la fragilisation. Le niveau de flux paraît effectivement jouer un rôle dans le processus de fragilisation mais cette tendance doit être confirmée.
- mise au point d'une maquette permettant d'appréhender les doses de dommages reçues d'une part par la paroi de la cuve et d'autre part sur les éprouvettes de surveillance au niveau de l'écran thermique.
- effet du recuit thermique sur les effets de l'irradiation. Quelques résultats importants sont rassemblés en (1).

5. PROCHAINES ETAPES.

Poursuite du programme annoncé en (4).

6. RELATIONS AVEC D'AUTRES ETUDES.

Sans objet.

7. DOCUMENTS DE REFERENCE.

- Irradiation effects on reactor pressure vessel steels - influence of flux and irradiation annealing behaviour. Specialist Meeting AIEA VIENNE 1979 26 Février au 1er mars.

  P. PETREQUIN et al.

8. DEGRE DE DISPONIBILITE.

Disponible.

145-3-01/4111-104		11.2.1
TITRE COMPORTEMENT DES LIGNES DE TUYAUTERIES LORS D'UN ACCIDENT.		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) BEHAVIOUR OF PIPINGS IN THE EVENT OF AN ACCIDENT		Organisme exécuteur CEA/DEMT Responsable
Date de démarrage 1/03/77	Etat actuel en cours	Scientifiques B.VRILLON (DEMT)
Date d'achèvement 31/12/82	Dernière mise à jour 1/1/79	

1. OBJECTIF GENERAL

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 auxilaires. Les essais sont réalisés dans des conditions proches de l'échelle réelle.

2. OBJECTIFS PARTICULIERS.

Le programme comporte essentiellement, outre la mise au point du dispositif expérimental :

- la détermination des critères de fouettement.
- la mesure des efforts de réaction.
- la prise en compte des interactions possibles.
- le dimensionnement des supportages et autres limiteurs de débattement.
- la tenue des organes d'isolement.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME.

Boucle HUREPOIX (CEN-SACLAY)

.../...

4. ETAT DE L'ETUDE.

Les dispositifs de déclenchement de la rupture se sont révélés très difficiles à mettre au point et les premiers résultats ont mis en évidence des dispersions importantes.
L'ensemble de la chaine de mesure a été calibrée par des essais statiques à froid.

5. PROCHAINES ETAPES.

Détermination des critères (fin 79)
Dimensionnement des supportages et comportement des organes d'isolement (fin 80).

141-1-06/4111-6-13		11.2.1
TITRE COMPORTEMENT DES ACIERS POUR CUVES : TENUE A L'IRRADIATION - PROGRAMME COORDONNE DE L'AIEA		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) RESEARCH PROGRAMME ON IRRADIATION EMBRITTLEMENT OF PRESSURE VESSEL STEELS. AIEA COORDINATED PROGRAMME.		Organisme exécuteur CEA/DTECH Responsable
Date de démarrage 1/01/78	Etat actuel En cours	Scientifiques
Date d'achèvement 31/12/80	Dernière mise à jour 1/01/79	

1. OBJECTIF GENERAL

L'Agence Internationale pour l'Energie Atomique (AIEA) organise un programme coordonné d'irradiation d'aciers de cuves de réacteurs PWR.

Il est prévu d'examiner dans au moins 5 pays le comportement à l'irradiation de 2 aciers français, 2 aciers japonais, et 2 soudures.

2. OBJECTIFS PARTICULIERS.

Il est prévu tout d'abord de fournir et caractériser avant irradiation un acier et un joint soudé, d'origine française qui seront irradiés dans les différents pays.

Les aciers français, les aciers japonais et les soudures seront ensuite caractérisés du point de vue de l'irradiation (Charpy V, Traction, J_1c, Charpy préfissuré)

3. INSTALLATIONS EXPERIMENTALES.

Les irradiations seront effectuées dans le réacteur expérimental de TRITON (CEN-Fontenay).

.../...

TNO - Metaalinstituut	CLASSIFICATION: 11.2.1
TITLE: Literatuurstudie: (1) Invloed van omgevingscondities op vermoeiingsscheurgroei. (2) Invloed van veroudering op de mechanische materiaaleigenschappen	COUNTRY: THE NETHERLANDS
TITLE (ENGLISH LANGUAGE): Literature study: (1) Influence of Environment on Fatigue crackgrowth. (2) Influence of aging on fracture related material properties.	SPONSOR: Ministry of Social Affairs ORGANIZATION: TNO-Metaalinstituut PROJECTLEADER: H.J.M. van Rongen
INITIATED : June 1974 / LAST UPDATING : May 1978 STATUS : completion / COMPLETED : Oct. 1977	SCIENTISTS: H.J.M. van Rongen, G.H.G. Vaessen

General aim

To improve the understanding of material behaviour during lifetime of nuclear vessels and other pressurized components as influenced by aquous environment and aging effects. A "state of the art" report will be submitted.

Particulare objectives

To review the knowledge in the areas
- influence of reactor coolant on fatigue crackgrowth in LWR pressure vessels, with particular attention to chemical composition of the coolant, type of pressure vessel steel, frequency, amplitude and threshold phenomena. Interpretation will be in fracture mechanics terminology.
- phenomena which deteriorate the fatigue and fracture related material parameters of nuclear grade pressure vessel steels, such as thermal aging and high strain aging.

Experimental facilities and program: Not applicable

Project status

Progress to date: final report submitted on Oct. 21 1977.

Relation with other projects: -

Reference documents:

1. Embrittling/strengthening phenomena
2. Environmental effects on fatigue crack growth in relation to nuclear reactor vessel steels under relevant conditions by H.J.M. van Rongen and G.H.G. Vaessen. nr. 77M/93/09913/ROH/VAE/KAG/ROS.

Degree of availability:

Through the Ministry of Social Affairs, C.R.V., Postbus 69, Voorburg, The Netherlands.

Budget: HFL. 50.000,--

Personnel: 2 scientists.

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

NIL-MI-TNO	CLASSIFICATION: 11.2.1/ 11.2.3
TITLE: Literatuurstudie van materiaalparameters die nodig zijn bij de interpretatie van defect-localisatie met AE-apparatuur.	COUNTRY: THE NETHERLANDS
	SPONSOR: Ministry of Social Affairs and others
TITLE (ENGLISH LANGUAGE): Literature survey of material parameters which are necessary for the diagnosis of defects localized with AE techniques.	ORGANIZATION: NIL-MI-TNO
	PROJECTLEADER: Kloots
INITIATED: Sept. 1974 — LAST UPDATING: May 1978	SCIENTISTS: Boerstoel, v.d. Brink
STATUS: Completed — COMPLETED: June 1978	

General aim

This particular survey is a preliminary study of effects to be included in an extensive study which will be initiated in the future and which can be characterized as follows:

"Evaluation of the practical application of Acoustical Emission (AE) techniques during construction, testing and operation of welded constructions, in particular pressurized components to be used in the energy and process industry in order to improve safety, reliability and economic construction of components".

Optimum utilization of AE-apparatus for defect localization and diagnosis is only possible if sufficient data on the AE behaviour of constructural material are available (AE material parameters). This study comprises the use of AE material parameters for nuclear vessels and their application to the diagnosis of defects localized in experiments.

Particular Objectives

Study and inventarization of AE research in literature.
Visits to industry and institutes in the Netherlands.

Experimental facilities and program: none

Project status: completed

Next steps: -

Relation with other projects: see under General aim.

Reference documents: NIL-lastechniek, 40e jaargang, no. 5, mei 1974.
NIL-lastechniek "Literatuurstudie betreffende de toepassing van akoestische emissietechniek bij materiaalonderzoek aan drukvatstalen" samengesteld door Dr. B.M. Boerstoel en Dr. ir. S.H. van den Brink, rapport nr. 77M/93/03465/BRI/KAG

Degree of availability: Through Nederlands Instituut voor Lastechniek, Den Haag.

Budget: Approx. Hfl. 30.000,--

Additional information:

Survey has been executed and completed by Metaal Instituut TNO.

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

PROJECT TITLE: Fracture Behaviour of Structural Steels	CSNI INDEX NUMBER: 11.2.1
SPONSORING COUNTRY: United Kingdom	ORGANISATION: RNPDL Risley
DATE INITIATED: 1978 DATE COMPLETED:	PROJECT LEADER: A Cowan

Description:

1. General Aim

   To apply fracture mechanics concepts for assessing structural integrity of pressure retaining components.

2. Particular Objectives

   i. To investigate factors influencing ductile crack initiation and growth in LWR materials.

   ii. To investigate ductile tearing instability criteria.

   iii. To examine methods of measuring crack arrest fracture toughness.

3. Experimental Facilities

   Equipment for failure tests on 1.5 m dia x 36.5 m long x 75 mm thick pressure vessels. Laboratory test equipment of up to 1000 KN (static) /500 KN (dynamic) capacity.

4. Project Status

   i. Experimental methods are being developed to compare single and multiple specimen techniques for monitoring ductile crack initiation and resistance to growth. Techniques under investigation are AC potential drop, DC potential drop, unloading compliance and multiple specimen interrupted testing. Initial comparisons are being made using mild steel specimens.

   ii. A programme to study factors influencing elastic-plastic fracture toughness has been defined and specimens are currently being manufactured from a 150 mm thick A533B steel plate. The programme will examine:-

   a) Effect of system compliance:- 40 mm thick compact (CT) and 20 mm thick edge notched bend (SENB).

   b) Effect of specimen geometry on initiation fracture toughness and R curves:- 10-100 mm thick CT, 20 mm SENB, 20 mm centre cracked tension (CCT), 20 mm double edge notched tension (DENT). All tests will be at upper shelf temperatures.

   c) Degradation of upper shelf toughness by strain ageing:- 40 mm CT.

   iii. The tearing instability programme has been detailed. Tests will be made on 12.5 mm thick 7075 Al alloy to study effects of specimen geometry (CT, CCT, DENT) and system compliance on instability. Material should be available by late 1979.

/Contd.

(iv) Crack arrest work has been limited to an examination of ASTM-candidate procedures for measuring crack arrest toughness.

Reference Documents

Internal documents.

	CLASSIFICATION: 11.2.1
TITLE (ORIGINAL LANGUAGE): METAL FRACTURE (STABILITY)	COUNTRY: UK
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: NPDL SPRINGFIELDS
	PROJECT LEADER: DR B TOMKINS
INITIATED: 1971 / COMPLETED: 1976	SCIENTISTS:
STATUS: / LAST UPDATING: MAY 1979	

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

1. Internal documents.

Conference on The Influence of Environment on Fatigue - Joint Conference. The Institution of Mechanical Engineers and Society of Environment Engineers. London 18-19 May 1977, B Tomkins.

3. The Role of Mechanics in Corrosion Fatigue - to be published in Metals Science in 1979, B Tomkins.

	CLASSIFICATION: 11.2.1
TITLE (ORIGINAL LANGUAGE): MECHANICS OF PLANE STRAIN FRACTURE	**COUNTRY:** UK
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	**ORGANISATION:** NPDL SPRINGFIELDS
	PROJECT LEADER: DR B TOMKINS
INITIATED: 1972 / **COMPLETED:** 1978	**SCIENTISTS:**
STATUS: / **LAST UPDATING:**	

DESCRIPTION

1. GENERAL AIM

To recognise and assess modes of failure of nuclear reactor pressure.

2. PARTICULAR OBJECTIVES

To measure the deformation in a thick cracked plate up to the point of fracture.

3. EXPERIMENTAL FACILITIES AND PROGRAMMES

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 studies 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

1. Internal documents

2. Conference on The Influence of Environment on Fatigue Joint conference The Institution of Mechanical Engineers and Society of Environment and Engineers London 18-19 May 1977, B Tomkins.

3. The Role of Mechanics in Corrosion Fatigue - to be published in Metals Science in 1979. B Tomkins.

	CLASSIFICATION: 11.2.1
TITLE (ORIGINAL LANGUAGE): METAL FRACTURE (CRACK GROWTH)	COUNTRY: UK
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: NPDL SPRINGFIELDS
	PROJECT LEADER: DR B TOMKINS
INITIATED: 1971 / COMPLETED:	SCIENTISTS:
STATUS: / LAST UPDATING: MAY 1979	

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 speciments 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. In addition to crack growth studies, information is being collected on the conditions for crack initiation on thermally shocked surfaces.

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 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.

c) A wide plate testing rig is being built to examine the approach to final fracture by tearing instability in standard sections and effects of standard constraints on future development will be studied in particular.

REFERENCE DOCUMENTS

Internal documents

Classification

11:2.1

Title 1 VESSEL FAILURE TESTS	COUNTRY UNITED KINGDOM SPONSOR UKAEA ORGANIZATION RDML 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.

TITLE (ORIGINAL LANGUAGE): Assessment of upper shelf toughness of PWR pressure vessel steels (M 3.7)		CLASSIFICATION: 11.2.1
		COUNTRY: UNITED KINGDOM
		SPONSOR:
TITLE (ENGLISH LANGUAGE):		ORGANISATION: UKAEA (Harwell)
		PROJECT LEADER: B Watkins (Risley) J B Sayers (Harwell)
INITIATED: March 1978	COMPLETED:	SCIENTISTS:
STATUS: In progress	LAST UPDATING: March 1979	

Background

Pressure vessel integrity is at present founded on the established design principles of linear elastic fracture mechanics. However the high toughness to yield strength ratio of A533B and A508 pressure vessel steels invalidates Lefm testing procedures for small scale tests and recourse is made to elastoplastic testing methods in order to provide an indirect measurement of K_{Ic}. Such procedures are not unilaterally accepted and at present several non consistent techniques are employed at different establishments throughout the world. The use of such techniques is critically dependent on the definition of crack initiation and the first stages of stable crack extension. The influence of metallurgical factors such as micro-structure and embrittling processes on fracture toughness are poorly quantified although it is accepted that the fracture micro-mechanisms of void initiation growth and coalescence will be controlled by these parameters.

The limited evidence on the micro-structural characterisation and embrittlement susceptibility of the pressure vessel steels can be summarised as follows:-

i Qualitative studies have shown that large variations in micro-structure of A533B can occur as a function of position within the plate. However no quantitative characterisation of A533B plate or A508 forging or welds exist.

ii A533B is susceptible to temper embrittlement and, although this is known to be associated with trace elements and heat treatment in a critical temperature range, the boundary conditions for embrittlement to occur are poorly defined.

iii Dynamic strain ageing of A533B has been noted in fracture tests conducted at typical PWR operating temperatures (~ 300°C). This is of importance to the in service toughness and fatigue crack growth characteristics of the vessel steels and needs to be properly understood.

iv Reheat cracking has been experienced during fabrication of PWR pressure vessels, which is confined to the heat affected zones (HAZ's) in the welded regions. Again the factors governing such cracking are not understood but the major parameters are known to be micro-structure, composition and residual stress.

M 3.7 contd

In view of this background it is considered essential that a thorough investigation be carried out to determine the role of micro-structure and embrittling processes on the fracture toughness of pressure vessel steels.

Objectives

i To comprehensively characterise the micro-structure of commercially produced A533B plate (and A508 forging*) as a function of position and orientation.

ii To provide a micro-structural characterisation of any material used in PWR corrosion fatigue programme.

iii To investigate susceptibility of A533B (and A508*) to temper embrittlement and the effect on fracture toughness, as a result of slow cooling from post weld heat treatment temperatures and long term ageing at reactor ambient temperature.

iv To investigate sensitivity of temper embrittlement to simulated HAZ micro-structures.

v To determine the susceptibility of A533B (and A508*) to stress relief cracking under simulated post weld heat treatment cooling conditions.

vi To investigate the susceptibility of commercial A533B plate (and A508 forging*) to dynamic strain ageing, its dependence on heat treatment condition and its effect on fracture toughness.

vii To interact with programmes at Risley Laboratories investigating possible geometry effect on upper shelf toughness.

*Work on A508 forging material subject to availability of a commercially produced forging.

Materials

This programme will be conducted using 148 mm thick A533B Class 1 steel commercially produced by Marrel Frere. Sufficient material for this programme is held in stock at Harwell. Work on A508 Class 3 material will be subject to availability of a commercially produced forging.

Instructions

1. Comprehensively characterise the micro-structure of A533B plate and A508 forging as a function of position and orientation using optical and electron microscopy supplemented by micro-analytical techniques.

2. Determine susceptibility of A533B and A508 in the as received and simulated HAZ condition to embrittlement associated with trace element segregation using a combination of impact and fracture mechanics mechanical testing and surface analytical techniques.

3. Investigate propensity to stress relief cracking as a function of heat treatment temperature, heating and cooling rates.

4. Determine critical conditions of micro-structure, strain rate and temperature for dynamic strain ageing and associated effect on fracture toughness.

M 3.7 contd

5. Following discussions with interested parties at Risley Laboratories, investigate specimen geometry dependence of fracture toughness of A533B and the relationship with fracture micro-mechanisms.

Facilities

The equipment required to carry out this programme is either available or presently being developed within Metallurgy and Materials Development Divisions at Harwell.

	CLASSIFICATION: 11.2.1
TITLE (ORIGINAL LANGUAGE): Elastic/Plastic fracture toughness (M 3.3)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Risley)
	PROJECT LEADER: B Watkins (Risley)
INITIATED: March 1978 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: March 1979	

Background

Materials used in the fabrication of PWR pressure vessels are selected, so that, for most of its life, the pressure vessel operates in the temperature range where any postulated fracture would occur when the material is in the upper shelf (ductile) condition. Elastic/plastic fracture mechanics are being developed to quantify upper shelf toughness. Irradiation embrittlement causes a reduction in the level of upper shelf toughness. In addition, work on a wide variety of steels has shown that fracture toughness decreases with increasing upper shelf temperature so that, at operating temperature, fracture toughness values can be significantly lower than those assumed in current fracture mechanics evaluations of pressure vessel integrity. The degree of (or lack of) conservatism in current design codes is critically dependent on the definition of fracture toughness for materials which fracture after ductile initiation and on the amount of credence which can be attached to the apparent increase in fracture resistance resulting from ductile crack extension. It is essential to resolve the significance of ductile initiation/crack extension and to understand the mechanism(s) associated with degradation of upper shelf toughness in both unirradiated and irradiated materials.

Objectives

1. To demonstrate the significance of ductile crack extension in laboratory fracture mechanics testing.

2. To investigate degradation in upper shelf fracture toughness - (unirradiated material)

   a. Examination of relationship between fracture toughness and mechanical properties.

   b. Examination of strain ageing phenomena.

   c. Examination of factors highlighted on completion of evaluation of current static and dynamic fracture toughness criteria and the effect of metallurgical factors on toughness.

3. To investigate degradation in upper shelf fracture toughness - irradiated materials.

M 3.3 contd

Programme

1. Characterise the effects of test piece geometry, test machine compliance and rate of loading on crack stability, using small-scale elastic/plastic fracture mechanics techniques for materials showing high and low resistance to crack growth.

2. Test simple geometries to assess effects of crack size, type of pressurisation and rate of pressurisation on critical conditions for crack stability and examine the relationship between laboratory and structural tests.

3. Obtain base-line upper shelf fracture toughness and conventional mechanical properties data on pressure vessel materials. Examine effects of metallurgical stability (ageing/strain ageing).

4. Feasibility study for irradiation and post-irradiation testing of fracture toughness specimens.

Facilities

Universal testing machines and pressurising facilities are available at Risley. Depending on the outcome of the feasibility study, a more elaborate post-irradiation testing facility may be required at a later date.

	CLASSIFICATION: 11.2.1
TITLE (ORIGINAL LANGUAGE): Fatigue corrosion of pressure vessel steels (M 2.5A)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Harwell and Springfields)
	PROJECT LEADER: B Watkins (Risley) J B Sayers (Harwell)

INITIATED:	COMPLETED:	SCIENTISTS:
March 1978		
STATUS: In progress	LAST UPDATING: March 1979	

Background

Corrosion fatigue has been shown to be an important aspect of the potential growth of defects in PWR pressure vessels and hence of the safety case for the vessels. Whilst much data have already been obtained outside the UK, a large number of variables need to be covered including stressing parameters (eg frequency, wave shape, K, R ratio and the interaction of different cycles), metallurgical factors (eg weld structures and variations in plate with depth in the plate) and water chemistry (eg pH). Additional work is required to evaluate these factors adequately and to obtain data on typical steels and welds to be used in a UK PWR programme.

A facility is almost completed at Harwell to study the corrosion fatigue of steel in pure oxygenated water. It consists of four separate units testing precracked 1-in or 2-in thick CMS specimens. It will be assigned to test PWR pressure vessel steels in pure deoxygenated water with no adaptation, but will require significant modification to handle boric acid-lithium hydroxide dosed PWR primary water. The proposed programme will investigate as required the relevant stressing variables and water chamistry parameters for a range of materials covering the basic plate, welded plate and the significance of the composition and structure of the basic plate within the specified ranges.

Objectives

The principal objectives will be to obtain data on:

1. typical steel and weldments to be used in a UK PWR programme, and

2. parameters not adequately covered by published work.

Experimental Programme

The detailed programme will typically involve experiments on the basic plate over the first 12-18 months to produce published work and then a major emphasis on welds. Initially studies will have to be confined to high purity deoxygenated water, but later, PWR water chemistries will be covered after the necessary alterations to the water treatment plant have been made.

M 2.5A contd

Metallurgical support from a specialist metallurgist team will be available to characterise the steels and to evaluate the critical metallurgical features in the fractured specimens.

Facilities

The corrosion fatigue facility at present being commissioned at Harwell will be used with modification, probably over the first year, of the water treatment plant to produce PWR primary water.

Reference Documents

	CLASSIFICATION: 11.2.1
TITLE (ORIGINAL LANGUAGE): Corrosion fatigue crack growth (M 2.5)	COUNTRY: UNITED KINGDOM
	SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA (Harwell, Risley and Springfields)
	PROJECT LEADER: B Watkins (Risley) J B Sayers (Harwell)
INITIATED: March 1978 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: March 1979	

Background

Structural integrity of LWR pressure vessels is based on fracture mechanics principles. Planar defects which exist in the pressure vessel can extend under imposed operational cyclic stresses and it is essential to demonstrate that such defects will not grow to a critical size for fast fracture during the service life of the pressure vessel. Fatigue crack growth laws presented in ASME XI Appendix A are used to estimate the growth of defects identified during periodic inspections. Current ASME XI crack growth laws are based on measurements of fatigue crack growth rates at high frequency and zero mean stress (low R ratio). Studies in the US and Japan have shown that PWR pressure vessel materials are susceptible to corrosion fatigue whereby crack growth rates can be increased significantly when tests are made in reactor water chemistry at low frequency and high mean stress (high R ratio). The implication of this work is that the ASME XI Appendix A crack growth analysis may not provide a conservative upper bound for crack growth. Corrosion fatigue crack growth data is being obtained using high pressure/temperature systems. Such tests are, of necessity, long term, and exploratory tests using simpler, ambient pressure/lower temperature testing systems are considered useful in obtaining both a basic understanding of corrosion fatigue mechanisms and an indication of trends in corrosion fatigue response at full temperature/pressure. A test facility suitable for full temperature/pressure work is available at AERE. Facilities for low temperature/pressure work are available at Risley and Springfields Laboratories.

Objective

To determine corrosion fatigue crack growth rates of PWR pressure vessel materials at ambient pressure/low temperature.

Programme

1. Design and construct loop for controlled PWR water chemistry.

2. Test A533B steel, A508 forging and weldments at ambient temperature and 90°C to assess effects on crack growth rates of PWR water, cyclic frequency and R ratio.

Facilities

Existing facilities and loops at Risley, Harwell and Springfields.

Classification 11.2.1/2

Title 1 Pressure vessel integrity

Title 2

Initiated 1974

Completed

Status Continuing

Last updating

Country U.K.

Sponsor CEGB

Organisation CERL Leatherhead

Project Leaders

I. L. Mogford

1. General aim

To assess and develop safety analyses for pressurised structures.

2. Particular objectives

To carry out and assess fracture mechanics analyses of reactor primary circuit, including final fast fracture and sub critical crack growth involving fatigue and environmental interactions.

3. Experimental facilities

Standard lab. mechancial testing plus laboratory operational environmental cracking studies.

4. Project status

Fracture mechanics design analyses and defect assessment routes using elastic-plastic techniques have been developed and are suitable for use with primary and secondary loading. Warm-prestressing effects and slow crack growth analyses can be incorporated. The fracture toughness of A533 B steel plus weldments has been studied as a function of temperature and specimen geometry. The mechanisms responsible for accelerated fatigue crack growth of pressure vessel steels in hot water are being investigated. Particular attention is being given to microstructural cracking mechanisms and fatigue cycle shape. A high pressure autoclave fatigue crack growth facility is being commissioned.

5. Reference Documents

Internal CEGB reports plus:-

"Post yield fracture mechanics theory and its application to pressure vessels" by Chell, International Pressure Vessel and Piping, 1977, Vol.5, 123.

"A combined linear elastic and post yield fracture mechanics theory and its engineering applications" by Chell, Fracture Mechanics in Engineering Practice, 1977, Applied Science Publishers Ltd., London.

"A unified approach to failure assessment of engineering structures" by Harrison and Milne, ibid.

"A procedure for assuring the significance of flaws in pressurised components" by Milne, Loosemoore and Harrison, Conf. on Tolerance of Flaws in Pressurised Components, 1978, London.

"Incorporation of thermal and residual loading into the failure assessment diagram" by Chell, ASTM STP 668, 'Elastic-Plastic fracture'.

"Effect of size on the J fracture criterion" by Milne and Chell, ibid.

"Evaluation of the defect tolerance of pressure vessels under combined thermal and pressure loads", by Milne, S.M.I.R.T. 5, Berlin, 1979.

"Fracture mechanics approach to predicting the effects of warm prestressing and its application to pressure vessels" by Chell, ibid.

"Failure analysis in the presence of ductile crack growth" by Milne, Materials Science and Engineering, 1979.

"Charpy energy transitions for weld metal and heat affected zone in SA533B cl. 1 plate" by Milne and Curry, I.C.M.3, 1979.

"The effect of frequency and temperature on environmentally assisted fatigue crack growth below K_{1SCC} in steels" by Atkinson and Lindley, Conf. on "The influence of environment on fatigue", Inst. Mech. Engineers, 1977, London.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78 - 31.12.78	11.2.2	RS 262
Vorhaben/Project Title		Land/Country
Untersuchungen zur Zeitfestigkeit im 2-achsigen Spannungsfeld mit reinem Zug und Druck Study on the Finite Fatigue Strength in a Biaxial Stress System Subjected to Purely Tensile and Compressive Loading		FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Fried. Krupp GmbH Krupp Forschungs- Institut
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.05.77	30.4.79	Dipl.-Ing. Spandick
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	December 1978	

1. General Aim

The codes for the design of pressure vessels permit the yield point to be exceeded considerably in the case of secondary stresses. Pressure vessel components are therefore often to be designed for finite fatigue strength. In doing so, multi-axial cyclic thermal deformations, partly or even fully prevented, must be allowed for up to the plastic range. The design practice according to the ASME code is based on uni-axial, mechanical tests and uses high safety margins. Theoretical studies known from literature give rise to serious doubts as to whether it is admissible -

- to compare multi-axial loads in the higher temperature range with uni-axial loads via the reference stress, e.g. according to v. Mises, and
- to treat thermal cycles merely as mechanical cycles at an equivalent temperature.

For safety reasons it is therefore necessary to more closely investigate the loading limits for the materials in question.

2. Particular Ojectives

In order to obtain experimental data in support of the design of pressure vessel components subjected to thermal multi-axial cycles loading this research project provides for thermal biaxial-load-cycle-tests to be carried out on selected materials. In these tests the fatigue life as a function of the restrained deformations is to be determined. Furthermore, the stress-strain curve is to be plotted as a Bauschinger loop for each temperature cycle.

3. Research Program

As set forth in the application the research project centers on the following main program points:

3.1. Evaluation of publications on recent work in the field of low-cycle fatigue throughout the term of the project.

3.2. Design and manufacture of suitable test facilities specially adapted to meet the project objectives.

3.3. Establishing the testing procedure

3.4. Testing

3.4.1. Tests on the material TTSTE 29
Heating to 200 °C, 300 °C, 400 °C, 500 °C, holding time 0 min., cooling to approx. 50 °C.

3.4.2. Tests on the material No. 4541 (stainless steel) like 3.4.1.

3.4.3. Tests on the material No. 4948 (stainless steel) like 3.4.1., however, only for one temperature cycle

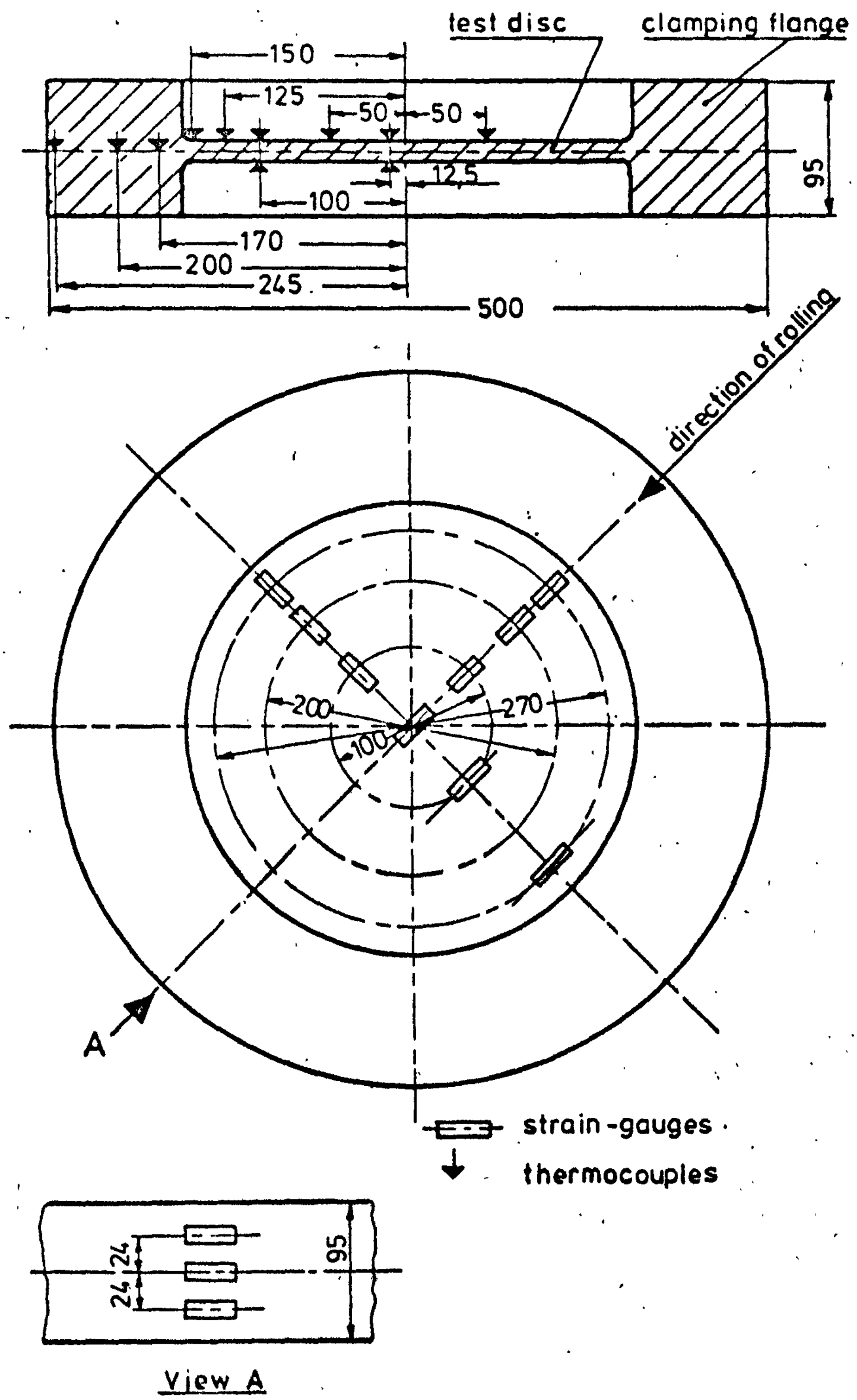
3.4.4. For comparison purposes specimens of the three above-mentioned materials will be subjected to mechanical uni-axial cycles at a maximum deformation of 5 °/oo, a constant testing temperature of 300 °C and a holding time of 0 min.

3.4.5. Tests as under 3.4.4. but with a holding time of 20 min.

3.5. Evaluation of the individual test results.

4. Experimental Facilities, Computer Codes

In the present drawing it's shown the assigned disc and its clamping-flange for the various experiments. The position of thermocouples and strain-gauges are appropriate marked.



5. Progress to Date

ad 3.1.

The following publications touching on the project have been evaluated:

- Thermal fatigue testing system for structural materials, P.I. Stocker, W.L. Morris jr. and W.M. Robertson (Science Center, Rockwell USA)
- Behaviour of annealed type 316 stainless steel under monotonic and cyclic biaxial loading at room-temperature I.R. Ellis, D.N. Robinson and C.E. Pugh (Oale Ridge National Laboratory)
- Special issue of "Nuclear Engineering and Design", Volume 51 (1978) No. 1, concerning life prediction of components and interactives among creep, cyclic plasticity and recovery

Although the described tests aren't comparable with those intended with the present project, so it's possible to compare the results very well, and the arrangements of computations and the testparameters provide important informations.

ad 3.4.1.

Up to now the test discs were subjected to cycles between the temperature limits of 50 °C - 300 °C, 50 °C - 400 °C and 50 °C - 500 °C until they fractured. The cyclic tests were evaluated using subsequent tests on material under load such as a hardness test, tensile test, impact tests and structure studies.

ad 3.4.2.

The cyclic tests were also carried out between the lower temperature limit of 50 °C and the upper limits of 300 °C, 400 °C and 500 °C on the discs made of austenitic material. In the evaluation the changes in the material properties were determined after cyclic loading with hardness, tensile and impact tests.

ad 3.4.3.

The specimen was made and thermocouples and strain gauges were attached to it. The test with one cycle between the temperature limits of 50 °C and 400 °C is being prepared.

ad 3.4.4.
Specimens were made for the uniaxial tests, the results of which are to be compared with those of the biaxial tests. In setting up the test apparatus it was found that the cyclic loading could not be controlled accurately enough over the measured piston path. A new setup had to be deviced where deformation of the specimen is measured in the furnace itself.

6. Results

Some interesting results are presented in the following:

Load cycle attained

Material	Cycle [°C]	Load cycle up to first incipient fracture	Load cycle up to fracture
TTSTE 29	50 - 300	5023	5388
	50 - 400	821	1127
	50 - 500	315	477
X10CrNiTi 18 9	50 - 300	6880	8053
	50 - 400	575	753
	50 - 500	231	542

It can be seen that from the time of the first incipient fracture (certainly detectable on a repeat test) until fracture, i.e. leakage, enough cycles can be withstood. This illustrates the significance of repeat testing and also shows that there would be sufficient time left to take appropriate remedial action

The mechanical tests on the discs following thermocyclic loading provide the following results, e.g. for the average range (measured, in the case of ferrite, in the direction of rolling):

Hardness test

Material	Hardness value [HV 30]			
	unloaded	50-300 °C	50-400 °C	50-500 °C
TTSTE 29	128	157	182	149
X10CrNiTi 18 9	200	181	201	200

Tensile test

Material	Cycle	$\sigma_{0,2}$ [N/mm²]	σ_B [N/mm²]	A_5 [%]	Z [%]
TTSTE 29	unloaded	263	430	38	75
"	50-300 °C	338	471	31	75
"	50-400 °C	338	477	29	75
"	50-500 °C	343	483	28	75
X10CrNiTi 18 9	unloaded	250	545	55	70
"	50-300 °C	296	580	54	70
"	50-400 °C	303	579	52	70
"	50-500 °C	356	599	52	65

Impact test

Material	Impact energy [Joule]			
	unloaded	50-300 °C	50-400 °C	50-500 °C
TTSTE 29	252	43	40	25
X10CrNiTi 18 9	138	148	150	115

As expected with the tensile test there was an increase in the tensile yield point after cyclic loading. The increase is about 30 % with TTSTE 29. The 20 % increase with austenite is not quite so pronounced. Knowledge of this change in strength is of particular importance in designing the liner anchorage in prestressed concrete, for example, as the loading of the anchors depends on the tensile yield point of the liner material. Fortunately the high elongation at fracture A_5, desirable for good deformation behaviour, decreases only slightly. Here, too, ferrite shows a somewhat greater decrease than austenite.

The different materials made themselves particularly felt in the result of the impact tests. Practically no change in impact energy was established with austenite. With ferrite the drop in impact energy is considerable although it does not fall below the admissible values. The results of the hardness test, particulary those for ferrite, also indicate slight embrittlement.

7. Next Steps

ad 3.1.
Evaluation of any new publications, concerning this project

ad 3.4.3.
Test on a disc at a temperature cycle between 50 °C and 400 °C.

ad 3.4.4.
Uniaxial tests.

ad 3.5.
More detailed evaluation of results.

8. Relations to Other Projects

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9. References

-

10. Degree of Availability

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78 - 31.12.78	11.2.2	RS 275
Vorhaben/Project Title		**Land/Country**
Untersuchungen zur Übertragbarkeit von Kennwerten für den Bruch und Rißfortschritt von Proben auf Bauteile im Hinblick auf die Sicherheit von Kernkraftwerkskomponenten Transferability of Fracture and Crack Propagation Characteristics from Specimens to Structures Considering the Safety of Components of Nuclear Power Stations		FRG **Fördernde Institution/Sponsor** BMFT **Auftragnehmer/Contractor** Bundesanstalt für Materialprüfung (BAM Abteilung 1 1000 Berlin 45
Arbeitsbeginn/Initiated	**Arbeitsende/Completed**	**Leiter des Vorhabens/Project Leader**
01.08.1977	31.07.1980	Dr. AURICH
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

Clarification of the influence of different states of nominal stresses, as usual in pressurized components of nuclear power stations and in conventional specimens, on the fracture behaviour of material with defects. In usual specimen , an uni-axial state of nominal stress is produced, but in pressurized components a multiaxial state of nominal stresses occurs. Such components are the cylindrical parts and the bottoms of pressure vessels, containments and pressurized pipes under tension and momentum load. The induced principal stresses in such structures should be simulated in test specimen , neglecting the influence of radial stresses. The effect of defects on the local stress distributions should be analyzed and their influence on the fracture behaviour in dependence of the nominal stress state should be evaluated.

2. Particular Objectives

2.1 Development of test specimens

in which states of nominal stresses can be simulated as they might occur in pressurized components of nuclear power stations by means of a FEM-stress analysis.

2.2 Analysis of local states of stresses

near defects, especially analysis of the K-factors and J-values in dependence of the nominal stress state by means of numerical and experimental procedures. Elastic and elastic-plastic material behaviour as fas as possible should be taken into account.

2.3 Test performance
on specimens with defined defects in dependence of temperature and under a nominal stress state as it occurs in components. Comparison of these results with those on conventional uniaxial stressed specimens including fracture mechanics tests. Cleavage fracture, ductile fracture and fatigues crack propagation should be considered. The results should be represented in dependence of the nominal stress state.

2.4 Development of transferability criterias
for characteristics of fracture, crack propagation and critical temperatures from one stress state to another stress state.

3. Research Program

3.1 Test specimen development by means of stress analysis

3.1.1 Development of the specimens shape and the shape of the joints, taking into consideration the experiences of pretests which were performed on our own account.

3.1.2 Evaluation of K_I- and J_I-values, if possible.

3.2 Evaluation of σ_{ij} (r, φ, z)
for the state of small scale yielding and general yield as far as financial means are available.

3.3 Investigation of Fracture Behaviour

3.3.1 Specimen with cracks
should be investigated for clarification of the influence on the fracture behaviour under different states of nominal stresses.

3.3.2 Conventional specimens
should be investigated in addition to determinate the basic mechanical properties such as yield strength in dependence of temperature, the cleavage fracture strength and so on.

3.4 Conception for transferability
should be tried on the basis of results under 3.2 in combination with the fracture hypotheses.

4. Experimental Facilities and Computer Codes

4.1 Specimen Development
The stress analysis should be performed by the FE-programs of the SAP-family, especially NONSAP and the advanced development ADINA. The MARC-CDC-program is not regarded for the time being owing to reasons of too high costs.

4.2 Evaluation of σ_{ij} (r, φ, z)
The elastic-plastic stress analysis should be performed by the NONSAP-FE-program. Additional possibilities should be considered.

4.3 Investigation of Fracture Behaviour
For cooling down the big specimens, a special cooling device is necessary. Experience in this field is available, however, the detailed development of the device cannot be started before a clarification of the specimen geometry is reached, even so the scientific instrumentation plan needs clarification of the specimen geometry.

5. Progress to Date

5.1 Development of Test Specimens
Stress analysis on disks and plates by finite element

method for optimization the geometry of specimen. Development of a plate shaped specimen.

5.2 Analysis of Local States of Stresses

Installation of the FE-Program ADINA . Two- and three-dimensional stress analysis on bend and tension specimens with cracks for evaluating the influence of the specimen thickness on the stress intensity factor. First test calculations in the elastic-perfectly-plastic range. Development of a three-dimensional FE-network for the semi-elliptical surface crack.

5.3 Investigation of the Fracture Behaviour

Impact tests with instrumentation were performed on ISO-V-notch-specimens of steel StE 47 which were taken in and transverse to the rolling direction. Pretests were carried out on model plates in the scale 1 : 2,5 for optimization the specimen shape and the testing device. The plates of 400 x 400 x 60 millimeters dimension provided with a semi-elliptical surface crack of 140 millimeters width and maximum depth of 23 millimeters were cracked under bending in the temperature range of -100 °C to 0 °C. The ratio of the nominal stresses in direction of crack opening and in direction parallel to the crack front was about 1 : 1.

6. Results

6.1 Specimen Development

Decision for the plate shaped specimen. Optimization of the geometrical dimensions in relation to certain ratios of the nominal stresses in direction of crack opening and in direction parallel to the crack front.

6.2 Analysis of Local States of Stresses

The influence of the specimen thickness on the stress intensity factor by finite element method is quite good reproducible. A precise analysis of the stress intensity factors should be done on three-dimensional basis. The usual assumption plane strain conditions will be reached in the plate center and plane stress condition will exist on the plate surface must not be correct in any case.

6.3 Investigation of the Fracture Behaviour

Results of the impact tests on steel StE 47 demonstrate that the properties of samples taken transverse to the rolling direction are better than of those taken in rolling direction. The measured plane strain fracture toughness of the material agrees quite well with results from literature.

7. Next Steps

7.1 Development of Test Specimens

7.2 Analysis of Local States of Stresses

Three-dimensional finite element evaluations for elastic-perfectly-plastic material behaviour will be continued in the stress analysises of plates with cracks. Tests of the FE-program ADINA in the three-dimensional range and for elastic-plastic material behaviour with hardening are intended.

7.3 Investigation of the Fracture Behaviour

The test installation for the model tests shall be improved. The model tests on bend plates of StE 47 will be continued. The universal 20-MN-Testing-Machine will be installed.

8. Relation with other Projects

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9. Literature

Technical Report "Three-Dimensional Stress Analysis for the Determination of K-Values in Specimens and Structures with Cracks"

10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 30.11.78	11.2.2	RS 302
Vorhaben/Project Title: Special Stress Conditions in Pipe Bends Grenzzustände im Tragverhalten von Rohrkrümmern in Rohrleitungen		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: SDK Ingenieurunternehmen 7850 Lörrach
Arbeitsbeginn/Initiated: 1.11.77	Arbeitsende/Completed: 30.11.78	Leiter des Vorhabens/Project Leader: Dipl.Ing. H. Hofmann
Stand der Arbeiten/Status: Completed	Berichtsdatum/Last Updating: December 1978	Bewilligte Mittel/Funds:

1. General Aim

The aim of the project is to investigate the structural behaviour of pipe bends subjected to three-dimensional loading and to establish limits up to which a linear elastic calculation as well as superposition is permissible.

2. Particular Objectives

Work is carried out analytically for a variety of parameters. Three different approaches for a calculation are considered:

a) beam-theory in combination with flexibility- and stress-intensification factors for the pipe bends
b) beam-theory using "special elements" for the pipe bends
c) beam-theory for the straight and thin-shell theory for the curved elements

By using a step by step procedure the linear elastic behaviour shall be investigated and limitations shall be evaluated with respect to:

1) Applicability of the calculation procedures mentioned above with respect to real pipeline problems.
2) Limitations for a linear elastic behaviour due to
   - geometric non-linearities (large displacements)
   - material non-linearities (plasticity).

3. Research Program

3.1 Preparations for analysis, selection of suitable parameters. Analytical and experimentally determined results of the most important publications are considered.

3.2 Preparation of the calculation models

3.3 Linear-elastic analysis (deformations, stresses)

3.4 Evaluation of results

3.5 Determination of the limitations
- material non-linearities
- geometric non-linearities

3.6 Evaluation of results, documentation, report

4. Computer Codes

Within the scope of this project computer codes are used, which were tested in several applications. The codes are based on the Finite-Element-Method.

5. Progress to Date

ref. 3.1 Work was started by looking through literature and gathering all essential information (experimental and analytical) on the behaviour of pipe bends.
The underlying principles for flexibility-, stress-intensification-factors and special elements for pipe bends (e.g. Marc) were studied.
In addition calculation parameters, concerning geometric and material data, were established.

ref. 3.2 Preparations for the calculation models were finished.
3.3 Linear elastic calculations were carried out.

ref. 3.4 Parallel to the change of parameters, results were evaluated.
3.5 The essential steps in finding limitations have been made.

6. Results

Results show that the applicability of beam-theory under consideration of flexibility- and stress-intensification-

factors (e.g. ASME) is justified within certain limitations. This is true for deformations and stresses and here in particular for the distribution along the circumference of a pipe cross-section.
The final report will give more details.

7. Next Steps

ref. 3.6 Documentation and final report will be finished.

8. Relation to Other Projects

-

9. References

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10. Degree of Availability of the Report

-

Berichtszeitraum/Period 1.1.78 - 31.12.78	Klassifikation/Classification 11.2.2	Kennzeichen/Project Number RS 325
Vorhaben/Project Title Ermüdungsuntersuchungen unter simulierten Druckwasserreaktor-Bedingungen Fatigue Crack Growth Tests Under Simulated LWR-Conditions		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Materialprüfungsanstalt Stuttgart (MPA)
Arbeitsbeginn/Initiated 1.2.78	Arbeitsende/Completed 4.3.79	Leiter des Vorhabens/Project Leader Dr. Sturm
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds DM 97.377,--

1. General Aim

Technical assignment over a one year period to work in Naval Research Laboratory (NRL) Washington, DC and in Oak Ridge National Laboratory (ORNL) Oak Ridge, TN, USA, in the field of reactor safety in particular strength- and fatigue behavior of reactor vessel steels.

2. Particular Objectives

NRL: Crack growth rate studies of reactor pressure vessel steels under service conditions. Investigation of the influence of load, rise- and holdtime, defect and defect size, material, water condition (temperature and pressure), water chemistry and irradiation.

ORNL: Elastic-plastic fracture mechanics as applied to the results of the Intermediate Size Vessel Tests conducted in the HSST-Program

3. Research Program

3.1. NRL

3.1.1. Crack growth rate studies on LWR-material in autoclaves

3.1.2. Fracture toughness, new concepts of elastic-plastic fracture-mechanics

3.1.3. Crack arrest tests

3.2. ORNL

3.2.1. Intermediate Size Vessel Tests

3.2.2. Nondestructive and destructive investigations of the circumferential weld of vessel V-9

3.3. Participation in meeting

4. Experimental Facilities

(3.1.1) NRL:

- 4 autoclaves to simulate PWR conditions: 288°C, 140 bar, volume ~150 l, capacity: CT-and WOL-specimens up to 100 mm thickness, daisy-chain arrangement possible
- 1 autoclave to simulate special service conditions: 93°C, 1 bar, volume ~6 l, capacity: CT-and WOL-specimens up to 25 mm thickness
- 1 autoclave to simulate PWR-conditions in hot cell: 288°C, 140 bar, volume ~1,5 l, capacity: CT-specimen up to 25 mm thickness

5. Progress to Date

(3.1.1.) - Crack growth rate studies with different rise and hold times
- Metallographic investigations

(3.1.2.) - Description of the R-curve test procedure (J-Integral)

(3.1.3.) - Round-Robin Crack Arrest Tests

(3.2.1.) - Compilation of available results

(3.2.2.) - Preparations to conduct nondestructive tests

(3.3.) - Symposium on Fatigue Mechanisms, 22.-24.5.78, Kansas City
- ASME/CSME Pressure Vessels and Piping Conference, 25.-30.6.78, Montreal/Canada
- 9th ASTM International Symposium on Effects of Radiation on Structural Materials, 11.-13.7.78, Richland.
- 6th Water Reactor Safety Research Information Meeting, 8.and 9.11.78 Gaithersburg
- International Cyclic Crack Growth Rate Group, 10.11.78, Bethesda.

6. Results

(3.1.1.) Up to now tests with WOL-specimens of A 508 Cl 2 material show, that the rise time influences the crack growth rate more than the holdtime. To prove this, further tests will be done with rise times up to 30 min and holdtimes up to 60 min.

(3.1.2.) Description of several testing methods

(3.1.3.) The results of the Round-Robin Crack Arrest Tests were reported on the 6th WRSRI-Meeting. At room temperature the K_{Ia}-values of the A 533 BCl 1 material are between 92 and 152 $MPa\sqrt{m}$, 117 $MPa\sqrt{m}$ (average).

(3.2.1.) not finished

(3.2.2.) Ultrasonic investigations have shown several smaller indications in the weld seam.

(3.3.) The essential results are compiled in special reports

7. Next Steps

(3.2.1.) Final compilation of the Intermediate Size Vessel Test results

(3.2.2.) Destructive investigation of the weld seam of vessel V-9

8. Relation with other projects

Research projects RS 245, RS 279, RS 304

9. References

- Hahn, G.T., R.C. Hoagland, A.R. Rosenfield and C.R. Branes: A Cooperative Program for Evaluating Crack-Arrest Testing Methods. ASTM Symposium on Crack Arrest Methodology and Applications, Philadelphia, Nov. 1978.
- Test of 6-inch-Thick Pressure Vessels. Intermediate Test Vessels. ORNL-4895, ORNL-5059, ORNL-NUREG-1, ORNL-NUREG-7, ORNL-NUREG-38

10. Degree of Availability

-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	11.2.2 - 11.2.1	RS 101A

Vorhaben/Project Title

Bau einer Prüfhalle der Staatlichen Materialprüfungsanstalt Stuttgart zur Durchführung des Forschungsprogramms Reaktorsicherheit

Construction of a Testing Hall by the MPA in order to conduct the Research Programme Reactor Safety

Land/Country
FRG

Fordernde Institution/Sponsor
BMFT

Auftragnehmer/Contractor
OFD-Stuttgart

Materialprüfungsanstalt Stuttgart (MPA)

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.7.76	31.12.78	Wagner (OFD) Doll (MPA)

Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	December 1978	

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 Risø National Lab. ORGANIZATION Risø National Lab.
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	

1. General aim. The purpose of the project is to establish the accuracy, which can be obtained by stress analysis of a complicated pressure vessel component and to determine the degree of sophistication, required in such calculations.

2. Particular objectives. The pump nozzle in a BWR-steel pressure vessel has been chosen as object for this investigation. The nozzle is located in the transition zone between the spherical bottom head and the cylindrical vessel part. The nozzle axis is parallel to the centerline of the vessel.

3. Experimental facilities and programme. During the manufacturers hydrotest of the vessel, strain measurements has been performed on the pump nozzle.

4. Project status

4.1. Progress to date A 3-dimensional finite element model of the nozzle has been generated, and 3 load cases run: hydrotest, stresses due to stationary temperatures, and stresses during normal operation conditions. Besides, the experimental program, i.e. strain measurements, has been performed.

4.2. Essential results The stresses and strains due to the internal pressure has been obtained, and both calculated and measured values are shown to be in good accordance with each other. Besides, the results from simplified calculations (2 D) has been compared to the 3D- and experimental results.

5. Next steps

None

6. Relations with other projects

The work is related to

1) Risø investigations of the validation of structural computer codes

2) Risø work on the safety of primary pressure system.

7. Reference documents

[1] S.I. Andersen, J. Reynen, P. Engbæk:
"Stress Analysis of a Main circulation Pump Nozzle".
Risø-TPM-76/1, Jan. 1976.

[2] S.I. Andersen, T. Henriksson, J. Reynen:
"Stress Analysis of a MCP Pressure Vessel Nozzle".
Paper G 8/4 at the 4th Int. Conf. on
Structural Mech. in Reactor Techn., San Francisco 1977.

8. Degree of availability

A limited amount of the results may be freely available.

141-1-05/4111-617 | 11.2.2

TITRE MESURE D'UNE CARACTERISTIQUE D'ARRET DE FISSURE DANS LES ACIERS POUR CUVES DE REACTEURS.		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) CRACK ARREST METHODOLOGY FOR NUCLEAR PRESSURE VESSEL STEELS.		Organisme exécuteur CEA/DTECH Responsable
Date de démarrage 01/01/78	Etat actuel en cours	Scientifiques
Date d'achèvement 01/01/81	Dernière mise à jour 01/01/79	

1. OBJECTIF GENERAL

Le programme proposé a pour but d'étudier les conditions mécaniques et métallurgiques qui conduisent à l'arrêt d'une fissure en cours de propagation.

2. OBJECTIFS PARTICULIERS.

Le programme concerne essentiellement les aciers pour cuves de réacteurs PWR.
La première phase de l'étude consiste en une revue bibliographique et la mise au point de méthodes d'essais.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME.

4. ETAT DE L'ETUDE.

Afin de se familiariser avec les techniques d'essai, nous avons participé à un programme coordonné mis au point par ASTM et NRC. .../...

.../...

Les résultats obtenus sur éprouvettes MRL et BCL ont été transmis au coordonnateur pour comparaison avec les autres laboratoires.

5. PROCHAINES ETAPES.

- 1979 Mise au point des essais

- 1979-80 essais.

6. RELATIONS AVEC D'AUTRES ETUDES.

Sans objet.

7. DOCUMENTS DE REFERENCE.

Néant.

8. DEGRE DE DISPONIBILITE DES DOCUMENTS.

Disponible.

145-1-14		11-2-2
TITRE INTERPRETATION THERMOHYDRAULIQUE DES ESSAIS DE DEPRESSURISATION SUR AQUITAINE II		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) Thermohydraulic interpretation of AQUITAINE II blowdown tests		Organisme exécuteur C.E.A. DRE/STRE Responsable
Date de démarrage 1977	Etat actuel En Cours	Scientifiques
Date d'achèvement 12/1980	Dernière mise à jour 01/1979	

1. OBJECTIF GENERAL

Utilisation de l'installation expérimentale AQUITAINE II pour qualifier des codes d'accident (RELAP IV...) et valider des modèles physiques du système POSEIDON.

2. OBJECTIFS PARTICULIERS

2.1 - Confrontation des expériences avec les codes opérationnels RELAP IV-mod5, BERTHA, EXPRESS ...

2.2 - Qualification d'HEXECO dans le cadre du système POSEIDON.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

AQUITAINE II: Installation implantée à CADARACHE (DRE).

4. ETAT D'AVANCEMENT

Avancement à ce jour

- Des calculs préliminaires ont été faits avec les codes WHAM, RELAP IV et BERTHA, dans le double but de préparer ces outils de calcul à l'interprétation des essais et de juger de leur aptitude à prévoir les résultats expérimentaux (références 7.1, 7.2 et 7.3).
- Puis les résultats de deux essais "simple guillotine" avec tube droit (l'un avec une ouverture totale et l'autre avec une ouverture réduite) ont été comparés à des calculs effectués avec RELAP IV-mod5 et BERTHA.

Principaux résultats

- Avec RELAP IV-mod5, un bon accord calcul-expérience a pu être obtenu jusqu'à 5 secondes de dépressurisation pour l'essai 2 (ouverture réduite; durée de la dépressurisation = 13 secondes) avec l'ensemble d'options suivant :
  - modèle de débit critique : Moody avec un facteur C_D = 0.7 ou modèle homogène (HF-HEM);
  - utilisation du "bubble rise model".

  Cet ensemble, conservé pour l'essai 7 (ouverture totale), a redonné de bons résultats.
- La chute de pression statique à la jonction cuve-tuyau est due essentiellement à la mise en vitesse du fluide, la perte de charge singulière liée à la forme du rétrécissement étant négligeable.
- Les modifications apportées à la méthode de calcul de la propagation de l'onde de dépression dans BERTHA permettent de retrouver avec une bonne précision la vitesse de cette onde dans l'eau (~ 800 m/s) qui a été mesurée sur l'installation HUREPOIX (banc d'essai pour l'instrumentation d'Aquitaine).
- BERTHA présente des difficultés de résolution numérique lorsque le front d'ébullition atteint l'entrée de la cuve, ce qui montre qu'un modèle monodimensionnel n'est pas bien adapté à la géométrie cuve + tuyau.

- Le modèle de déséquilibre utilisé dans BERTHA, qui contient la corrélation du modèle SERINGUE pour la constante de temps θ, ne permet pas de retrouver les chutes de pression très importantes au-dessous de la saturation qui sont observées sur les expériences.

5. PROCHAINES ETAPES

Quelques essais à caractère plus thermohydraulique auront lieu prochainement avec une instrumentation dont les mises au point devraient permettre d'augmenter la précision et la fidélité des mesures.
Ces essais serviront de base à la comparaison calcul-expérience et à la qualification d'HEXECO lorsque l'assemblage POSEIDON pour AQUITAINE II sera réalisé.

6. RELATIONS AVEC D'AUTRES ETUDES

- L'objectif 2.1 rentre dans le cadre de l'étude n° 145-1-01 : "Accident de perte de caloporteur dans les réacteurs à eau pressurisée : codes de 1ère génération".
- L'objectif 2.2 rentre dans le cadre de l'étude n° 145-1-02 : "Développement de modèles et codes de calcul de 2ème génération pour l'étude de l'accident de perte de caloporteur dans les réacteurs à eau pressurisée : programme général coordonné".

7. DOCUMENTS DE REFERENCE

7.1 - "Transitoires monophasiques - Code WHAM"
P. AUJOLLET
Note technique DRE/STRE/LET 77/054

7.2 - "Etudes de dépressurisation dans AQUITAINE II au moyen du code BERTHA"
D. MENESSIER, Y. OCERAIES
Note technique DRE/STRE/LET 77/090

7.3 - "Etudes de dépressurisation sur AQUITAINE II"
R. GINIER, B. RANSON
Communication au séminaire franco-soviétique des 28.01 au 07.02.1978.

7.4 - "Essais d'interprétation des mesures de pression effectuées
sur AQUITAINE II"
J.L. HUET
Note technique DRE/STRE/LMA 78/146

Classification 11.2.1/2

Title 1 Pressure vessel integrity

Title 2

Initiated 1974

Completed

Status Continuing

Last updating

Country U.K.

Sponsor CEGB

Organisation CERL Leatherhead

Project leaders

I. L. Mogford

TITLE (ORIGINAL LANGUAGE): Dimensionnement par ultrasons des défauts de soudure.		CLASSIFICATION: 11.2.3.
		COUNTRY: Belgium.
		SPONSOR: E.S.C.C.
TITLE (ENGLISH LANGUAGE): Size Evaluation of Welding Defects.		ORGANISATION: S.A. COCKERILL.
		PROJECT LEADER: R.V.SALKIN.
INITIATED: 1-1976.	COMPLETED: 12-1979.	SCIENTISTS: Paul de MARNEFFE, Jacques LIMBIOUL, Robert BAR.
STATUS: In progress	LAST UPDATING:	

GENERAL AIM : To evaluate the size of artificial defects inbedded in heavy weldments using different procedures of UT.

PARTICULAR OBJECTIVES : To develop new procedures,
To keep ready for use a collection of typical defects.

EXPERIMENTAL FACILITIES : Five testing blocks thickness 48 to 128 mm, each involving about 12 artificial defects and each duplicated for examination by cutting.

PROJECT STATUS :
- Collection of blocks completed,
- Usual UT methods investigated,
- New UT procedure using cylindrical waves in progress.

NEXT STEPS : To introduce known defects in an actual structure and to evaluate their size using new procedures.

RELATION TO OTHER PROJECTS AND CODES : ASME XI Code
PISC work

REFERENCE DOCUMENTS :
- Semestrial reports to ESCC, not available before end of investigation.
- Publication : Revue de la Soudure, n° 2/1979.

	CLASSIFICATION: Water reactor 11.2.3
TITLE (ORIGINAL LANGUAGE): Contrôle ultrasonore des aciers austénitiques	COUNTRY: Belgium/France
	SPONSOR: Association Vinçotte Framatome
TITLE (ENGLISH LANGUAGE): Ultrasonic testing of austenitic steels	ORGANISATION: Association Vinçotte
	PROJECT LEADER: P. CAUSSIN
INITIATED: 1973 / COMPLETED: 1980	SCIENTISTS: J. CERMAK, L. JACQUES-HOUSSA
STATUS: In progress / LAST UPDATING: May 23, 1979	

1. General aim

Improvement of the capability of ultrasonic testing austenitic stainless steel structures, including welds and castings.

2. Particular objectives

- Development of a technique complying with the ASME code requirements for the inspection of pressure vessels and piping systems of nuclear installations.
- Inspectability of heavy castings according to the French requirements.

3. Experimental facilities and programme

Development of focusing longitudinal wave angle probes and of procedures for applying them.
Facilities are available for manufacturing and characterizing the probes.
The performances are tested on various kinds of welded blocks and castings.

4. Project status

14 probes with refracted angles between 45 and 70 degress have been developed and are used for inspecting, in the field, specimens from 20 up to 100 mm in thickness. The structures inspected include pipe welds, safe-end welds, castings, etc. In that thickness range forgings, plates, statically and centrifugally cast pieces, etc., are inspected in compliance with the ASME code requirements. Procedures have been developed for the inspection of those items and also for testing thin (5 to 20 mm thick) specimens using standard probes. Cast test blocks typical of the welds and of the ligaments of the primary pumps are available. They include natural and artificial flaws.

5. Next step

Development of a technique for inspecting heavy castings up to 200 mm in thickness.

6. References

- Ultrasonic testing of austenitic steel castings and welds
  J.P. PELSENEER, G. LOUIS
  Br. J. of NDT, July 1974, pp. 107-113.

- Ultrasonic testing of austenitic stainless steel structures
  P. CAUSSIN
  Vinçotte report to OECD-CSNI Working Group in Safety Aspects of Steel Components in Nuclear Installations, Revision 2, OECD-NEA Ref.: SINDOC (78) 190, September 14, 1978, 29 p.

- Field efficiency of ultrasonic testing of austenitic steel components
  P. CAUSSIN, J. CERMAK
  Proceedings of Nuclex 78, Basle (CH), October 3-7, 1978, 10 p.

- Performances of the ultrasonic examination of austenitic steel components
  P. CAUSSIN, J. CERMAK
  Proceedings of the Conference on Periodic Inspection for Pressurized Components, London (UK), May 8 - 10, 1979, pp. 207-217.

7. Availability

Details available at

ASSOCIATION VINCOTTE
Département Etudes
B - 1640 RHODE-SAINT-GENESE (BELGIUM)

	CLASSIFICATION: Water reactor : 11/2/3
TITLE (ORIGINAL LANGUAGE): Emission acoustique	COUNTRY: Belgium
	SPONSOR: IRSIA
TITLE (ENGLISH LANGUAGE): Acoustic emission	ORGANISATION: Association Vinçotte
	PROJECT LEADER: P. Caussin
INITIATED: December 1974 / COMPLETED: 1981	SCIENTISTS: L. Jacques-Houssa, W. Sys, P. Lefebvre
STATUS: In progress / LAST UPDATING: 23 May 1979	

1. General aim

Analysis of the diagnosis capabilities of multi-channels acoustic emission (AE) systems used for monitoring of proof tests of pressure vessels.

2. Particular objectives

Study of the correlation between the AE signal characteristics and the initiation and the development of cracks in welded items.

3. Experimental facilities and program

AE monitoring of

- tensile tests of 5 1000 x 600 x 20 mm steel specimens at room and - 20°C temperature
- tensile tests of 20 1000 x 800 x 50 mm steel specimens at room and - 20°C temperature (the specimens contain various kinds of stress raisers including weld defects)
- 2 large pressure vessels.

The facilities include :

- a 8 M Newton tensile machine
- a 60 M Newton tensile machine
- a computerized 24 channels AE system fitted with 6 transient recorders for waveform analysis (FFT, ...)
- Equipment for C.O.D. measurements, Moiré photographs, etc...

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4. Project status

Specification and acquisition of the AE system.
Check of the tensile machines for noise pollution in the 0.1 to 1 MHz frequency range.
Preparation of the AE equipment and of test specimens.

5. References

Diagnosis abilities of acoustic emission multichannel systems : experimental set-up.

P. CAUSSIN, L. JACQUES-HOUSSA, W. SYS.

Proceedings of the Institute of Acoustic, Conference on Fundamental Aspects and applications of A.E., London, December 20-21, 1976.

6. Co-operation

This program is conducted in co-operation with the Laboratory for the Strength of Materials, Ghent University.

7. Availability

Details available at

ASSOCIATION VINCOTTE
Départment Etudes
B - 1640 RHODE-SAINT-GENESE

9. GUIDELINES FOR COMPLETING AN NSRI EXPERIMENT DESCRIPTION

	CLASSIFICATION: WATER REACTORS: 11.2.3
TITLE (ORIGINAL LANGUAGE): ETUDE DES DEFAUTS ARTIFICIELS DE REFERENCE ET DE CALIBRATION POUR LES CONTROLES PAR COURANTS DE FOUCAULT MULTIPARAMETRES DES TUBES DE GENERATEUR DE VAPEUR.	COUNTRY: BELGIUM SPONSOR: IRSIA
TITLE (ENGLISH LANGUAGE): STUDY OF STANDARD AND CALIBRATION ARTIFICIAL DEFECTS FOR THE MULTIPARAMETER EDDY CURRENT INSPECTION OF STEAM GENERATOR TUBING	ORGANISATION: LABORELEC PROJECT LEADER: R. WILPUTTE
INITIATED: 1 APRIL 1978 / COMPLETED: 1 APRIL 1980 STATUS: IN PROGRESS / LAST UPDATING: JUNE 1978	SCIENTISTS: C. van MELSEN, M. ZAVADSKY, D. DOBBENI, R. DE GRAEVE, J. VROMAN

1. General Aim: (a short paragraph on the subject area studied)

2. Particular Objectives: (a paragraph on the specific goal of the project)

3. Experimental Facilities and Programme: (paragraphs as needed)

4. Project Status:

    1. Progress to Date: (a short paragraph)

    2. Essential Results: (a short paragraph)

5. Next Steps: (a short paragraph on future work proposed)

6. Relation to Other Projects and Codes: (titles of related codes or experimental work)

7. Reference Documents: (bibliography of available progress and topical reports and articles)

8. Degree of Availability: (restrictions on distribution of descriptive material, name and address of contact person)

9. Additional Information: (other information of importance)

NOTE: If necessary, please continue the description on a second and separate sheet of paper (one side only).

1. GENERAL AIM

Research and development of equipments and methods for multiparameter Eddy Current inspection of steamgenerator tubing with a multifrequency inside probe.

2. PARTICULAR OBJECTIVES

- Study of the influence of the manufacturing process of artificial defects for Eddy Current calibration
- Definition of a standard artificial defect for multiparameter Eddy Current equipments.

3. EXPERIMENTAL FACILITIES AND PROGRAMME

3.1. Equipments

- Multifrequency Eddy Current equipment using three differential frequencies in the differential mode and one frequency in the absolute mode.
- Analyser equipment using three mixing units which allows the possibility to separate special parameters from the previously described multifrequency unit.
- Magnetic recorders and memory scopes to record and compare the results of the analyse of the artificial defects.

3.2. Programme

- Manufacturing of artificial defects (200 defects) by several methods (mechanic, spark etching, ...) and several shapes (holes feedthrough and flat-bottom, groves, ...).
- Analysing unit including large screen memory scope and hard copy system.
- Internal and external probes.

4. PROJECT STATUS

4.1. Progress to date

Acquisition of the items described in § 3.2.

4.2. Essential results

None.

5. NEXT STEPS

See G 2.

6. RELATION TO OTHER PROJECTS AND CODES

6.1. Research and development of equipments and methods of Eddy Current multiparameter data logging for the inspection of steamgenerator tubing (water reactors : 11.2.3).

6.2. Research and development of equipments and methods for the analyse of the multiparameter Eddy Current signals obtained during the inspection of steam generator tubing.

7. REFERENCE DOCUMENTS

None.

8. DEGREE OF AVAILABILITY

Irsia.

9. ADDITIONAL INFORMATION

None.

GUIDELINES FOR COMPLETING AN NSRI EXPERIMENT DESCRIPTION

	CLASSIFICATION: WATER REACTORS : 11.2.3
TITLE (ORIGINAL LANGUAGE): ETUDE D'EQUIPEMENTS ET DE METHODES POUR L'ACQUISITION DES SIGNAUX MULTIPARAMETRES DE COURANTS DE FOUCAULT POUR LE CONTROLE DES TUBES DE GENERATEUR DE VAPEUR.	COUNTRY: BELGIUM SPONSOR: LABORELEC BELGIUM NUCLEAR POWER PLANTS
TITLE (ENGLISH LANGUAGE): RESEARCH AND DEVELOPMENT OF EQUIPMENTS AND METHODS OF EDDY CURRENT MULTIPARAMETER DATA LOGGING FOR THE INSPECTION OF STEAM GENERATOR TUBING	ORGANISATION: LABORELEC PROJECT LEADER: R. WILPUTTE

INITIATED:	COMPLETED:	SCIENTISTS:
1 JANUARY 1978	1 JANUARY 1981	C. van MELSEN M. ZAVADSKY D. DOBBENI R. DE GRAEVE
STATUS: IN PROGRESS	LAST UPDATING: JUNE 1978	

1. General Aim: (a short paragraph on the subject area studied)

2. Particular Objectives: (a paragraph on the specific goal of the project)

3. Experimental Facilities and Programme: (paragraphs as needed)

4. Project Status:

   1. Progress to Date: (a short paragraph)

   2. Essential Results: (a short paragraph)

5. Next Steps: (a short paragraph on future work proposed)

6. Relation to Other Projects and Codes: (titles of related codes or experimental work)

7. Reference Documents: (bibliography of available progress and topical reports and articles)

8. Degree of Availability: (restrictions on distribution of descriptive material, name and address of contact person)

9. Additional Information: (other information of importance)

NOTE: If necessary, please continue the description on a second and separate sheet of paper (one side only).

LABORELEC
SECTION 6 : ETUDE DES MATERIAUX
SECTIE 6 : MATERIALENKENNIS

1. GENERAL AIM

Research and development of equipments and methods for multiparameter Eddy-Current inspection of teamgenerator tubing with a multifrequency inside probe.

2. PARTICULAR OBJECTIVES

- R & D of a specialized recording system giving the possibility to record non treated Eddy Current signals with a sufficient signal to noise ratio.
- R & D of an electronic equipment which records the co-ordinates of the measured tube and several inspection parameters.
- Improvement of the signal to noise ratio of the Eddy Current multifrequency unit.
- Improvement of the limit of the detection of an outside and inside defect.

These research are conduct to offer the possibility to analyse the Eddy Current data in an off line system which allows the modification of the analysing parameters without disturbing the acquisition unit.

3. EXPERIMENTAL FACILITIES AND PROGRAMME

3.1. Equipments

- Multifrequency Eddy Current equipment using three different frequencies in the differential mode and one frequency in the absolute mode.
- Analyser equipment using three mixing units which allows the possibility to separate special parameters from the previously described multifrequency unit.
- Magnetic recorders and memory scopes to analyse the results of the experiments.

3.2. Programme

See G 2.

4. PROJECT STATUS

4.1. Progress to date

- Acquisition of a modified digital recording system which has been adapted to the analogic recorder. It records the three differential requencies and the absolute frequency with two axis for each parameter.

- Improvement of the signal to noise ratio of the Eddy Current system which allows the detection of very weak defects (quantification of these defects in progress).

4.2. Essential results

Signal to noise ratio obtained for the total recording system more than 70 db. It allows the possibility to separate the acquisition of the EC data and the analyse and printing on the ship chart recorder of the mixed parameters.

5. NEXT STEPS

See G 2.

6. RELATION TO OTHER PROJECTS AND CODES

6.1. Study of standard and calibration articial defects for the multiparameter Eddy Current inspection of steam generator tubing (water reactor : 11.2.3).

6.2. Research and development of equipments and methods for the analyse of the multiparameter Eddy Current signals obtained during the inspection of steamgenerator tubing (Water reactor : 11.2.3).

7. REFERENCE DOCUMENTS

None.

8. DEGREE OF AVAILABILITY

Laborelec.

9. ADDITIONAL INFORMATION

None.

Berichtszeitraum/Period 01.01. - 30.06.1978	Klassifikation/Classification 11.2.3	Kennzeichen/Project Number RS 132
Vorhaben/Project Title Durchführung von Untersuchungen zur Rißerkennung an druckführenden Reaktorbauteilen mit Hilfe der optischen Holographie Crack detection in pressurized vessels and reactor components by optical holography		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Institut für Verfahrenstechnik der T.U. Hannover Callinstraße 36 3000 Hannover 1
Arbeitsbeginn/Initiated July 1974	Arbeitsende/Completed June 1978	Leiter des Vorhabens/Project Leader Prof.Dr.-Ing.F.Mayinger
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General aim

The aim of the activities is to develop a quick and reliable optical method in order to detect cracks in reactor components during fabrication and later in repeating tests.

2. Particular objectives

In the last few years a new nondestructive technique - holographic interferometry - was developed by which material defects are detectable. Up to now it was, however, only used for nonmetallic components.

Our aim is to apply this technique also for the testing of steel. The special objective of this investigation is to detect cracks in or near the surface of pressure vessels using surface waves which are recorded double-pulse holography.

3. Research program

At first, basic experiments with test pieces of simple geometric shape were made. The aim of these preliminary experiments was to find out the smallest crack that can be measured with holographic interferometry in connection with the impact excitation of the test pieces. The indication of cracks is influenced by various parameters:

1. crack-parameters (shape, size, orientation and position of the crack)
2. test piece-parameters (geometric shape, thickness and material)
3. test-method-parameters (impact-parameters sensitivity of the method)
4. influence of a small liquid layer on the surface of the test piece.

Further experiments shall be carried out with test pieces of any geometric shape. After these preliminary investigations experiments will be made with real reactor components like pipelines, elbows, T-pieces, etc. to proof the reliability of this new nondestructive test-method. For these measurements in a nuclear power plant during a repeating test the holographic set-up must be reconstructed to a compact and versatile device.

4. Experimental facilities

The principle of this holographic technique is to visualize the deformation of surface and bending waves due to small irregularities in the material. These waves cause surface deformations which can be measured with holographic interferometry.

To produce the waves it is usually necessary to strike the test section in a suitable manner. This impact excitation may be generated e.g. by a free falling steel ball, by the bullet of an air-gun, an ultrasonic transducer, or any other stress wave generator. The local impact is only a short time excitation after which the waves will travel in all directions. In order to record this very fast event holographically a Q-switched ruby laser is used which produces two short light pulses. The first giant pulse illuminates the test piece shortly before the impact and thus the test piece in its unstrained condition is recorded. The second laser pulse appears at a certain time after the impact to make visible the propagation of the wave. The whole experimental set-up was already discussed in the annual reports A 74, A 75.

5. Progress to date

The theoretical and experimental investigations due to the project RS 132 were completed in June 1978.

In this last report we like to discuss some results, which were made by a somewhat modified holographic interference technique. This modified method is a holographic difference technique. In a normal holographic interference procedure the first hologram will be made from the undeformed object. The second hologram will be made after the deformation of the object, in this special project after the steel ball impact.

In the reconstruction of these both holograms, which were stored in the same hologram plate, the deformation will be visible by an interference pattern which shows the deformation of the test-piece surface in relation to the undeformed object.

With the aid of the new difference technique it is possible to compare two surface configurations after the impact. It is therefore possible to suppress the difficult to survey shape of the interference pattern which will occur at geometric complicated test pieces.

With the aid of a wave-detector, fixed to the surface of the test-piece and a variable delay unit it is possible to switch on both laser pulses at any reasonable time after the steel ball impact.

Another advantage of this technique is the possibility to make visible material flaws which are located beneath the surface of the object.

6. Results

To proof the advantage of this new holographic difference technique some experiments were carried out with a gasket clamp made from cast iron. A result of these experiments can be seen in Fig. 1.

In Fig. 1 the small hairline crack is clearly visible due to a sharp break of some interference fringes (arrow in Fig. 1). Another section of this cast iron gasket shows a very complicated interference pattern (dotted arrow in Fig. 1). This section shows a lot of small cracks, which could be later also detected with a destructive test method.

To proof this new technique for the detection of material flaws beneath the surface a 80 mm thick steelplate would be prepared as shown in Fig. 2. Due to this flaw configuration there is a small plate above the flaw, located in the somewhat big test-piece.

A short steel ball impact will produce waves with different frequency (could be calculated by the Fourier-transform).

If there is any frequency produced by the impact which can generate the small plate in an eigenfrequency, a special vibration pattern must be seen in the double exposure hologram.

Fig. 3 shows the result of these investigations. This interferogram was made with the aid of a small liquid layer on the surface of the test-piece to amplify the small amplitude of such thick steel plate.

The flaw is clearly visible due to the vibration pattern of the small area above the flaw. This special vibration pattern belong to the second eigenvibration of this small plate.

7. Next steps

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8. Relations with other projects

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9. Reference documents

Annual report A 74, A 75, A 76, A 77
Quarterly reports in the series GRS-Forschungsberichte
Report-period Jan. 1978 - March 1978 IRS-F.
" " Apr. 1978 - June 1978 IRS-F.

10. Degree of availability

The quarterly reports are available by Gesellschaft für Reaktorsicherheit (GRS), Cologne.

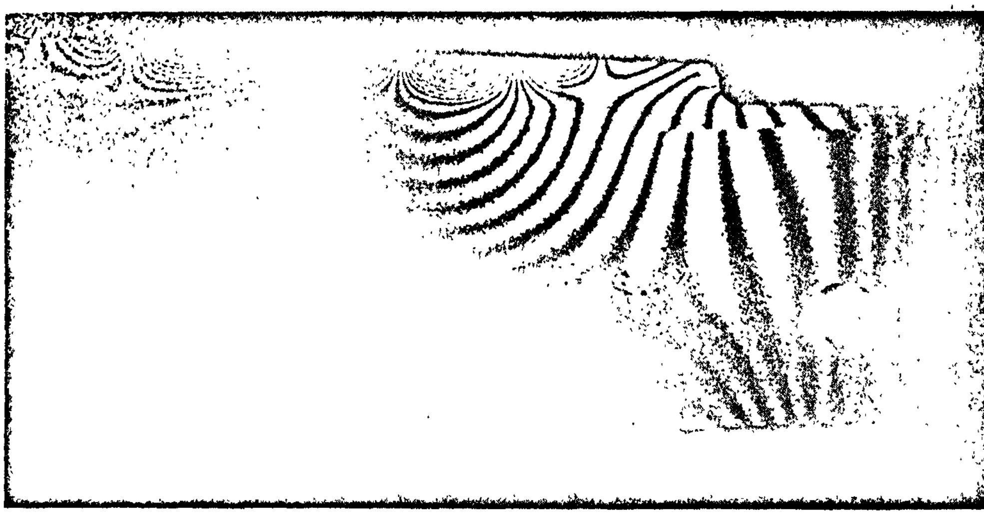

Fig. 1: Interference pattern of a gasket clamp, made by the difference technique, t = 2 μs

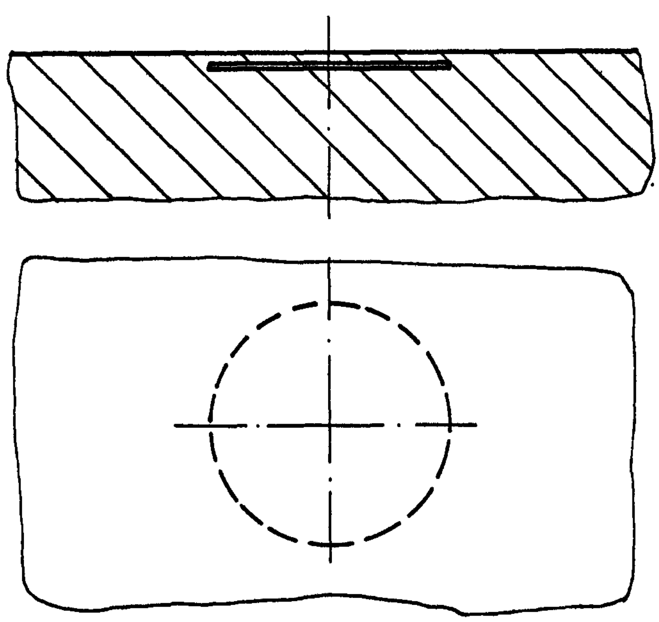

Fig. 2: Material flaw in a 80 mm thick test-piece, located parallel beneath the surface

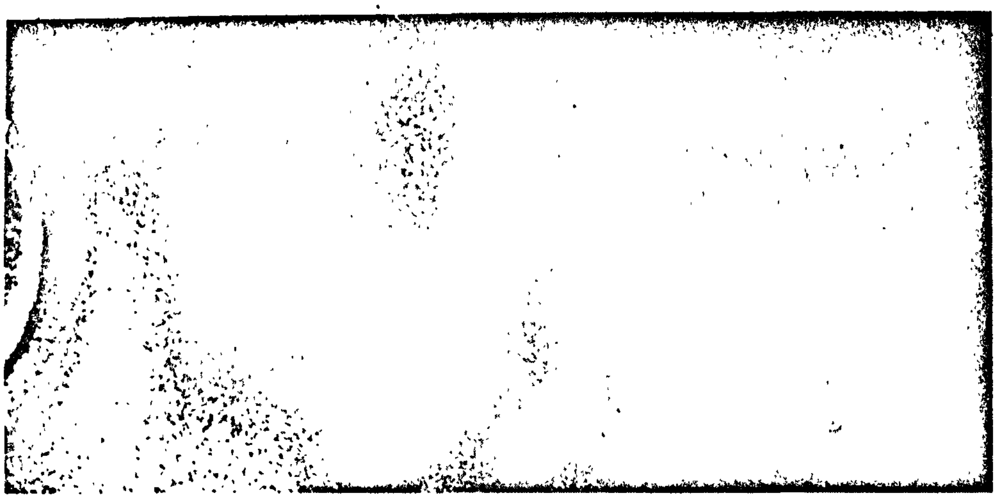

Fig. 3: Detection of a internal flaw (area 1 cm²), located 1 mm beneath the surface of a 80 mm thick test-piece

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	11.2.3	RS 298

Vorhaben/Project Title	
Universalsteuerpult Universal Control Panel	Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 512, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 10. 77	31. 12. 78	K. Ruthrof
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 78	

1. General Aim

Under the scope of developing devices for automated testing of reactor pressure vessels, a universally applicable prototype control panel is to be developed and constructed which would perform by computer control US-testing for complicated geometries, by simultaneous multidirectional motion.

2. Particular Objectives

A panel for manipulator control is to be developed which a) will be relatively independent of both the manipulator drive method and power, and b) allows for complicated geometries to be followed reproducibly with a high degree of accuracy. Furthermore control of up to six different, independent and - in some applications - dependent motions of the manipulator will be required. Various modes of control operation including manual and automated operation have to be provided to ensure the highest possible serviceability.

The control panel is presently being developed as a prototype panel. If during the testing period the experiences with various manipulators are found to be favourable, follow-on models of lower cost will be constructed.

3. Research Program

Based on the experiences to date, various independent modes of operation will be used as a base for the panel concept:

3.1 Possible operating modes of the control panel
- operating modes without process computer
- operating modes with process computer

3.2 Hardware
Control panel
Computer portion

3.3 Software package
Support programs
Operation programs

4. Test Facilities

No special equipment and computer programs are existing for this program.

5. Progress to Date

To 3.1 Because of a several 100 m distance which may exist between control desk and manipulator, it was agreed to instal the power drive units not in the control desk but to arrange them as a separate unit in the vicinity of the manipulator. Since the power unit design must be specific to the drive, it is convenient to include the power units in the scope of the manipulator supply. The exact determination of the interconnections between control desk and power units as well as of the position indicators and manipulator limit switches will be carried out at a later date.

The control structure of the manipulator drives as well as the concept for the interconnections between the measurement device and drive control were prepared and discussed with INTERATOM and MAN. The control circuits to be provided were analyzed and operating instructions for the control parameters prepared accordingly.

To 3.2 The hardware arrangement including the interconnections were discussed and determined. A detailled specification will be prepared.

To 3.3 In parallel, the software structure was roughly determined, in so far as a direct relationship with the hardware exists.

6. Results

To 3.1 A digital positioning control circuit is to be provided with a micro-computer as P-control. Theoretical and, actual value will be checked at regular time intervals. Both stepping motors as well as direct current motors can be used as drive motors. With d. c. motors a supporting analog speed control circuit has to be provided. The binary (output of the possition control) is recommended as an interconnection between control desk and motor drive unit.

To 3.2 A process computer (main computer) which communicates with the external devices, assumes the loading of selected programs and the theoretical computation of the position values. Two microcomputers, each of them serving three axes of motion, perform the position control. Every 5 ms the actual position values will be monitored.

The main computer also provides the theoretical position values for complicated motions of manipulator.

The transmission of the position data, control orders and signals between control desk and electronics is performed in a bit-serial-asynchronous manner. In the electronics cabinet the measuring equipment and the control drive units will be matched to the transmission link.

7. Next Steps

To 3.2 and to 3.3 Hardware specification will be completed, order for hardware be placed, detailed listing of the operating modes, preparation of the software specification.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	11.2.3	RS 267

Vorhaben/Project Title	
Ermittlung der optimalen Ankoppelungsspaltdicke für mechanisierte Kont akttechnik-Ultraschallprüfung von Reaktordruckbehältern Investigating the optimal thickness of the coupling layer for mechanized Ultrasonic Inspections of reactor pressure vessels using the contact technique	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Krautkrämer

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.9.77	28.2.79	Seiger
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General aim

Decrease of the sensitivity variations when ultrasonically testing reactor pressure vessels.

2. Particular objectives

Determination of the optimal thickness of the coupling layer for the mechanized ultrasonic inspection of reactor pressure vessels using the contact technique.

3. Research programme

For the determination of the optimal thickness of the coupling layer the following two contradictory effects are to be taken into consideration:

a) Decrease in the oscillations of the transmission factor with increasing thickness of the coupling layer.

b) Increase in the interferences of the frequency spectrum with increasing of the thickness of the coupling layer.

Thereby the following values are to be measured as a function of the probe location using the thickness of the layer as a parameter: reflector echo amplitudes, acoustic noise as interference level, echo frequency spectra. The thickness of the layer is increased in steps of 0.1 mm starting with 0 mm until the oscillations of the transmission factor fade away. The measurements are to be carried out on the pressure vessel specimen for 1 and 2 MHz using differently sized ultrasonic transducers and different beaming angles using the tandem and single probe techniques.

4. Experimental facilities, computer codes

Pressure vessel specimen, ultrasonic testing device, manipulator device, recorder, probes, path pick-up for recording the variations in the thickness of the coupling layer, holders.

5. Progress to date

The measurements described under 3 have been carried out. The evaluation of the measurements and the formulation of the result report have been started.

6. Results

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7. Next steps

1. Continuation of the evaluation of the measurements.
2. Documentation of the results.

8. Relation with other projects

RS 27

9. References

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10. Degree of availability of the reports

Gesellschaft für Reaktorsicherheit
Glockengasse 2
5 Köln 1

	CLASSIFICATION: 11.2.3.
TITLE (ORIGINAL LANGUAGE): Akustisk Emission fra Stålkonstruktioner	COUNTRY: Denmark
	SPONSOR: EEC - European Coal and Steel Community
TITLE (ENGLISH LANGUAGE): Acoustic Emission from Steel Structures	ORGANISATION: Risø National Lab.
	PROJECT LEADER: C.. P. Debel

INITIATED: 1977	COMPLETED: 1979	SCIENTISTS: C. P. Debel, W. E. Swindlehurst
STATUS: Completed .	LAST UPDATING: 1979	

1. General aim:

To provide a basis for the evaluation of AE signals obtained during surveillance of steel structures.

2. Particular objectives:

A.E. generated during plastic deformation and crack extension in weld metals and heat affected zones have been given particular attention.

3. Experimental facilities and programme:

Testing of small specimens in the laboratory on a modified creep testing machine as well as field-testing of medium-sized pressure vessels pressurized with water.

4. Project status:

The testing is finished and the final report delivered to the sponsor.

5. Next steps: - -

6. Relation with other projects:

Industrial projects on app. of AE technique.

7. Reference documents:

Final report on project 7210-GA/9/901 (E46/1/76) "Applicability of Acoustic Emission to Steel Structures" delivered to the EEC.

8. Degree of availability:

As decided by the sponsor.

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 Risø National Lab.

ORGANIZATION Risø National Lab.

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:

S.I. Andersen

Initiated (date)

74.04.01

Completed: 1977

Status: concluded

Last updating (date)

April 1977

141-3-03/4111-10
151-3-03

11-2-3

TITRE	Pays
Etude des filtrages optiques de films radiographiques	FRANCE
	Organisme Directeur CEA/DgCS/DSN
TITLE (Anglais) Study of optical filtering of radiographic films	**Organisme Exécuteur** CETIM
	Responsable

Date de démarrage	Etat actuel	Scientifiques
19/5/76	en cours	
Date d'achèvement	**Dernière mise à jour**	

1. OBJECTIF GENERAL

Il s'agit de rechercher un procédé, notamment par filtrage optique, permettant d'augmenter le contenu d'informations délivrées par les films radiographiques, notamment en présence d'un rayonnement γ (cas des inspections périodiques de circuits primaires)

2. OBJECTIFS PARTICULIERS

1/ Etude bibliographique
- Cette étude est maintenant quasiment terminée et a permis de faire une revue exhaustive des procédés envisageables.

2/ Etude et réalisation d'étalons de fissures
- Elle a pour but la réalisation de fissures de profondeur et d'épaisseur étalonnées, permettant une approche du problème libérée des incertitudes relatives à la dimension des défauts. Elle a comporté notamment la réalisation d'éprouvettes comportant des fissures calibrées, de profondeur et d'épaisseur croissantes, situées à des profondeurs différentes, mais également des fissures de profondeur variable.Le procédé utilisé est celui mis au point au CETIM, qui fait appel à un usinage précis du défaut étalon lequel est ensuite inclus dans l'éprouvette par soudage par diffusion.

3/ Radiographie des éprouvettes.
Elles ont été effectuées en utilisant les procédures courantes suivies en fabrication, et en respectant au mieux les consignes permettant une bonne reproductibilité.

4/ Dépouillement densitométrique
Une étude systématique du dépouillement densitométrique est entreprise pour tenter de rechercher les paramètres caractéristiques de la détection des fissures.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Les installations sont celles des laboratoires de contrôle non destructif (M. FLAMBARD) et d'optique (M. PARASKEVAS) du CETIM.

Le programme prévoit l'étude systématique des étalons définis en 2, puis celle de défauts réels radiographiés dans diverses conditions.

4. ETAT DE L'ETUDE

L'examen systématique de fissures étalons rectilignes a montré que l'examen au micro densitomètre était suffisant pour permettre sur des fissures fines l'obtention d'un signal caractéristique. L'examen de divers paramètres mis en jeu a montré que l'intégration du signal par une pupille optimisée permettrait peut être d'atteindre le résultat recherché. Les résultats acquis avec un densitomètre modifié ne sont pas concluants car conduiraient à un dispositif d'analyse extrêmement complexe. Divers contacts ont été pris à l'extérieur du CETIM pour voir si d'autres approches ne seraient pas possibles. L'une consisterait à étudier la perturbation apportée au spectre d'un réseau par le défaut. Un des obstacles reste la très grande densité des films en radiographie industrielle.

5. PROCHAINES ETAPES

L'approche par étude du spectre du réseau sera poursuivie et des contacts seront pris, notamment avec les universités parisiennes pour apprécier l'aptitude des traitements d'image sur ordinateur à résoudre ce problème. Un rapport sera rédigé à mi-année qui permettra de faire le point, d'une part des possibilités de l'analyse spectrale du réseau, d'autre part des chances offertes par l'analyse informatique du signal.

6. RELATIONS AVEC D'AUTRES ETUDES

Néant

7. DOCUMENTS DE REFERENCE

Contrat N° SA - 5460 avec le CETIM
comptes rendus d'essais à paraître.
Rapport partiel "Amélioration de la lisibilité des films radiographiques" par D. PARASKEVAS et D. HAUX CETIM

8. DEGRE DE DISPONIBILITE

Documents disponibles sous réserve des accords existants.

9. BUDGET

150 KF en 1979.

141-3-05/4111-6-2		11-2-3
TITRE Emission acoustique. Développement des équipements et des méthodes		**Pays** FRANCE **Organisme Directeur** CEA/DgCS
TITLE (Anglais) Acoustic emission. Development of equipment and methods.		**Organisme Exécuteur** DTECH/STA/SCND - SACLAY **Responsable**
Date de démarrage 1977	**Etat actuel** en cours	**Scientifiques**
Date d'achèvement 1980	**Dernière mise à jour** 12/78	

1. OBJECTIF GENERAL

Mise au point d'équipements et méthodes propres à rendre possible l'écoute de l'émission acoustique des matériaux soumis à contrainte en tant que moyen de détection des évolutions dangereuses des défauts, (en particulier, suivi en continu des circuits primaires de réacteurs).

2. OBJECTIFS PARTICULIERS

Amélioration du traitement des données recueillies lors des essais pour en faciliter l'exploitation et améliorer leur contenu en informations.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Les installations sont essentiellement celles développées au CEA depuis plusieurs années .

- capteurs piézoélectriques avec et sans guides d'ondes.
- préamplificateurs
- amplificateurs
- dispositif de mesures des différences de temps de parcours.

Les programmes en cours sont essentiellement dirigés vers la surveillance des installations en fonctionnement, notamment sur Fessenheim 1, le pressuriseur.

4. ETAT DE L'ETUDE

L'écoute sur le pressuriseur de Fessenheim 1 s'est poursuivie au cours de l'année 1978. Elle s'est traduite par le relèvement de l'activité acoustique en fonction du temps. Bien qu'aucune corrélation n'ai pu être faite avec le programme de fonctionnement du réacteur cette expérience de longue durée a permis d'apprécier l'excellente fiabilité du matériel installé qu'il s'agisse de capteurs ou de l'électronique associée. Aucune panne n'a en effet été relevée.

Simultanément l'étude du traitement informatique s'est poursuivie, essentiellement en ce qui concerne la partie relative à l'aspect traitement des données recueillies, c'est à dire par mise au point d'un fichier de données permettant l'interface entre la bande perforée (résultat des mesures) et l'ordinateur.

5. PROCHAINES ETAPES

Il s'agira essentiellement de compléter l'établissement de ce fichier de données permettant en définitive la visualisation des résultats.

Le programme d'écoute sur le pressuriseur de FH 1 sera également poursuivi.

Un compte rendu intermédiaire en milieu d'année fera le point de l'avancement des travaux au niveau du fichier des données.

6. RELATIONS AVEC D'AUTRES ETUDES

L'écoute de bruit pendant la phase de soudage a été reportée sur une étude qui n'est plus financée par des crédits Sûreté.

7. Documents de référence

Rapport d'activité 1978 à paraître.

8. DEGRE DE DISPONIBILITE

Disponible avec les autorisations et les limitations habituelles.

141-3-07/4111-6-21		11-2-3
TITRE Détection et dimensionnement par courants de Foucault des fissures sous revêtements (PWR)		**Pays** FRANCE **Organisme Directeur** CEA/DgCS
TITLE (Anglais) Under cladding cracks detection and sizing by eddy currents method.		**Organisme Exécuteur** CEA/DTECH/STA/SCND -SACLAY **Responsable**
Date de démarrage 1978	**Etat actuel** A lancer	**Scientifiques**
Date d'achèvement 1979	**Dernière mise à jour** 12/78	

1. OBJECTIF GENERAL

Mise au point d'une méthode de détection d'éventuelles fissures se développant sous le revêtement en acier inoxydable des cuves.

2. OBJECTIFS PARTICULIERS

Les méthodes de détection actuellement envisagées par ultrasons ne sont pas entièrement satisfaisantes. L'utilisation des courants de Foucault peut constituer une solution plus satisfaisante à condition de travailler à basse fréquence.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

L'étude comporte :

- la réalisation de maquettes représentatives
- l'étude et la réalisation de sondes spéciales
- le choix du mode d'excitation.
- la recherche des conditions d'essais optimales en particulier le choix d'un dispositif mono ou multifréquence
- l'adaptation du dispositif et de la méthode aux cuves de réacteurs PWR.

4. ETAT D'AVANCEMENT

Des essais préliminaires ont été effectués sur des plaques d'inox reposant sur un support en acier ferritique, lui même comportant des défauts artificiels. L'appareillage était un dispositif à courants de Foucault basse fréquence (BF) pour assurer une bonne pénétration. En outre cet appareil était associé à trois sondes de différentes dimensions.

5. PROCHAINES ETAPES

Au cours de l'année 1979, on se propose :

- de réaliser des éprouvettes représentatives d'éléments de cuve et de revêtements.
- de rechercher à l'aide d'essais complémentaires le meilleur mode d'alimentation de la sonde : soit en basse fréquence, soit en impulsion.
- de mieux définir les caractéristiques des sondes à utiliser.
- d'adapter cet ensemble sonde-appareillage aux contrôles sur cuves PWR.

Un rapport intermédiaire en cours d'année permettra de faire le point notamment sur le choix du mode d'alimentation de la sonde, décisif pour la suite du programme.

141-3-08/4111-6-7		11-2-3
TITRE Contrôle par ultrasons (et radiographie) des soudures mixtes. (Collaboration avec RFA)		Pays FRANCE Organisme Directeur CEA/DgCS/DSN
TITLE (Anglais) Ultrasonic (and radiographic) testing of dissimilar metal welds. (cooperation with FRG)		Organisme exécuteur CEA/DTECH/STA/SCND Responsable
Date de démarrage 1978	Etat actuel en cours	Scientifiques
Date d'achèvement 1980	Dernière mise à jour 12/78	

1. OBJECTIF GENERAL

Dans le cadre de l'accord qui doit être passé avec la R.F.A., mise en commun des expériences dans le domaine du contrôle par ultrasons (et éventuellement par radiographie) des soudures d'acier inoxydable austénitique.

2. OBJECTIFS PARTICULIERS

Recueillir le maximum d'informations sur les paramètres intervenant dans ce type de contrôle afin de mieux les maîtriser et d'optimiser les procédés.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Il s'agit essentiellement sur le plan français d'utiliser les traducteurs focalisés développés au CEA, notamment en ondes obliques longitudinales.

Deux étalons seront utilisés :

- Bloc étalon n° 1 représentatif d'une soudure mixte de tubulure de générateur de vapeur.

- Bloc étalon n° 2 représentatif d'une soudure de safe-end (embout de sécurité) d'une tubulure de cuve.

On utilisera essentiellement des ondes L et Ten incidence normale et oblique avec des traducteurs focalisés, en émetteur récepteur ou à fonctions séparées, ainsi que des traducteurs du type VINÇOTTE.

Les présentations type A et C seront normalement retenues.

La présentation type B sera limitée aux zones intéressantes.

Pour chaque type de défaut rencontré on étudiera l'influence :

- du type d'ondes
- de l'incidence
- de la forme du faisceau ultrasonore
- de la fréquence
- du type de méthode

en relation avec les caractéristiques des soudures.

Des essais du même type seront effectués par nos équipes sur des blocs en provenance de RFA.

De même les équipes RFA effectueront sur les blocs français des essais dont le type reste encore à préciser.

4. ETAT DE L'ETUDE

Une seule réunion a pu avoir lieu au cours de l'année 1978. De ce fait l'étude n'a pas encore commencée. La prochaine réunion ne pourra avoir lieu d'ici la fin de l'année, par suite de l'impossibilité de réunir tous les participants.

5. PROCHAINES ETAPES

- Définition des capteurs à utiliser et réalisation
- réalisation du programme
- Exploitation
- Eventuellement étude et rédaction de nouvelles spécifications.

6. RELATIONS AVEC D'AUTRES ETUDES

Néant

7. DOCUMENTS DE REFERENCE

Accord CEA-BMFT du 28/09/78

8. DISPONIBILITE

Suivant l'article 9 du dit accord.

TITRE Contrôle des rebords internes des piquages		Pays FRANCE
		Organisme Directeur CEA/DgCS/DSN
TITLE (Anglais) Control of the nozzle inner radius		Organisme Exécuteur CEA/DTECH/STA/SCND
		Responsable
Date de démarrage 1979	Etat actuel à lancer	Scientifiques
Date d'achèvement 1980	Dernière mise à jour 12/78	

1. OBJECTIF GENERAL

Recherche d'une méthode de contrôle non destructif adaptée à la géométrie des rebords internes des piquages sur les cuves de réacteur.

2. OBJECTIFS PARTICULIERS

Il s'agit de rechercher la meilleure méthode permettant de contrôler les rebords internes des piquages des cuves de réacteurs PWR (qui constituent une zone où le champ de contraintes est particulièrement intense) et d'adapter cette méthode aux contraintes particulières des inspections périodiques.

La méthode peut faire appel soit aux ultrasons, soit aux courants de Foucault ou à tout autre technique adaptée. Une première phase consistera à travailler sur une maquette pourvue de défauts étalons artificiels. Cette phase permettra le choix de la méthode ainsi que celui des paramètres essentiels du contrôle.

Une seconde phase consistera en la réalisation d'un appareillage prototype susceptible d'être utilisé en vraie grandeur (par exemple en atelier).

Enfin la troisième phase consistera en l'étude et la mise au point de l'appareillage d'inspection.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Le CEA-STA/SCND possède une portion de piquage de cuve ayant servi à la mise au point de la méthode de contrôle des soudures de piquage.

Ce dispositif pourrait être réutilisé dans la nouvelle optique. Les phases seront les suivantes :

- adaptation de la maquette de piquage et réalisation de défauts étalons
- expérimentation des diverses méthodes ce qui nécessite la réalisation de divers capteurs
- réalisation d'un prototype de faisabilité
- étude et réalisation d'un appareillage d'inspection

4. ETAT DE L'ETUDE

A lancer en 1979. Un rapport d'avancement sera diffusé à la fin de l'adaptation de la maquette (soit vers le milieu de 1979).

	CLASSIFICATION: 11.2.3
TITLE (ORIGINAL LANGUAGE): Utilizzazione dell'emissione acustica del recipiente a pressione a fini di controllo non distruttivo e di sorveglianza	COUNTRY: Italy SPONSOR: ENEL
TITLE (ENGLISH LANGUAGE): Pressure vessel acoustic emission monitoring as an aid to in-service inspection and operation surveillance	ORGANISATION: CISE PROJECT LEADERS G. Possa - F. Tonolini
INITIATED: October 1971 | COMPLETED: 1979	SCIENTISTS:
STATUS: In progress | LAST UPDATING: July 1979	

1. General Aim

To develop non destructive techniques based on the acoustic emission monitoring of a nuclear pressure vessel for in service inspection and operation surveillance.

2. Particular Objectives

To develop and test acoustic emission monitoring systems and methods to be utilized:

1) during pressure vessel hydrotests
2) during power operation

3. Experimental facilities and Programme

3.1. Experimental facilities: None

3.2. Programme

3.2.1. Development and test of an acoustic emission instrumentation system capable of detecting and locating acoustically active defects during pressure vessel hydrotests.

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 at the presence of operational plant noises.

39.507

TITLE (ENGLISH LANGUAGE): Pressure vessel acoustic emission monitoring as an aid to in-service inspection and operation surveillance -2-	CLASSIFICATION: 11.2.3.

4. Project Status

4.1. Progress to date

(3.2.1.): 24 channel system for on-line acoustic emission source location designed, manufactured and tested

(3.2.2.): A preliminary system (2 channels) designed, built and installed on the Caorso BWR pressure vessel

(3.2.3.): Measurements of operational plant noises on the Caorso BWR pressure vessel now under way

4.2. Essential results

Design, construction and test of a multi-channel on-line acoustic emission measurement system capable of precise source location on the pressure vessel wall surface.
Wide experience in the application of the above system to pressure vessel hydrotests.
Experience in the acoustic emission monitoring of a nuclear pressure vessel during power operation.

5. Next steps

Development and testing of a reliable instrumentation system for the automatic continuous measurement and recording of acoustic emission.

6. Relation to other projects and codes

None.

7. Reference Documents

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. 1, 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
4) C. Panzani et alii "Acoustic Emission Monitoring of Pressure Vessels During Workshop Hydrotests", Paper presented at the Conference on New Developments and Special Methods on NDT, Mainz (RFG), April 1978
5) G. Possa, F. Tonolini "Apparecchiature per rilievi diagnostici acustici negli impianti nucleari: ricerca, sviluppo e produzione industriale", Paper presented at the XXIV Congresso Nucleare, Roma, March 1979, published on Energia Nucleare, vol. 26 n. 5, p. 254-266, 1979

TITLE (ENGLISH LANGUAGE): Pressure vessel acoustic emission monitoring as an aid to in-service inspection and operation surveillance	CLASSIFICATION: 11.2.3.
-3-	

6) C. Panzani et alii "CISE Experience on the Acoustic Emission Monitoring in Pressure Vessel Hydrotests", Paper presented at the Symposium "Die Schallemission", Bad Nauheim (RFG), April 1979.

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.

(CISE, C.P.3986, I-20100 Milano)

		CLASSIFICATION: LWR 11.2.3. and LWR 11.2.4.
TITLE (ORIGINAL LANGUAGE): Failure Detection in Reactor Structure		COUNTRY: CEC
		SPONSOR: CEC
TITLE (ENGLISH LANGUAGE): Failure Detection in Reactor Structure		ORGANISATION: JRC-ISPRA
		PROJECT LEADER: CRUTZEN Serge
INITIATED: 1977	COMPLETED: —	SCIENTISTS: LUCIA A. JEHENSON P. BREDAEL I. BORLOO E. WINTERER G.
STATUS: in progress	LAST UPDATING: January 1979	

1. General aim

Characterization of ultrasonic techniques, equipment and analysis of codes (specifications) could lead to better performances of NDT used for quality control, pre-service and in-service inspection, mainly for what concerns sizing of defects.

Use of the JRC know-how acoustic emission in particular cases.

2. Particular objectives

2.1. Study of the transfer function of the complete ultrasonic equipment with particular importance put on transducer and matching of this transducer with the other parts of the equipment.

2.2. Definition and fabrication of artificial defects for calibration of equipment.

2.3. Particular applications :

a. PISC programme (HSST) Destructive examination of plates
Evaluation of results
Recommandations.

b. Ultrasonic test on irradiated fuel bundles

c. Crack detection for material damage evaluation in association with mechanical tests.

2.4. Use of ultrasonic equipment in particular cases of application, in collaboration with national centres.

2.5. Small scale vessels testing with acoustic emission when put under pressure cycling.

3. Experimental facilities

Two characterization laboratories

- use of DPO/PDP 11-34 for characterization

- Schlieren bench (high definition)
- electronic equipment
- characterization equipment for transducers
- correction equipment for transducers
- NDT equipment for I.S.I.

4. Code developped

Computer code for NDT results evaluation.

5. Project status

5.1. Characterization of transducers is made on routine base for external customers.

5.2. PISC programme evaluation has been completed by Ispra (destructive examination of blocks, evaluation of results).

5.3. Reference defects for calibration of equipment are fabricated using electro-erosion or punch technique with high shape reproducibility.

5.4. Tests were performed using acoustic emission on small scale vessels and demonstrated the capability of acoustic emission for early detection of cracks.
These tests are conducted together with ultrasonic and radiographic tests.

6. Next Step

Characterization of equipment and large contribution to the PISC programme is the aim of the activity until the end of 1979 : particular importance will be given to the evaluation of the European test procedures (ISI) compared with the ASME , XI procedure, on the bases of the PISC material and available DATA.
Continuation of A.E. tests on small scale vessels together with ultrasonic inspection and destructive examination for the evaluation of the results.

7. Degree of availability

6 PISC Reports to be published for the SMIRT Conference in Berlin, August 1979.

R. Denis "Characterization of Ultrasonic transducers using cholesteric liquid crystals" EUR 5710.e, 1976

E. Borloo "Pourquoi et comment faut-il caractériser les équipements ultrasonores" Papier présenté à la réunion de L'Association Prof. pour les Essais Non Destructifs"(APEND) Marseille 27.11.1978

Contact people : Serge CRUTZEN
N.D.T. Laboratories
Materials Science Division
Joint Research Centre
21020- ISPRA (Varese)Italy

N.V. KEMA	CLASSIFICATION : 11.2.3
TITLE : Automatisch ultrasoon onderzoek van lassen en componenten	COUNTRY: THE NETHERLANDS
	SPONSOR : P.Z.E.M. ORGANIZATION : KEMA
TITLE (ENGLISH LANGUAGE): Automatic ultrasonic examination of welds and components	PROJECTLEADER : K. Boer
INITIATED : 1975 — LAST UPDATING : 1978	SCIENTISTS : De Jong, Tempelman
STATUS : to be continued — COMPLETED : –	

General aim

The performance of inservice inspection (ISI) by ultrasonic examination in accordance with the ASME-code and the Dutch-code.

Particular objectives

Development of an automatic system for the volumetric inspection by ultrasonic to remote areas where manual access is retricted for manual inspection operations.
Parts of the inspection system are special developed ultrasonic probes, manupulators, electronics and computer. The special purpose computer is used for control and acquiring all inspection data in accordance with the codes. On-line and off-line evaluation is possible.

Programm

The programm consists of:
- ultrasonic inspection of longitudinal, meriodinal and circumferential welds and nozzles of steamgenerators, pressurizer, pumps, main loops and reactor vessel.
- ultrasonic inspection of studs, threaded holes and ligaments of pumps and reactor vessel.

Project status

Tests are performed during the shutdown at February 1976, February 1977 and November 1977 for a PWR.

Next steps

The work is continued during the coming shutdown in 1978 and 1979 to complete the first interval of a PWR. Simular steps are taken for the second interval of a BWR.

Relation to other projects

Not applicable.

Reference documents

General description.

Degree of availability

Through the organization KEMA.

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

NIL-MI-TNO	CLASSIFICATION : 11.2.1/ 11.2.3
TITLE : Literatuurstudie van materiaalparameters die nodig zijn bij de interpretatie van defect-localisatie met AE-apparatuur.	COUNTRY: THE NETHERLANDS
	SPONSOR : Ministry of Social Affairs and others
TITLE (ENGLISH LANGUAGE): Literature survey of material parameters which are necessary for the diagnosis of defects localized with AE techniques.	ORGANIZATION : NIL-MI-TNO
	PROJECTLEADER : Kloots
INITIATED : Sept. 1974 / LAST UPDATING : May 1978	SCIENTISTS : Boerstoel, v.d. Brink
STATUS : Completed / COMPLETED : June 1978	

	CLASSIFICATION: 11.2.3
TITLE (ORIGINAL LANGUAGE): DEVELOPMENT OF ACOUSTIC EMISSION MEASUREMENT	COUNTRY: UK SPONSOR:
TITLE (ENGLISH LANGUAGE):	ORGANISATION: RNL UKAEA PROJECT LEADER: P BENTLEY
INITIATED: 1969 / COMPLETED:	SCIENTISTS:
STATUS: / LAST UPDATING: MAY 1979	

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 to 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 tests to be compared in a more quantitative manner, calibration devices are under development.

REFERENCE DOCUMENTS

Internal documents

Classification: 11.2.4.

Title: Trykprøvningsfacilitet	Country: DENMARK Sponsor: Risø National Laboratory
Title: Pressure Testing Facility	Organization: Risø National Laboratory
Initiated date: 1977 Completed date: 1978 Status: In progress	Scientists: C. Debel

1. General aim: To provide a facility for testing to failure of intermediate size vessels under controlled conditions.

2. Particular objectives: Testing of experimental vessels particularly under gas pressure in order to facilitate the testing at different temperatures.

3. Experimental facilities and programme: The aim is to provide a facility for fracture mechanics and acoustic emission experiments on vessels.

4. Project status: Designing is in progress.

5. Next steps: Construction.

6. Relation to other projects: The facility should be available for the projects Dynamic Fracture Mechanics and Acoustic Emission.

7. Reference documents: None so far.

8. Degree of availability: In principle no limitations.

		CLASSIFICATION: LWR 11.2.3. and LWR 11.2.4.
TITLE (ORIGINAL LANGUAGE): Failure Detection in Reactor Structure		COUNTRY: CEC SPONSOR: CEC
TITLE (ENGLISH LANGUAGE): Failure Detection in Reactor Structure		ORGANISATION: JRC-ISPRA PROJECT LEADER: CRUTZEN Serge
INITIATED: 1977	COMPLETED: —	SCIENTISTS: LUCIA A. JEHENSON P. BREDAEL I. BORLOO E. WINTERER G.
STATUS: in progress	LAST UPDATING: January 1979	

	CLASSIFICATION: 11.3.1
TITLE (ORIGINAL LANGUAGE): Programma di ricerca sul comportamento del calcestruzzo per contenitori di reattori nucleari.	COUNTRY: ITALY
	SPONSOR: ENEL
TITLE (ENGLISH LANGUAGE): Research Program on Concrete for Nuclear Reactor Vessels	ORGANISATION: ENEL
	PROJECT LEADER: P. BERTACCHI
INITIATED: 1970 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: June 1979	

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

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Research Program on Concrete for Nuclear Reactor Vessels	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).

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.

Classification
11.3.2/11.3.4

Title 1 Brudundersøgelse af Betontank-lågmodeller	COUNTRY Denmark SPONSOR Risø National Laboratory ORGANIZATION Risø National Laboratory
Title 2 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

are taken into account. A new failure criteria and a proposal for a constitutive model for concrete are used by the finite element program.

5. Next steps. None.

6. Relations with other projects. The programme is part of a joint Nordic development work on a PCRV for BWR application.

7. Reference documents.

Ultimate load behaviour of PCRV top closures.
S.I. Andersen, N.S. Ottosen.
Paper H 4/3, 2. Int. Conf. on Struct. Mech. in Reactor Technology Berlin (1973).

Theoretical and Experimental Studies for Optimization of PCRV Top Closures.
N.S. Ottosen, S.I. Andersen.
Paper H 3/6, 3. Int. Conf. on Struct. Mech. in Reactor Technology London (1975).

A Failure Criterion for Concrete
N.S. Ottosen
Journal of the Engineering Mechanics Division, Proceedings of the American Society of Civil Engineers. Vol. 103, No. Eh4, Aug. 1977.

Structural Failure of Thick-Walled Concrete Elements.
N.S. Ottosen, S.I. Andersen.
Paper H 4/3 4. Int. Conf. on Structural Mech. in Reactor Technology. San Francisco (1977).

8. Degree of availability

A limited amount of the results may be available on an exchange basis.

142-1-07 /4111.10		11-4
TITRE COMPORTEMENT DES ENCEINTES DE CONFINEMENT DES PWR AU-DELA DE LA PRESSION DE DIMENSIONNEMENT.		Pays FRANCE Organisme Directeur CEA/DSN EDF/SEPTEN
TITLE (Anglais) OVERALL BEHAVIOUR OF PRESTRESSED CONCRETE CONTAINMENT SUBJECTED TO INTERNAL PRESSURE GREATER THAN THE DESIGN VALUE.		Organisme exécuteur BUREAUX D'ETUDES Responsable
Date de démarrage 1978	Etat actuel EN COURS	Scientifiques
Date d'achèvement 1980	Dernière mise à jour FIN DECEMBRE 1978	

1. OBJECTIF GENERAL

- Le rapport RASMUSSEN (WASH 1400) ayant mis en évidence que des séquences accidentelles (avec fusion du coeur) pouvaient donner lieu à des élévations de pression supérieures à celles actuellement prises en compte dans le dimensionnement des enceintes de confinement des réacteurs PWR, il a été jugé utile, du point de vue de la sûreté, de connaître le comportement de ces ouvrages au-delà de la pression de calcul.

- Un groupe de travail EDF (SEPTEN) - CEA (DSN) a été créé en Juillet 77 et a reçu pour mission de proposer et de réaliser un programme d'actions permettant d'évaluer le comportement mécanique des enceintes au-delà de la pression de calcul.

2 OBJECTIFS PARTICULIERS

Le programme doit permettre de déterminer les points suivants :

1/ Les parties ou composants les moins résistants en pression : coque en béton ou points singuliers (sas, vannes)

2/ Pour les parties en béton :
 . la pression de fissuration traversante,
 . la pression de rupture et les modes de ruine.

3/ L'évolution de la fissuration en fonction du chargement.

4/ Le taux de fuite associé aux différentes pressions.

3. INSTALLATION EXPERIMENTALE ET PROGRAMME

31. Pas d'installation expérimentale, car il s'agit pour le moment essentiellement de calculs, la phase essais n'étant prévue que pour 1980.

32. Programme :

- les conditions accidentelles à prendre en compte sont les suivantes :
  . accident de référence (LOCA) associé à la perte d'injection de secours et au refroidissement de l'eau d'aspersion soit traduit en termes d'évolution de pression et de température :
    . montée lente en pression 0,3 bar/heure
    . montée en température 2° C/heure.
- Les types d'enceinte à étudier sont les suivantes :
  . PWR 900 avec peau d'étanchéité,
  . PWR 1300 enceinte double sans peau,
  . PWR 900 type Cruas.
- Le programme proprement dit comprend cinq étapes :
  . 1è étape : comportement des enceintes jusqu'à fissuration,
  . 2è étape : comportement dans le domaine élasto-plastique :
    - étude d'un cas test,
    - étude de l'enceinte hors traversées
  . 3è étape : comportement des points singuliers (sas matériel, traversées vapeur)
  . 4è étape : deux orientations possibles en fonction des résultats des étapes précédentes :
    - soit développer calcul des parties métalliques (si ce sont effectivement les points faibles)
    - soit étudier l'influence des défauts de réalisation (σ béton, précontrainte)
  . 5è étape : maquette pour validation expérimentale.

4. ETAT DE L'ETUDE

41. Avancement à ce jour

- La première étape relative à la fissuration en phase élastique a été lancée avec le Bureau d'Etudes Coyne et Bellier.
  Les résultats seront disponibles pour Janvier 79.
- La première partie de la deuxième étape, le cas test, est au niveau des consultations. Cette étape est considérée comme fondamentale car elle doit permettre de faire le point sur les possibilités actuelles des Bureaux d'Etudes et Services spécialisés d'EdF et du CEA, dans le domaine du calcul élastoplastique.

  Les résultats sont attendus pour la mi-79.

5. PROCHAINES ETAPES

- Présentation des résultats des travaux engagés en 1978 (1è étape et cas test)

- Lancement des travaux de la 2ème partie de la 2ème étape, c'est-à-dire l'étude du comportement des parties béton (hors traversées).

6. RELATION AVEC D'AUTRES ETUDES

Aucune relation.

7. DOCUMENTS DE REFERENCE

Les documents établis en 1978 sont les suivants :

- DSN/SETSSR-T-78-1786 du 10 avril 1978 : Programme général SEPTEN GC 78-04

- DSN/SASR/SETS 78-2100 du 11/10/78 : Définition du cas test.

8. DEGRE DE DISPONIBILITE

Les documents ci-dessus sont disponibles après accord préalable du CEA et de l'EDF.

142-1-05 / 4111-01		11-4
TITRE ETUDE DU COMPORTEMENT DU PUITS DE CUVE DES REACTEURS PWR 900 MWe EN CAS DE RUPTURE LIMITEE DE CUVE.		Pays FRANCE Organisme Directeur CEA/DgCS
TITLE (Anglais) PRIMARY SHIELD WALL BEHAVIOUR OF PWR'S IN CASE OF RESTRICTED PRESSURE VESSEL RUPTURE.		Organisme exécuteur CEA/DEMT Responsable
Date de démarrage 1/75	Etat actuel EN COURS	Scientifiques
Date d'achèvement 12/80	Dernière mise à jour 12/78	

1. <u>OBJECTIF GENERAL</u>

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

2. <u>OBJECTIFS PARTICULIERS</u>

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

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

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

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

3. <u>INSTALLATIONS EXPERIMENTALES ET PROGRAMME</u>

L'étude et la mise au point de l'installation expérimentale ont été confiées au DEMT.

Le programme expérimental comprend les étapes suivantes :

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

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

4. ETAT DE L'ETUDE

41. Avancement à ce jour

Les travaux effectués en 78 sont les suivants :

- consultations pour connaître les devis de réalisation des maquettes acier du circuit primaire,
- lancement des commandes au GEF de Saclay (novembre 1978)
- études d'avant-projet pour la maquette béton de puits de cuve : contacts avec le Bureau d'étude Séchaud et Metz (retenu par EdF pour les structures internes du bâtiment réacteur).
- devis de réalisation de Séchaud et Metz des maquettes en béton après critique de l'avant-projet du DEMT. Commande lancée fin novembre 78.

42. Résultats

Pas de résultat car on est dans la phase de définition et de réalisation des structures acier et béton.

5. PROCHAINES ETAPES

51. Préparation du planning de réalisation de la partie mécanique :

- Modification de la "station" d'essais (blockhauss)
- Circuits d'essais et de pressurisation
- Essais préliminaires de détermination des épaisseurs d'affaiblissement pour rupture guillotine et boutonnière. Ces essais seront réalisés sur la boucle PRIMEAU du DEMT.

52. Réalisation des maquettes en béton du puits de cuve

53. Premiers essais prévus fin 79

6. RELATIONS AVEC D'AUTRES ETUDES

Néant.

7. DOCUMENT DE REFERENCE

Les principaux documents émis en 78 sont les suivants :

- Note DSN/SETSSR-T-78 du 3/3/78 : définition des maquettes béton au 1/20.
- Note DEMT/SMTS/RDMS 78-263 du 11/8/78 : commentaires sur la similitude dynamique.
- Dossier d'avant-projet du puits de cuve en béton : Plans de principe de maquettage (DEMT).

8. DEGRE DE DISPONIBILITE

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	11.5	06.01.10 (PNS 4239)
Vorhaben/Project Title		Land/Country
Auswirkung von Kühlkanalblockaden auf die Kernnotkühlung (FEBA-Programm)		FRG
		Fördernde Institution/Sponsor
		BMFT
Influence of Coolant Channel Blockages upon Core Cooling in the Reflood Phase of a LOCA (FEBA-Program)		Auftragnehmer/Contractor: Kernforschungszentrum Karlsruhe (KfK) Projekt Nukleare Sicherheit (PNS)
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1. 1973	31.12.1981	P. Ihle
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	December 1978	

1. General Aim

The influence of coolant channel blockages upon the cooling effect during the reflood phase of a loss of coolant accident is investigated.

2. Particular Objectives

The objective of the program is to investigate the influence of size and shape of coolant channel blockages on the flow and the heat transfer conditions in the vicinity of blockages. Not considered is the influence of the cooling conditions on the propagation of the cladding deformation.

3. Research Program

The program consists of three major steps to investigate separate effects of the reflood cooling conditions in a PWR-geometry:
(FEBA, Flooding Experiments with Blocked Arrays)

3.1 Experiments with a 5-rod row, all subchannels blocked by the same blockage ratio, variation of size and shape of the blockages. These tests serve mainly for optical observation of the two-phase flow and for qualitative studies of the cooling effects.

3.2 Experiments with a 25-rod bundle.
- Similar objectives as in 3.1, but quanitative results.
- First tests to study the two-phase flow bypassing blockages.

3.3 Experiments in larger bundles up to 50 rods, with some coolant channels partly blocked to study the effect of flow redistribution.

4. Experimental Facilities, Computer Codes

The test rig is designed for the separate effect test program simulating the reflood phase of a LOCA in a PWR, excluding system effects. The heat transfer analysis code calculates the instantaneous values of the stored heat, surface temperature, heat flux and heat transfer coefficients. In a further option the simulation quality of the heater rods and sleeve-type blockages used can be calculated.

5. Progress to Date

The Flooding Experiments with Blocked Arrays (FEBA) have been continued to determine the influence of the shape of coolant channel blockages. To simulate ballooned fuel rods solid as well as hollow sleeves have been attached to the heater rods at the mid plane of the heated zone of the 5-rod row. The effect of these blockages upon the cooling conditions during reflood was investigated. The simulation quality of the sleeve blockages used was calculated and tested.
A final series of experiments was performed with the five-rod-row and plate-type blockages in the mid plane of the array. This additional series was necessary for a direct comparison of different blockage shapes without superimposed influences from different test section characteristics and initial temperature distributions. Single phase pressure drop coefficients were measured in water for the sleeve-type and the plate-type 62% blockage. In order to measure steam superheat temperatures in the transient two-phase flow, probes of different design were used and their signals compared and analysed to define the best design. To detect the local water content high frequency probes were tested. The evaluation of experimental results was conticontinued.

6. Results

The five-rod tests with sleeves attached to simulate ballooned fuel rods, produced results qualitatively similar to those obtained from the earlier referenced tests and the final test series using plate blockages. This was true for both the massive and hollow sleeves; i. e. without and with a radial gap between the sleeve and rod surface, respectively. However, the sleeves, which have tapered ends, influenced the transient two phase flow considerably less than the plates

which create sudden reductions and expansions of the flow path.
Under forced feed conditions an obstacle in the coolant channel may cause improved cooling effectiveness in a region of axially limited length behind the obstacle. The degree of the improvement and the length of the region depend on the water content in the two-phase flow and on the geometrical shape of the obstacle. The water content increases with increasing flooding rate; obstacles with sharp edges cause a significantly stronger influence than slim rounded blockages. Starting from a spot of emphasized cooling an additional quench front may spread.
The results show that at low flooding rates (2 cm/s) the influence of a sleeve-type 62% blockage with tapered ends is hardly detectable and does not produce an additional quench front. However, at the same flooding rate a plate-type 62% blockage causes somewhat lower turn-around temperatures in a region of at least 300 mm downstream from the blockage and the initiation of a new quench front. At higher flooding rates (6.7 cm/s) also the sleeve-type blockage leads to somewhat reduced turnaround-temperatures, but in a region of less than 100 mm only behind the upper end of the sleeve. At this flooding rate the plate-type blockage causes a strong cooling improvement over more than 300 mm with decreasing rod temperatures from the start of reflooding and with very early quenching.
Since the single phase pressure drop coefficients of the sleeve- and plate-type blockages were in the same order of magnitude the results can be applied to partly blocked bundles also where the flow redistribution is taken into account.
These results confirm PWR-FLECHT blocked bundle data as far as improved cooling behind plate-type blockages is concerned but they further show that it is important to perform blockage experiments with slim rounded blockage sleeves similar to ballooned fuel rods, in order to avoid taking unjustified high credit for the cooling effect of a blockage.

The simulation quality of the sleeve blockages used was calculated and tested. Taking into consideration the total length of the bundle, both types were in good agreement with the predicted behavior of a blocked fuel rod array. The hollow sleeves however gave information about the surface temperature at the blocked part itself with surprisingly good agreement with the calculated values. The influence of spacer grids

compared with the influence of a 62% sleeve blockage was demonstrated. The heat transfer coefficient downstream of a spacer grid was up to 50% higher than the upstream value.
FEBA tests conducted under conditions similar to REBEKA-tests resulted in good agreement inspite of different simulators and bundle sizes.

The local steam superheat temperature, difficult to investigate in transient two phase flow, can be measured within the first part of the reflood phase at flooding velocities less than 5 cm/s with an accuracy of about 10%. The steam probe was developed within the FEBA-program. During the first seconds of the reflood phase, near the mid plane of the bundle at all arrays investigated, the steam temperature was in the range of 100^{o}C less than the correspondent cladding temperature. Later in the transient, as expected,the steam temperature-clad temperature-difference was dependent on the axial level and the blockage used. At flooding velocities less than 3 cm/s the superheated steam leaving the mid plane region of the bundle was heating up part of the upper end of the bundle. This effect was observed within the first half of the reflood phase inspite of the presence of water droplets. The local water content was measured with high frequency probes installed in subchannels at various levels.

7. Next Steps

The results for a 5-rod row described in the preceding paragraph will be checked quantitatively with a small number of tests in a 25-rod bundle with reduced wall effects. In addition the influence of spacer grids on the axial distribution of the two-phase heat transfer coefficients will be measured. As a bridge to blockage experiments in a 50-rod bundle some tests in a nonuniformly blocked 25-rod bundle will also be performed.
Tests with a long blockage in a 16-rod bundle for blockage bypass, mounted in the 25-rod bundle housing, serve as screening tests to investigate long term cooling problems.

8. Relations to Other Projects

Low pressure tests performed by KWU; FLECHT/SEASET-Program (USNRC, Westinghouse, EPRI); 2D-Experiment in Japan.

9. References

PNS Semi-Annual Reports (german, english abstract)
1977/2, KfK 2600
1978/1, KfK 2700
A. Fiege et. al: Stand und Ergebnisse des theoretischen und experimentellen Forschungsvorhabens zum LWR-Brennstabverhalten bei Reaktorstörfällen; KfK-Ext. 28/78-1, Sept. 1978.

10. Degree of Availability of the Reports
Unrestricted Distribution

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	11.5	RS 194

Vorhaben/Project Title	Land/Country
Blockierte Kühlkanäle Blocked Cooling Channels	FRG
	Fordernde Institution/Sponsor BMFT
	Auftragnehmer/Contractor KRAFTWERK UNION AG Reaktortechnik R 513, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 7. 76	30. 6. 78	D. Hein
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Completed	31. 12. 78	

1. General Aim

Investigation on the effect of ballooned fuel claddings during a LOCA on the emergency cooling of a BWR.

2. Particular Objectives

Based on the measured heat transfer rates in the blocked region, the influence of the flow distribution on the heat transfer will be evaluated upstream and downstream of the blocked region for two degrees of blockages.

3. Research Program

- Planning and preparation of blockage tests
- Modification of the test facility for BWR geometry blockage tests
- Assembly of the bundle in the rod bundle container
- Performance of tests with 37 % and 70 % blockage
- Evaluation of tests.

4. Test Facility

The BWR tests will be performed in a double-bundle test facility which was installed within the scope of Task RS 36. During test performance, the cooling channels of the 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

Two tests with two different degrees of blockage were performed. During the initial test phase, the blockage of the free flow cross-section of the fuel bundle was approx. 37 %, during the second test phase it was approx. 70 %. Although in the first case the blockage was symmetrical, it was not uniformly arranged across the fuel bundle cross-section. In the second case the blockage was uniformly arranged. The peripheral zones were only slightly blocked because of construction limitations.

For evaluation of the measuring devices, a small test arrangement was prepared in which, in one of the 9 heated rods comprision the test section, in-core measurement devices for low pressure emergency cooling tests could be tested under operating conditions.

6. Results

In comparison to results without flow blockage (RS 36/C) the results can be summarized as fellows:

- the influence of blockage in the experimental range of 37 to 70 % is small.
- During spraying of the fuel bundle from the top, no effect was observed on the quench-period and heat-up intervals (these are turnarround temperatures minus initial temperatures) outside the blocked region.
- Within the blocked area an improved cooling performance (shorter heat-up interval) was observed with spraying. A reduction of the quench periods was only detected with pressures of 10 bar. During the 10 bar tests a new wetting front was observed on the blockage plates-similar to what was observed on several spacer grids.
- During flooding tests no increase of the heat-up intervals were measured, but increased quench-periods were observed above the blocked area.
- The measuring point at 25 mm above the blockage shows improved cooling performance during flooding (shorter

heat-up intervals); in a few tests, however, the quench-times are smaller than with the unblocked tests.

- Below the blockage no changes were absorved in the cooling conditions.

In summary it can be said that the changes caused by a blockage simulated by plates in a BWR fuel bundle have hardly any effect on the cooling performance of the total bundle. Within the blockage the results compare approximately to those observed in the spacer grid areas. Possible flow redistributions within the coolant, which could be caused by varying degrees of blockage, cannot be discerned because of the fuel bundle container, which limits the free flow field.

The test facility for evaluation of the measuring devices is ready for operation.

The testing of the measurement pick-up device has begun.

The evaluation of further measurement methods of interest will be continued under RS 287 scope. These results can also be applied to PWR emergency cooling tests with partly blocked cooling channels.

7. Next Steps

Project completed.

8. Relation with Other Projects

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9. References

B. Brand, H.-P. Gaul, J. Sarkar
"Emergency Cooling Program - Low Pressure Tests, Blocked Cooling Channels with BWR Geometry, BKKS".

10. Degree of Availability

	CLASSIFICATION: 11.5
TITLE (ORIGINAL LANGUAGE): Effetti dello scoppio di tubi a pressione CIRENE	COUNTRY: Italy
	SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): CIRENE Power Channel Pressure Tube Burst Tests	ORGANISATION: CISE
	PROJECT LEADER: G. Possa
INITIATED: October 1970 / COMPLETED:	SCIENTISTS: M. Famiglietti
STATUS: Almost completed / LAST UPDATING: July 1979	

1. General aim : to investigate the consequences of the hypothetical explosion of a power channel pressure tube for the CIRENE reactor.

2. Particular objectives: to measure the pressure peaks produced by the explosion in the D_2O tank and to determine the associated stresses on major mechanical structural components.

3. Experimental facilities and programme

3.1. Experimental facilities

- BETULLA: facility located in CCR Euratom at Ispra including a pressure vessel (with nearly CIRENE dimensions) for explosion containment
- MARC: a 1:5 scaled down model of the CIRENE reactor structures surrounding the core

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 at full scale (dimensions and structural materials) the CIRENE power channel.
3.2.4. Burst tests in the MARC assembly, mainly to determine the structural effects of an explosion in lateral position.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
CIRENE Power Channel Pressure Tube Burst Tests	11.5

4. Project Status

4.1. Progress to date

(3.2.1.): Completed
(3.2.2.): Tests and analysis completed
(3.2.3.): Tests and analysis completed
(3.2.4.): Tests and analysis completed

4.2. Essential results

- Basic understanding of the dynamics of underwater explosion in a confined volume
- Understanding of the rupture process of a pressurized Zircaloy tube artificially defected, under typical conditions of a pressure tube reactor
- Information on the explosion stresses in reactor structural components

5. Next steps

Programme completed

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
2) F. Dallavalle, M. Famiglietti, W. Hotz, G. Possa "Explosive Rupture of a Power Channel Pressure Tube in a D_2O Reactor", presented at the 4th SMIRT Conference, San Francisco, August 15-19, 1977
3) G. Chevallard, M. Famiglietti, A. Parmeggiani, G. Possa "Simulation of the Dynamic Loads and Stresses Due to Explosive Rupture of a Pressure Tube in a Scaled-Down Model of the CIRENE Reactor Structure", Paper presented at the 5th SMIRT Conference, Berlin, August 13-17, 1979

7. Degree of availability

To a limited extent.
(CISE, C.P.3986, I-20100 Milano)

12. QUALITY ASSURANCE

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen Project Number
29.9.78 bis 31.12.78	12	RS 345

Vorhaben/Project Title	
Vergleich und Anwendung internationaler technischer Regelwerke - dargestellt am ASME Boiler und Pressure Vessel Code und den einschlägigen deutschen Regeln Comparison and Application of International Technical Codes - Represented by the ASME Boiler and Pressure Vessel Code and Corresponding Rules in the Federal Republic of Germany	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: TÜV Rheinland e.V. Köln

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Loader
20.09.1978	19.09.1979	Dr. Ringelstein
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

Comparison between the nuclear technical rules of the ASME Boiler and Pressure Vessel Code and those of the corresponding Codes in the Federal Republic of Germany with special regard to the parts concerning the primary coolant system and the containment.

2. Particular Objectives

Detailed comparison of legal basis, contents and fundamental requirements concerning material, design, fabrication, installation, examination, testing, protection against overpressure, nameplates, stamps and reports.

3. Research Programm

3.1 Collecting suitable literature references

3.2 Procuring literature references

3.3 Detailed comparison of the fundamental requirements

3.4 Reports each quarter of a year

3.5 Quantitative comparison illustrated by a vessel working in a primary coolant system

3.6 Research report

4. Experimental Facilities, Computer Codes

5. Progress to Date

5.1 Collecting suitable literature references

5.2 Procuring literature references

5.3 Detailed comparison of fundamental requirements

The comparison concerning legal basis and contents of the Codes has been finished to a great extent. The comparison of the fundamental requirements concerning design is being prepared.

5.4 First report

6. Results

6.1 Main subjects were formulated and detailed on the basis of the procured literature references

6.2 The suitable procedure for preparing the comparison was ascertained

6.3 The situation of the National Codes was examined with regard to their legal basis and their contents

7. Next Steps

7.1 Procuring of literature references will be continued

7.2 Comparison of fundamental requirements according to the National Codes will be continued

8. Relation with other projects

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9. References

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10. Degree of Availibility of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	12.1	RS 317

Vorhaben/Project Title	
Statusanalyse der Qualitätssicherungssysteme bei den Betreibern für Herstellung und Betrieb von Kernkraftwerken Status analysis of quality assurance systems of the operators for construction and operation of nuclear power plants	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: VGB Essen HA "Wärmekraftwerke" AK "Statusbericht QS"

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.10.77	30.9.79	Dipl.-Ing. Plate

Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
continuing	31.12.78	

1. General Aim

Determination of the quality assurance (QA) over all phases of the nuclear power production (design, construction, operation). Deficiencies between individual QA-systems and/ or insufficient single systems influence the entire system. Therefore also a description of the QA-systems of the power plant operators and following preparation of a general status report is necessary.

2. Particular Objectives

Survey on measures of the power plant operators. Detailed description of the existing condition, on activities of the operators in the scope of the QA.

3. Research Program

Elaboration to the following technical fields:

3.1 General viewpoints on the QA of nuclear power plant-primary circuit components at layout, manufacturing and construction, start-up and operation, including maintenance.

3.2 Performance of the QA measures

3.2.1 System of the QA of the operator at manufacturing, construction, start-up and operation of nuclear power plant facilities.

3.2.2 Quality organization
3.2.3 Ordering
3.2.4 Design principles
3.2.5 Construction and calculation
3.2.6 Fabrication and testing
3.2.7 Transportation, packing, preservation
3.2.8 Construction
3.2.9 Start-up
3.2.10 Operation
3.2.11 Measures in the cases of quality miss match
3.2.12 Maintenance
3.2.13 Alteration and extrusion of plant parts
3.2.14 Fuel elements
3.2.15 Repeating tests
3.2.16 Documentation
3.2.17 Control of the QA-systems of the operator and his deliverers
3.3 Valuation of the QA-measures, presently performed by the operator and within the scope of the licensing procedure

4. Experimental Facilities and Computer Codes

5. Progress to Date

to 3.- All chapters were written in a first version. As sponsor asked for more details writing a new version was started. Until now the chapters 1 - 2.13 were formulated in the new version.

6. Results

to 3.- The chapters 1 - 2.13 are written. Moreover appendixes were procured for illustration of the report

7. Next Steps

to 3.- Elaboration of chapters 2.14 and following. Insertion of appendixes existing and to be procured. Consideration of the cross-linkings between the engaged power plant operators. Diskussion for the final version

8. Relation with Other Projects

RS 124 KWU

as well as analogue orders of BMI

9. References

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10. Degree of Availability of the Reports

-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	12.1	RS 263/10
Vorhaben/Project Title Analytische Tätigkeiten der GRS im Rahmen des Reaktorsicherheitsforschungsprogramms des BMFT Rechnergestütztes Störungsanalysesystem Analytical Activities of the GRS in the Frame of the BMFT Research Program on Reactor Safety Computer Supported Disturbance Analysis System		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit (GRS) mbH
Arbeitsbeginn/Initiated 1.1.1977	Arbeitsende/Completed 31.12.1980	Leiter des Vorhabens/Project Leader Dr. R. Grumbach
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 31.12.1978	Bewilligte Mittel/Funds

1. General Aim

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

2. Particular Objectives

The prime subject of this project is the development of procedures and computer-based systems for a fast analysis of disturbances in a nuclear power station. The on-line analysis should yield primary causes, the present situation and the expected propagation of disturbance. The analysis of disturbance can be autonomous, i.e. exclusively by a process computer and without interference by the operating staff, or interactive, which implies the possibility of additional information display upon a request from the operating staff.

3. Research Program

During 1978, the following subtasks were pursued (the decimal numbers following the chapter number refer to the original numbering of tasks in the overall working plan):

3.4 Intermediate installation of the hardware configuration for the disturbance analysis system at the GRS laboratories at Garching. This configuration is to be shipped to the nuclear power station at a later date.

3.5 Adaptation of the program system for disturbance analysis, ALSAN, to the frame specifications of the disturbance analysis system.

3.6 Development of a structure for the program for the communication module ALKOM.

3.7 Development and adaptation of the program system for the data preparation, MOGEN, to the codes ALSAN and ALKOM.

3.8 Development and implementation of control routines for the coordination of the different modules of the disturbance analysis system.

3.9 Development of strategies for a specific training of the operators in the usage of a disturbance analysis system.

3.10 System analysis of those parts of the nuclear power station which shall be covered by the disturbance analysis system.

3.11 Evaluation of cause consequence models on the basic of the system analysis of task 3.10.

3.12 Real time tests of the on-line codes ALSAN and ALKOM, and the related control routines.

3.13 Functional tests of ALSAN and ALKOM.

3.14 Installation of the hardware/software configuration at the Grafenrheinfeld nuclear power station.

4. Experimental Facilities, Computer Codes

Ad 3.5 The ASLAN version to be used in this project is based upon a software package developed within the PDV-project P6.1/24.

Ad 3.10, 3.11 5 subsystems of the Grafenrheinfeld nuclear power station have been analyzed by the plant vendor, employing the cause-consequence analysis method. The results of this analysis, which was carried out in close cooperation with GRS staff, have been made available for the setting up of the input to the model generator program MOGEN.

Ad 3.12, 3.13 The experimental process computer facility at the Halden project, comprising a model control room and a real-time simulator, have been made available for the execution of functional and real-time tests of the programs ALSAN and ALKOM.

5. Progress to Date

Ad 3.5, 3.8 The on-line analysis routine ALSAN and the related control routines have been implemented and are available in an

6. Results

Basic operational versions are available of the functional routines MOGEN, ALSAN and ALKOM, together with the corresponding control routines. Cause-consequence diagrams have been established for 5 subsystems of the Grafenrheinfeld plant; the diagrams are in a form which is compatible with the input requirements of the disturbance analysis system. The hardware configuration of the disturbance analysis system, including the communication system, is in an operational status.

7. Next Steps

During 1979, main efforts will be concentrated on the integration of the available functional software modules, their testing in realistic simulations using event chains evaluated in the system analysis of the Grafenrheinfeld plant, and the installation of the hardware and software configuration at the plant site. During the commissioning of the plant, in the second half of 1979, selected test schemes may be executed at the power station in connection with the experimental, non-nuclear operation of plant subsystems. Otherwise, validation tests and real-time testing employing the operators' dialog procedures will be carried out with the help of simulation facilities available at the GRS laboratories at Garching and at the experimental computer control facility of the Halden project.
It is expected that, during 1979, the tests at the plant and at the simulation facilities will yield some preliminary practical experience permitting a first critical review of the prototype system by end 1979.

8. Relation with other Projects

The development of the disturbance analysis system by GRS has been carried out in close coordination with the research and development activities on operator-process communication pursed at the Halden project. The dialog part, comprising the communication module ALKOM, was developed as a subproject of the international working program of the Halden operational version. Further adaptations are being carried out in the course of the integration with the model generator program MOGEN and the dialog routine ALKOM.

Ad 3.6 The development and implementation of the communication routine ALKOM has been completed. ALKOM has been interfaced to the analysis program ALSAN and functional tests of the two routines have been started.

Ad 3.7 The model generator MOGEN, a program system for the preparation of the data base for the on-line disturbance analysis, has been completed and is available as an operational module.

Ad 3.8 See above

Ad 3.8 Investigations as to the education and training of plant operators in the usage of advanced on-line analysis systems have been conducted throughout the year. Part of this work has been carried out, in the frame of a subcontract, by Institutt for Atomenergi, Halden, Norway, and is documented in an intermediate report.

Ad 3.10 In cooperation with Kraftwerk Union, system analyses were carried out on the feedwater system and the main condensate system of the Grafenrheinfeld power plant. These activities are continuing.

Ad 3.11 The cause-consequence model for the feedwater system has been further developed and a first version of the cause-consequence model for the main condensate system has been produced. The models are being adapted to the results of the continued system analysis work at Kraftwerk Union.

Ad 3.12 Preliminary tests, which have so far been limited to the communication module ALKOM, have been taken up in connection with the real-time simulator available at Halden.

Ad 3.13 Functional tests of the ALSAN and ALKOM routines have been started, using simplified logic test inputs.

Ad 3.14 The hardware configuration, consisting of a process computer (Siemens-330) for the current analysis and a color-display communication system (with a NORD-12 mini computer) has been established and interfaced, and works according to specifications.

project and has been integrated with the software developed by GRS. In addition to its installation and plant operation at the Grafenrheinfeld nuclear power plant, the GRS disturbance analysis system will be implemented at the Halden project for further studies and development in the field of computer-based operator communication. In particular, test with that system will support the futher evaluation of user procedures for computerized surveillance and status analysis of nuclear plants.

. References

L. Felkel, R. Grumbach, E. Sädtler, D. Wach: Treatment, Analysis and Presentation of Information about Components Faults and Plant Disturbances. IAEA Symposium on NPPCI, Cannes, 1978

F. Oewre, L. Felkel: Functional Description of the Disturbance Analysis System for the Grafenrheinfeld Nuclear Power Plant. Enlarged Halden Program Group Meeting Loen, 1978

10. Degree of Availability of the Reports

Available as proceedings of the mentioned meetings.

	CLASSIFICATION: 12.2.
TITLE (ORIGINAL LANGUAGE): Bouchonnage par explosifs des tubes de Générateurs de Vapeur.	COUNTRY: Belgium.
	SPONSOR: Ministry of Economic Affairs
TITLE (ENGLISH LANGUAGE): Explosive Plugging of Steam Generators Tubes.	ORGANISATION: S.A. COCKERILL.
	PROJECT LEADER: R.V.SALKIN.

INITIATED: 1976.	COMPLETED: 12-1979	SCIENTISTS: Claude HICTER.
STATUS: In progress	LAST UPDATING:	

) GENERAL AIM : To develop a new type of explosive plug with high reliability and extended welded zone.

) PARTICULAR OBJECTIVE : To investigate different cases of steam generators on use on the European continent.

) EXPERIMENTAL FACILITY : - Facilities of Centre de Technologies Nouvelles, Liège (Belgium).
- Mock up at Tihange Power Station.

) PROJECT STATUS : - Procedure agreed for one type of S.G.
- Investigations in progress for other types.

) NEXT STEPS : Acceptance tests by the official authorities in Belgium for SENA and Doel Power Stations.

) RELATION TO OTHER PROJECTS AND CODES : Nil.

) REFERENCE DOCUMENTS : British Nuclear Energy Society Conference, London, 5 April 1979.

TITLE (ORIGINAL LANGUAGE):		CLASSIFICATION: Water reactor 12.3
Caractérisation des équipements ultrasonores		COUNTRY: Belgium
		SPONSOR: -
TITLE (ENGLISH LANGUAGE): Characterization of ultrasonic equipment		ORGANISATION: Association Vinçotte
		PROJECT LEADER: P. CAUSSIN
INITIATED: 1975	COMPLETED: 1980	SCIENTISTS: D. VERSPEELT
STATUS: In progress	LAST UPDATING: May 23, 1979	

1. General aim

Development of the measurement techniques of ultrasonic equipment characteristics for assessing the performances and the reproductibility of inspections.

2. Particular objectives

Development of accurate procedures in compliance with the US-NRC 10 CFR 50 and ASME Codes. Analysis of the actual influence of the equipment characteristics on the performances achieved.

3. Experimental facilities and program

Facilities :

- Acquisition of electronic measuring equipments (spectrum analyser, vector impedance meter, ...)
- Development of equipments for measuring the ultrasonic beam spread in steel for metal paths ranging between 0 and 350 mm, and refracted angles between 0 and 80°
- Development of electronic equipments for recording the beam spread measurement results and for gating the echoes.

Program :

- Development of procedures for accurately checking the characteristics of ultrasonic equipment : apparatus, cables, probes.
- Establishment of a scheme for the periodic inspection of the equipment.
- Analysis of the influence of the equipment characteristics on the performances of the inspections.
- Establishment of a specification for good and reliable equipment.

4. Project status

The measuring equipment is available and the characterization scheme has been defined.
Ultrasonic equipment with independently variable characteristics was developed.
It was established that :

- the characteristics strongly affect the performances of the inspections
- commercially available ultrasonic equipments, which are nominally identical, can have very various characteristics and performances.

5. Next steps

Quantitative analysis of the influence of various characteristics on the performances and the reproductibility of ultrasonic inspections.

6. References

Characterization scheme of an ultrasonic equipment for industrial application

P. CAUSSIN

Proceedings of the Ispra Courses on Characterization of Ultrasonic Equipment, CEC-JRC-Ispra, June 1-3, 1977, 37 p.

7. Availability

Details available at

ASSOCIATION VINCOTTE
Département Etudes
B - 1640 RHODE-SAINT-GENESE (Belgium)

	CLASSIFICATION: Water reactor 12.3
TITLE (ORIGINAL LANGUAGE): Participation au programme du Plate Inspection Steering Committee (PISC)	COUNTRY: Belgium/CEC
	SPONSOR: Association Vinçotte CEC
TITLE (ENGLISH LANGUAGE): Participation in the program of the Plate Inspection Steering Committee (PISC)	ORGANISATION: Association Vinçotte
	PROJECT LEADER: P. CAUSSIN
INITIATED: 1974 / COMPLETED: 1980	SCIENTISTS: A. VINCKIER, J. CERMAK, H. SWEERTS
STATUS: In progress / LAST UPDATING: May 23, 1979	

1. General aim

The PISC program aims at assessing the reliability of the ultrasonic examination of reactor pressure vessel welds.

2. Particular objectives

- Statistical evaluation of ultrasonic testing results obtained with a procedure complying with the ASME code, Section XI requirements (PISC procedure)
- Comparison of the results obtained with the PISC procedure and with alternative European procedures.

3. Experimantal facilities and program

The US-Pressure Vessel Committee (PVRC) made available 3 test blocks :

50 - 52 : electroslag weld
51 - 53 : submerged arc weld
204 : set-in nozzle weld

with intended natural defects.

The PISC organized round robin tests involving more than 30 inspection teams from about 10 European countries. The blocks were destructively examined to provide a basis for the evaluation of non-destructive testing results. A detailed metallurgical analysis was performed in a few cases.

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The Association Vinçotte participated in that program in the following way :

- Inspection of the blocks according to the PISC procedure
- Inspection of the blocks according to house procedures
- Destructive examination of part of the block 51-53 under the sponsorship of the Commission of European Communities and in co-operation with the Laboratory for the Strength of Material of the Ghent University
- Analysis of the influence of the plate segregation on the ultrasonic shear wave propagation
- Organization of the Belgian participation
- Participation in the organizing committee and sub-committees

4. Project status

- The round robin tests are completed
- The blocks were destructively analyzed
- The PISC procedure results were analyzed

5. Next steps

- Analysis of alternative NDT methods
- Analysis of the significance of some peculiar flaws

6. Degree of availability

The reports will be published by the CEC and the OECD by mid-1979.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1978 - 31.12.1978	12.3	RS 222

Vorhaben/Project Title	
Analysis of possible improvements of objectivity and documentation of non-destructive-testing methods for reactor components of the primary circuit Analyse bestehender Materialprüfverfahren auf Objektivierbarkeit und Dokumentierfähigkeit für Reaktorbauteile des Primärkreislaufes	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Dornier System GmbH Friedrichshafen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
10.1.1977	31.3.1979	Dr. Sahm
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aims

Feasable methods of non-destructive-testing (NDT) during the work inspection of reactor components of the primary circuit are to be analysed either with respect to objectivity and automatic control of the measuring process either with respect to objective and automatic documentation of the test results. General aim of this analysis is to eliminate individual influences in the test process and to reduce time and cost consumption in NDT.

2. Particular Objectives

An extensive documentation of the actual NDT status is basically used to define the problems concerning objectivity and automation in NDT. A careful inventory of the applied inspection techniques will be especially important in this context. Solutions to the above problems will be given by a critical review of all feasable NDT methods. The usability of future inspection techniques not yet specified in NDT of reactor components, will be analysed with regard to physical limits of flaw detection and flaw diagnostic and to technical realization.

3. Research Program

Actual test situations are fundamental to the research program. Hence the first part deals with the definition of typical representative test problems. The analysis will then proceed in two steps. First the actual situation of NDT will be reviewed and secondary the future development will be analysed. The research program includes the following points:

Estimating the degree of objectivity of NDT methods the following useful procedures have been carried out:

- analysis of the test operating sequence
- a list of queries
- visits of practical NDT tests
- interviews with NDT-engineers
- statistical analysis of US-multiple test results.

6. Results

The evaluation of a large number of individual test situations lead so far to 18 different typical test problems. Each of them is described by a prescribed form containing dimension, geometry, materials, possible flaws (with regard to the flaw catalogue), test sequence, test methods, test specifications, degree of objectivity, degree of automation and special problems.

The flaw catalogue is arranged according to natural flaws such as cracks, pores and so on analogous to DIN 8524. A well defined flaw is characterized by a number of 6 digits. The first three describe the flaw typ, the fourth describes the flaw shape (6 classes), the fifth describes orientation (7 classes) and the sixth describes the flaw position (6 classes).
Additionally the catalogue contains sizes and frequency of flaws.

The analysis of NDT methods results in a preliminary list of the weakest points of manual Ultrasonic testing and surface crack testing. These are for example: personal qualification, recording and documentation of results, handling of test equipment and poor reproducibility of instrumentation properties.

In most cases the design of test equipment and the testing performance do strongly neglect ergonomical aspects.

7. Next steps

- completation of the typical test problems by evaluating single test problems of further components
- continuation of the analysis of NDT methods with respect to objectivity and automation. Listing of the most important problems together with possible solutions.

2. Review of significant NDT procedures and their background during the work inspection of reactor components for different types of reactors
2.1 LWR
2.2 LMFBR
2.3 HTR
2.4 List of representative test problems
3. Definitions
3.2 Analysis of NDT methods
3.2.1 Objectivity of applied methods
3.2.2 Fundamentals and physical limits of NDT methods
3.2.3 Actual standard of NDT methods
3.3 Analysis of future development
3.4 Analysis of problems

4. Experimental Facilities, Computer Codes

5. Progress to date

In cooperation with KWU, Erlangen, HRB, Mannheim and Interatom, Bensberg all nondestructive test problems existing in components of the primary circuit are collected and listed in a prescribed form (2.1 - 2.3).

In a following step these "single test problems" are summarized to typical, representative test problems no more related to specific components (2.4).

Also in cooperation with KWU an extensive flaw catalogue has been made up, which correlates to the single test problems as well as to the typical test problems (2.4).

The analysis of applied NDT methods has been carried out under uniform aspects in cooperation with BAM, Berlin and IzfP, Saarbrücken (3.2, 3.3). Especially the fundamental and physical limits as well as the status and future potential of development have been investigated.

8. Relation with other Projects

- Qualitätssicherung - Darstellung des Istzustandes
- Studie über Personalqualifikation für zerstörungsfreie Prüfungen im Bereich Reaktorsicherheit (RS 216)

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1977 - 31.12.1977	12.3	RS 169

Vorhaben/Project Title	
Entwicklung und Bau einer Ultraschall-Prüf- und Auswerteeinheit einschließlich Vorverstärkerkasten. Development and construction of an ultrasonic test- and evaluation electronic including preamplifier box.	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Krautkrämer GmbH Köln

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.7.1975	31.3.1979	Ing. grad. G. Gutmann
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

Developing an ultrasound electronic evaluation system to meet current safety requirements and being universally employable with all types of reactors.

On the basis of experiences gathered up to date and of the technical development foreseeable for the future this electronic evaluation system is to meet the following requirements:

2. Particular Objectives

Maximum number of channels: 60

Maximum travel rate of the manipulator mechanism: 100 mm/sec.

Scanning rate: one (1) shot per millimeter.

Also the system shall continuously be monitored for performance and stability.

Diminution of preamplifier unit should avoid interference.

3. Research Program

- Development of the detailed performance scheme with network map.
- Development of a number of new analysing modules.
  a) Min./max. value store
  b) Interference analyses
- Development of mini-transmission equipment and preamplifiers.
- Development and design of the electronic evaluation system in accordance with the stipulated requirements.
- Testing interaction of the developed modules.
- Planning work for prototype building.

4. Progress to Date

The works of the research program as specified under 3. as well as the design of the testing and evaluation electronic system have been concluded.
The works concerning these two points have been delayed due to re-checking of research program RS 102/17.

5. Results

All electronic modules especially required for this electronic system have been developed and tested as functional samples.
The requirements relating to performance, accuracy and stability of the modules were met.
Designing and development work for the complete task was continued, the following major requirements have been met:
a) maximum number of channels: 60
b) maximum travel rate of the manipulator mechanism: 100 mm/sec.
c) scanning rate of the testing system of one (1) shot per millimeter
d) fully-automatic and continously effevtive performance and stability monitoring
e) maximum acceptable stability tolerance of ± 1.5 dB.
f) wiring schemes in hybride have been made and transformed to original size.

Designing work for the building of the prototype evaluation electronic was terminated.
Planning for pre-amplifier unit miniturizing was terminated and comprised into a funds raise application.

6. Next Steps

a) Pre-amplifier box
- testing of pre-amplifier unit miniaturized in a probe under prototype conditions.

b) Electronic
- commencing prototype building
- testing the electronic system

c) Testing the complete system.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	12.3	RS 266

Vorhaben/Project Title	
Entwicklung und Bau einer US-Prüfeinrichtung auf Laser-Basis zur Prüfung von Reaktorkomponenten Development and Construction of Ultrasonic Test Equipment Based on Lasers for Testing Reactor Components	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Krautkrämer

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.7.77	31.12.79	Kaule
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	DM 2.413.106

1. General aim

With conventional ultrasonic reactor tests the probes used for transmitting and receiving sound have to be coupled to the material to be tested by means of a coupling agent. Lasers used as sound generator and receiver allow a non-contact coupling. Due to non-occurring variations in the coupling the reflectors are reliably detected with this method which can be used especially with high temperatures and complicated geometries as well as narrow gaps. The reflectors can be directly analyzed on account of the provided control of the lasers. The aim of the project is to construct the prototype of ultrasonic testing equipment, based on lasers for testing reactor components, which is planned to be tested on-site.

2. Particular objectives

For generating sound by means of laser pulses lasers are required with a high pulse power and a high pulse repetition rate. Such lasers are technically realizable but expensive. In order to optimize the available lasers for this application the relations between the individual parameters with the generation of sound shall be examined throughly so as to be able to produce the sound pulses required for a special test assignment with the least possible expenditure in view of the lasers.

3. Research programme

3.1.3 Transmittance or, resp. losses in the light conducting components

3.2 Determining the laser output capacity required for the method

3.2.1 Examining the relationship between the optical power density on the surface of the material and the generated sound amplitude

3.2.3 Examining for surface damage and changes in the grain structure depending on the power density on the surface of the material

3.2.4 Examining the dependency of the time constant of the plasma from the power density on the surface of the material

3.3 Determining the shape of the laser pulse best suited for the method

3.3.2 Examining the influence of the width of the light pulse on the shape of the generated sound pulse

3.3.3 Examining the influence of the falling flank of the light pulse on the shape of the generated sound pulse

3.3.4 Examining the suitability of the generated ultrasonic spectrum for the current test assignment

3.3.5 Selection of a suitable laser based on para. 3.3.1 to 3.3.4

3.3.6 Selection of a suitable Q-switch arrangement based on para. 3.3.1 to 3.3.4

3.4 Determining the largest possible pulse repetition frequency which can be obtained using this method

3.4.1 Examining the time characteristic of the plasma in front of the surface of the material - especially the decaying time

3.4.2 Testing for possible ultrasonic side effects caused by low frequencies when the plasma is forming and decaying

3.4.3 Examining the results of thermal effects on the laser depending on the pulse repetition frequency

3.5 Determining a suitable method of generating the directional characteristic of the sound beam as required by the test assignment

3.5.1 Examining the guiding characteristic with the Gauss distribution of the intensity above the cross section of the laser beam

3.5.2 Examining the guiding characteristic with the multi-mode output of the laser

3.5.3 Examining the guiding characteristic with the simultaneous excitation of partial areas

3.5.6 Selection of suitable methods for illuminating partial areas

3.5.7 Examining the effect of a time delay when exciting partial areas.

4. Experimental facilities - computer codes

To 3.2 The laser system for 100 MW output capacity has been completed. The system now consists of the following stages: One oscillator and two subsequent amplifiers both of which are equipped with glass rods marked Nd as the laser medium. The oscillator is equipped with a Pockels cell Q-switch device for generating giant pulses in the duration range of some 10 ns. The resulting output energy is between 380 and

650 mJ - corresponding light powers of from 9 to 21 MW. After double stage amplification the system delivers pulse energy of a maximum of 2.3 J, corresponding to a power of 100 MW. The pulse sequence is 0.1 Hz.

In order to measure the cross section of the sound beam a diffusing screen is placed in the beam and is scanned by means of an infrared sensitive TV camera. An electronic attachment is provided by means of which the brilliance curve along a previously selected line on the TV display is shown on an oscilloscope. The half value width of the beam in the scanning direction can be obtained from this screen display.

To 3.3.2 By converting the laser oscillator to Nd-YAG as a laser medium pulse durations can be obtained down to 8 ns half value width (HVW).

To 3.4 After previous clarification of the technical realization of high pulse sequences with power pulse lasers and the reliability of such systems an Nd YAG laser system for a pulse repetition frequency of 50 Hz was obtained. It consists of an oscillator and an amplifier with YAG rods as the laser media. The pump light lamps for both rods are fired simultaneously. After an adjustable delay time, which has been optimized at 200 μs, the Pockels cell Q-switch opens the laser resonator within a few ns. With a pulse width of approx. 15 ns the energy delivered per pulse is 400 mJ. The pulse repetition frequency is adjustable up to 50 Hz.

To 3.5 A device was provided for measuring the directional characteristic of the generated sound beam.

To 3.5.1 A gauss distribution of the intensity above the cross section of the sound beam can only be given in the TEM00-mode. By using a mode diaphragm attempts were made to prevent the oscillator from oscillating out of this mode.

To 3.5.3 A suitably illuminated grid-shaped screening was provided for local intensity modulation.

To 3.5.6 After clarification of the feasibility radiation-resistive optical components were ordered in order to be able to build a device for local intensity modulation by subjecting the laser light to interference.

To 3.5.7 An amplifier head of the 100 MW system was converted into an oscillator. A triggering device was made for releasing two Q-switch Pockels cells in a timed sequence.

5. Progress to date

To 3.1.3 The losses for laser pulses using glass fibre conductors sere measured.

To 3.2 Regarding experiments for 3.2 and 3.3 the laser power and pulse width have to be varied over as wide a range as possible in order to guarantee coverage of the interval important for the measurement. For these trials tentatively a suitable laser system was built from commercially available modules. Thereby the large variation width in the capacity could only be obtained at the expense of the pulse sequence. This did not, however, prevent the experiments from being carried out.

It was seen though that in the system individual parts were heavily stressed. At the highest output capacity the load is higher than with normal operation, for which a long component service life is guaranteed. This limits the service life of the individual components. For this reason, contrary to the original plan with several measuring rows, the execution of the trials has to be such that all the relevant data is recorded with one laser pulse.

To 3.2.3 Different sample plates have already been subjected to the effects of laser pulses.

To 3.3 In the operating data the system has been so optimized that for each trigger pulse only one single pulse, free from prior and after pulsing, is generated.

To 3.3.3 The light pulse and sound pressure curves were measured and compared.

To 3.4 In 50 Hz operation and with 400 mJ energy/pulse examinations were carried out of the service life of the components and the stability of the beam parameters.

To 3.4.3 The power and the divergence were measured with different pulse sequences.

To 3.5 The device for measuring the directional characteristic was tested for correct operation.

To 3.5.1, 3.5.2, 3.5.3 The planned measurements were carried out.

To 3.5.7 The delay between the laser pulses and the pulse jitter was measured with the operation of two oscillators from a common triggering device.

6. Results

3.1.3 The established attenuation was within the limits given in the data sheets. Suitable types of glass fibre are available for future application of infrared light (λ = 1060 nm).

To 3.2 The relationship between the laser energy beamed onto a specific area and the sound pressure generated is given in Fig. 1. Three sections can be seen:

1. between approx. 0.01 J and approx. 0.1 J the range of linear relationship between laser energy and sound amplitude,
2. Between approx. 0.1 J and approx. 1 J the range of overproportional increase of the sound energy combined with the formation of plasma,

3. Over approx. 1 J the range of decreasing effect for the sound conversion.
   Apart from the formation of plasma the sound generation is influenced by the application of layers of foreign material such as water. The transition between the ranges 2 and 3 is joined to the beginning of material denudation to a varying degree depending on the material and surface property.

To 3.2.4 The life of the plasma, measured as a half value width of the visible plasma light, varies between 50 ns with approx. 0.7 J laser energy (35 MW capacity) and 80 ns with approx. 1.9 J (95 MW capacity) for a sound beam cross section of approx. 14 mm^2. Half value widths resulted of 70 ns with approx. 0.1 J (5 MW) to 350 ns with approx. 1.8 J (90 MW).

To 3.3 In order to suppress prior pulses the damping in the Q-switch device must be at least 1:100. After pulses are effectively suppressed if, when opening the Q-switch, the pump light is already dying away. The highest power of the after pulses can be 100 W, otherwise the resulting sound pulse can be mistaken for a flaw echo. By optimizing the operating data of the laser they have been completely suppressed.

To 3.3.2 In the range between 8 ns and 20 ns pulse duration (HVW) the generated sound amplitude depends only on the highest power and not on the duration of the laser pulse. However, the frequency content of the generated shock wave depends on the pulse width.

To 3.3.3 Contrary to the drop in the light pulse the drop in the sound pressure is delayed by the formation of plasma, e.g. if a 23 ns HVW laser pulse with an energy density of 1.8 J on 15 mm^2 generates a 30 ns HVW whereas the same laser pulse with an energy density of 0.1 J on 15 mm^2 generates a sound pulse of 15 ns HVW.

To 3.3.4 It has been shown that the ultrasonic spectrum generated with YAG 15 ns HVW laser pulses has a higher frequency portion than is used with normal ultrasonic testing - however it is most favourable for high resolution flaw detection.

To 3.3.5 and 3.3.6 A YAG laser with a Pockels cell equipped Q-switch was found to be suitable.

To 3.4 With the system in 50 Hz operation the output power and beam parameter are constant for over 50 hours. After 100 hours operation there was a slight power drop - however damages to the optical components were not noticed.

To 3.4.1 Experiments have shown that the effect of duration of the plasma of approx. 50 ns has no effect on a high pulse repetition frequency.

To 3.4.2 Low frequencies which are dependent on plasma were not established at any amplitude worth mentioning.

To 3.4.3 At a high pulse sequence the thermally reduced changes of the refraction index in the laser material cause changes in the optical properties of the laser rod. For a previously given pulse repetition frequency in the range of up to 50 Hz this change however is constant and can thus be compensated for.

To 3.5 The measuring device for the directional characteristic meets with the requirements.

To 3.5.1 As the power produced in the TEM00-mode is too low and let thermal effects on the laser rod result in the instability of the beam this mode of operation cannot be used for the intended application.

To 3.5.2 The distribution of the intensity over the cross section of the beam shows a wide, clipped maximum with a steeper drop at the edge. As the speed of light is very large when compared to the sound velocity the sound radiation, independent from the refraction angle of the light, always enters the material perpendicularly to the surface. This is one advantage of the method. The directional characteristic is identical with that of a shock wave probe with the same area of excitation and the same frequency spectrum.

To 3.5.3 There was a sound distribution of several diffraction orders which is symmetrical to the perpendicular sound beam. Certain diffraction orders can be suppressed by a suitable choice of the gap width.

To 3.5.7 The triggering device met with the requirements. The uncertainty in the delay between both laser pulses is a maximum of 2 ns and therefore small when compared to the pulse width.

7. Next steps

To 3.2.3 Examinations of the grain structure are to be carried out on austenite samples which had been subjected to the effects of laser pulses.

To 3.4 The life tests, power measurements, and measurements of the sound beam parameter are to be extended over times of more than 100 hours.

To 3.5.3 The examinations are to be continued.

To 3.5.6 Other methods are to be examined.

To 3.5.7 The effect of a delay when exciting the partial areas is to be examined.

8. Relation to other projects

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9. References

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10. Degree of the availability of the reports

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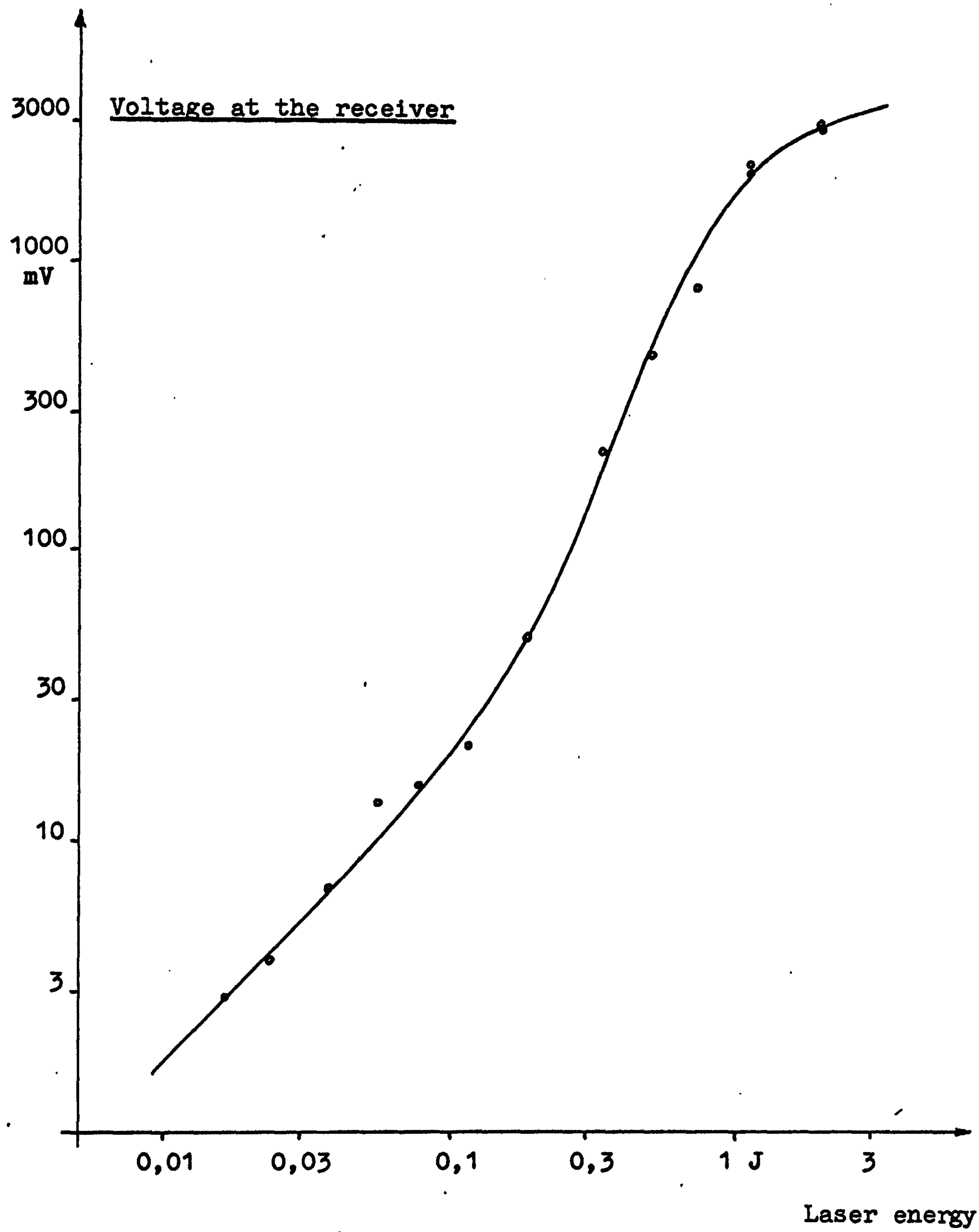


Fig. 1:

Voltage at the receiver as a function of the beamed laser energy pulse.

Half value width of the pulse: approx. 20 ns

Excited area: Ellipse appr. 5.3 x 2.7 mm^2 = 14 mm^2

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1978 - 31.12.1978	12.3	RS 2703

Vorhaben/Project Title	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: BAM, Berlin; Krautkrämer, Cologne; KWU, Erlangen; M.A.N., Nürnberg
Weiterentwicklung von Ultraschall-Prüfverfahren und dazu erforderlichen Einrichtungen für wiederkehrende Prüfungen an Reaktordruckbehältern Onward Development of Non-Destructive Testing Methods for In-Service Inspection of Reactor Power Plants	

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.2.1976	30.9.1978	Dipl.-Ing. J.Lindner/M.A.N.
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	December 1978	

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:

1. Basic studies of ultrasonic inspection methods and optimization of detection and analysing techniques.

2. Conceptual design of inspection systems to improve the probability of fault detection in different inspection areas.

3. Development, fabrication and trials of internal and external inspection manipulators with a view to achieving the goal of complete volumetric inspection of the reactor pressure vessel by automated equipment.

4. Onward development of components and methods to improve data logging, data processing and display/ printout systems.

3. Research Programme

The research programme embraces the following work:

3.1 General

The general aim of the research programme is to make further progress in the development of ultrasonic inspection techniques with a view to increasing the probability of fault detection and improving the interpretation of the inspection results. This includes research into techniques for the analysis of indications, e.g., focus technology or line holography, and preparations for practical application. The tasks call for extensive theoretical and experimental research.

3.2 Inspection Systems

Special inspection systems must be developed for the inspection of close-to-surface zones and of lugs and supports welded to the inside and outside of the reactor pressure vessel. The task involves testing a method of monitoring the stability of the probes during the course of inspection work. The task also includes the development of a probe with adjustable sound beam parameters for special inspection applications, e.g., for inspecting the ligaments in areas of penetration holes for control rods.

3.3 Manipulators

New manipulators are being developed and current manipulators are being improved for remote-controlled inspection of the following areas of reactor pressure vessels:

- Ligaments in areas of control rod penetration holes in the bottom closure and closure head
- External inspection of transition weld between cylindrical area of vessel and spherical bottom (BWR)
- Inspection of inner radii of small nozzles
- External inspection of flange at bottom of vessel and at closure head
- Extended external inspection of cylindrical area of vessel with automatic transfer of inspection car from one longitudinal rail to another. Motions under computer control.

3.4 Data Logging

For processing the data of automatic ultrasonic inspection the capacity of data logging equipment must be increased to accommodate the increased flow of data. Existing programmes are being increased to improve software and the interpretation quality of display and printouts.

4. Experimental Facilities, Computer Codes

The research programme envisages laboratory tests to establish the capability of inspection equipment for [illegible]al 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

Basic work carried out on ultrasonic inspection,which is being given preference in recurrent inspection of reactor pressure vessels,included experimental studies of flaw detectability and interpretation of indications on specially made test blocks. The test flaws were partly circular and partly elliptical. The results obtained were compared with those predicted on the basis of theory.

With a view to the further development of probe modules, noise and V-pattern tests were made on an original component representing present standards. In this connection, measurements were also made to inspect the areas under the lugs on the reactor pressure vessel when the crystals of the probes were systematically changed.

Special efforts were made to examine the inspectability of near-surface zones including, in particular, the zone near the cladding.

A special test programme was drawn up for a probe developed under the project with variable sound parameters.

Experiments were made to establish to what extent flaw size can be determined by means of focussing probes.

New boundary conditions have been found in the development of manipulators, especially the universal manipulator for OD inspection of the cylindrical parts of the reactor pressure vessel and nozzles, which partly considerably affect the concept so that a partial revision of the concept has become necessary.

An improved design has been developed of a pipe-to-nozzle weld manipulator for use in restricted space.

Work on adapting the data acquisition system to suit the extended ultrasonic electronics system has been continued.

6. Results

The graph prepared for elliptical flat reflectors showed good agreement with experimental results.

The results of flaw detection in the region close to the cladding were compiled in a technical report.

Flaw size determination with focussing probes showed that, when using relative thresholds of 12 and 20 dB in line with practical inspection, flaw sizes measured were on the high side with considerable relative variations.

Measurements made in examining the zones under the lugs have shown that it is indispensable for a systematic investigation into flaw detectability and to determine the causes of indications that a test block is prepared which reproduces the original material and geometric conditions and contains defined artificial reflectors.

While work on the manipulator for the inspection of top head closure, the top head closure flange and the pressure vessel flange has been completed, it is necessary, because of the changed boundary conditions, to continue the development work for the universal manipulator with automatic transfer facility from one longitudinal track to another, the manipulator for extended ligament inspection and the manipulator for the inspection of the bottom closure weld.

Work to adapt the data acquisition system to the extended ultrasonic electronics system has been completed.

7. Next Steps

The project RS 2703 was completed in 1978. However, since some items of the work could not be completed, this part of the work is being continued under a follow-up project RS 2704. This will be covered by a separate report.

8. Relation with Other Projects

RS 169, RS 273, RS 298, RS 349

9. References

During the period under review, the following summarizing reports were prepared in the German language:

Author	Title	Date
BAM	Vergleich von Testreflektoren und Prüftechniken für den plattierungsnahen Bereich (Comparison of test reflectors and inspection techniques for zone near cladding)	Jan.1978
KK	Stabilität der Prüfköpfe.Entwicklung eines Systems zur Überwachung der elektrischen und akustischen Prüfkopfeigenschaften (Stability of probes. Development of a system to monitor electrical and acoustic probe properties)	March 1978
KK	Versuche zur Fehlergrößenbestimmung aus mit fokussierenden Winkelprüfköpfen für Direktkontakt gemessenen Amplituden-Echodynamiken (Tests to determine flaw size from amplitude echodynamics measured with focussing direct contact angle probes)	June 1978
KK	Verbesserung der US-Prüfung der prüfkopfseitigen oberflächennahen Zone dickwandiger Reaktorkomponenten durch Verwendung von Prüfköpfen mit schwächerem seitlichem Empfindlichkeitsabfall (Improvement of ultrasonic inspection of probe-side near-surface zone of thick-walled reactor components using probes with weaker lateral sensitivity attenuation)	June 1978

KWU	Untersuchungen zur Fehlererkennbarkeit bei der Prüfung im oberflächennahen Bereich der Anschweißstellen zum unteren Boden von Reaktordruckbehältern (DWR) (Investigations relating to flaw detectability when inspecting zones near the surface of lugs on bottom closure of reactor pressure vessels (PWR)	August 1978
M.A.N.	Weiterentwicklung zerstörungsfreier Prüfverfahren für wiederkehrende Prüfungen in Reaktoranlagen (Abschlußbericht RS 2703) Further development of non-destructive inspection methods for in-service inspection of reactor plant (final report RS 2703)	October 1978

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.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1978 - 31.12.1978	12.3	RS 2704

Vorhaben/Project Title	
Weiterentwicklung Zerstörungsfreier Prüfverfahren für Wiederkehrende Prüfungen in Reaktoranlagen Onward Development of Nondestructive Testing Methods for In-Service Inspection of Reactor Power Plants	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: BAM, Berlin Krautkrämer, Cologne KWU, Erlangen M.A.N., Nürnberg

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.4.1978	31.12.1980	Dipl.-Ing. J.Lindner/M.A.N
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

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:

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 complete research programme is divided into the following areas in line with the specified objectives:

3.1 Basic Information

Objectives included the study of ultrasonic techniques with a view to improving the probability of depicting flaws in thick-walled pressure vessels and to refine interpretation of results. This also involved the review of methods of analysing indications and preparing these for practical application.

3.2 Inspection Modules

For specific inspection applications, e.g. inspecting the ligaments in areas of penetrations or the analysis of indications, the probe with variable sound beam parameters is being further developed. Taking into account the experience gained in inspecting spherical closures, efforts are being made to optimize the inspection method. In addition, inspection methods are being developed for the inspection of studs, the inspection of nozzle radii from the outside, and the inspection of near-surface zones.

3.3 Manipulators

Manipulators are being developed (partly new and partly improved from existing designs) for outside diameter inspection. The development of individual manipulators is being completed by the construction of a prototype.

4. Experimental Facilities, Computer Codes

In addition to laboratory tests to determine the capabilities of individual testing methods, prototypes are being constructed of manipulators. The methods developed are proposed to be tested on a full-size mockup of a BWR bottom closure section.

Various computer programmes are being used in designing the probe modules.

5. Progress to Date

Using a special probe and basing on the 6 dB attenuation for surface inspection, work has been started on preparing a comparison of predicted and experimental results regarding the distance between scanning traces to be maintained.

Measurements made to detect flaws in through-thickness or nearly through-thickness orientation by means of single probe techniques (indications from crack edges) were evaluated with the aim of improving the evaluation algorism for flaw size of the indications obtained in recurrent ultrasonic tests.

Studies relating to the classification of indications have been started, the aim of classification being to decide whether the range of indications has to be investigated by means of further methods of analysis.

Work has been continued to develop the inspection of deeper levels of the wall zone near the surface by means of single probe techniques. In particular, a test block has been designed with elliptical flaws having different axial ratios and embrittlement surfaces. The flaws were introduced at various depths. This test block is intended to be used in testing the first probe prototypes using closely realistic flaws.

The mathematical relationships have been derived to develop a control programme for the probe with variable sound field parameters.

Investigations were made to optimize inspection of ligaments on the basis of experience gained in inspecting spherical closures.

Additional experiments to inspect near-surface zones were intended to provide further information on inspection possibilities in these areas.

For the purpose of inspecting the ligaments of reactor pressure vessels for boiling water reactors, it is proposed to develop a manipulator to permit the lateral extension of a suitable probe module into the adjacent lane between nozzles. During the period under review, work was carried out on this manipulator in respect of boom design and probe guidance.

Work has been started on basic studies on the use of digitally-controlled drives for computer-controlled manipulators.

6. Results

The comparison of predicted and experimentally observed values during the inspection of the near-surface zone failed to show sufficient agreement so that the computer models used hitherto will have to be modified.

The evaluation of indications from crack edges calls for additional experimental studies on the pressure vessel wall mockup.

The investigations in connection with the ligament manipulator have shown that a suitable boom design will permit the probe to be used for the inspection to be guided so that its axis will be substantially perpendicular to the spherical closure in all inspection areas. This will greatly facilitate inspection procedure.

7. Next Steps

Forthcoming work will concentrate on the following areas:

- Methods of classifying indications
- Investigations with the probe having variable sound parameters
- Optimization of ligament inspection
- Inspection of near-surface zones
- Design work on universal manipulator
- Design work on ligament manipulator for extended ligament inspection

8. Relation with Other Projects

RS 169, RS 273, RS 298, RS 349

9. References

No reports were completed during the period under review.

10. Degree of Availability of the Reports

Not applicable

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78 - 31.12.78	12.3	RS 247
Vorhaben/Project Title Non-destructive testing of three HSSt-plates Zerstörungsfreie Prüfungen an 3 HSST-Platten		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Fraunhofer-Gesellsch. München IzfP Saarbrücken
Arbeitsbeginn/Initiated 10.01.1977	Arbeitsende/Completed 30.04.1978	Leiter des Vorhabens/Project Leader Dr. G. Deuster
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

It shall be demonstrated (on 3 test plates of the American HSST-program) on international level how agreeing resp. different results of NDT corresponding to pretended specifications can be if every country makes the best of equipments and personnel available.

2. Particular Objectives

2.1. NDT analysis of the defect regions detected by standard procedures.
2.2. Defect description (shape, orientation, kind and structure).
2.3. Detectability, possibilities of defect description, limits of testing procedure.

3. Research Program

All standard procedures which are available at present and which are used in production and in base- and inservice inspection were used for the 3 HSST-plates (Fig. 1, 2, 3). As far as these are automated such a testing also was executed comparing to manual testing. Thereafter analysis procedures for interpreting of defect regions were used after detection of defect regions.

For inside defects:

3.1. US-impulse-echo, focus	IzfP, BAM, KWU, RWTÜV
3.2. graphical documentation (A-fig.)	IABG
3.3. US-tandem	IzfP, BAM, KWU

3.4.	evaluation, interpretation, reconstruction, ref. to RS 27	IzfP, BAM, KWU
	KWU-BAM, ref. to RS 102-17, IzfP	
3.5.	US-holography	IzfP, BAM
	optical reconstruction, numerical reconstruction	
3.6.	US-HF-scattering and absorption	IzfP
3.7.	Radiography (MPA/Stgt.)	MPA

For surface defects:

3.8.	magnetic particle test	RWTÜV, IzfP
2.9.	multi-frequency-eddy-current	IzfP
3.10	potential and magnetic leakage	IzfP
3.11	US-surface waves	

4. Experimental Facilities

Manipulator resp. data recording and data handling system from RS programs 102-16, 102-17, 102-18, 102-20, RS 27 (see correspond. reports), NDT-standard equipments of RWTÜV, radiographic instruments of MPA/Stgt.

5. Progress to Date

For the NDT recording of the 3 HSST test pieces (nozzle test piece TK 204, electro-slag weld seam piece TK 50-52, submerged arc weld seam piece TK 51-53) a total program was set up together with all participating institutes. The work had to be done in time because other European industrial nations had to record these test pieces till end of 1978. At first in the IzfP an impulse echo examination was made by RWTÜV and IABG according to the prescribed test procedure and German guide lines.
For some areas the impulse echo testing results were recorded by the IABG automatically. A magnetic particle test was made by RWTÜV to record surface defects. IzfP made US-impulse-echo, tandem and transmission measurements and also investigations with focus probes. All 3 test specimens were investigated with acoustic holo-

graphy and US-high-frequency scattering in order to determine grain size and inhomogenities of structure. For surface inspection there were used eddy-current, potential leakage flux and surface wave techniques.
Also radiographic measurements were made by the MPA/Stgt. The BAM/Berlin performed also US-impulse-echo, tandem and focus techniques, in defect areas there was used the line-holography.

KWU/Erlangen used volumetric automatic inservice inspection methods for all plates - especially in the area of weldseam and handled the results with the TIM-procedure. In addition hand tests were made. After closing of all tests IzfP made evaluation and comparison of the results after receipt of the corresponding reports (see final report RS 247, Nr. 780738-TW , IzfP Saarbrücken, Dr. Deuster, E. Jakobs).

6. Results

The participating institutes used procedures which were search methods for determination of defect orientation and analysis methods for defect interpretation. Because of the many results it should be shown here with the example of TK 50-52 (Fig. 2,4) how good the results agree if every institute uses all possibilities.
We distinguish 3 regions of the test pieces:
- weld seam
- heat affected zone
- base material

Evaluation of TK 50-52 showed many single defects and one continous defect region in the weld seam and the heat affected zone when in the area of y = 0 to 350 mm (Figs. 4, 5, 6, 7, 8, 9). This shows good agreement of the institutes participating referring to continuous defect region. Most times there is no possibility for exact location of single defects because of the great number of single defects. But defects which show quite the same coordinates can be appointed to one another. Combining of single defects means tolerances of ± 15 mm referring to testing procedure and depth of defect.

The surface procedures:

- magnetic particle
- multi-frequency-eddy-current
- surface-wave

show good agreement of results as it is shown for TK 51-53 (Figs. 10, 11, 12). For the cracks found by RWTÜV the IzfP made crack depth-measurements with the potential and magnetic leakage flux-technique and found a max. depth of 3,4 mm.
The US-scattering technique serves for the detection of material anomalies and for grain-size-determination in the regions of weld seam, heat affected zone and base material. The most specimen structures were of very different qualities and inhomogenities were found mainly in the base material in the centre of wall thickness. Details about kind of including and the size of defects are not possible. By grained-size-determination of a two-frequency-procedure there were values of ASTM 6 and 7.

7. Next Steps

The project RS 247 has been concluded on 30.4.78. It is continued by RS 342.

8. Relations to Other Projects

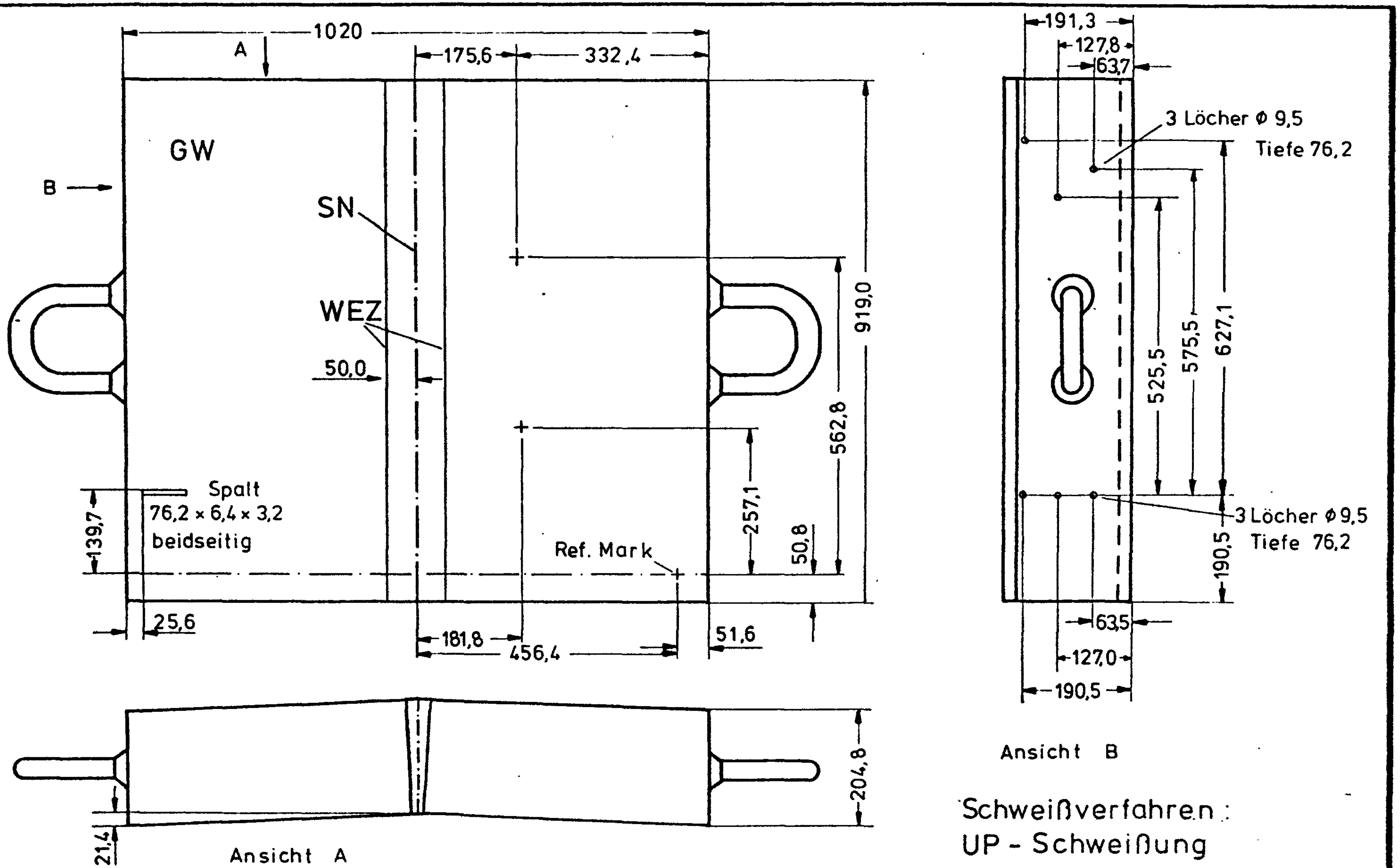
For detection and interpretation of defect regions see RS reports: RS 102-16, 102-17, 102-18, 102-20, RS 27, follow-up project RS 342.

9. References

See final report RS 247 (No. 780738-TW) and reports mentioned in point 4 and 8.

10. Availability of Reports

Through GRS or contractors of the RS-projects.



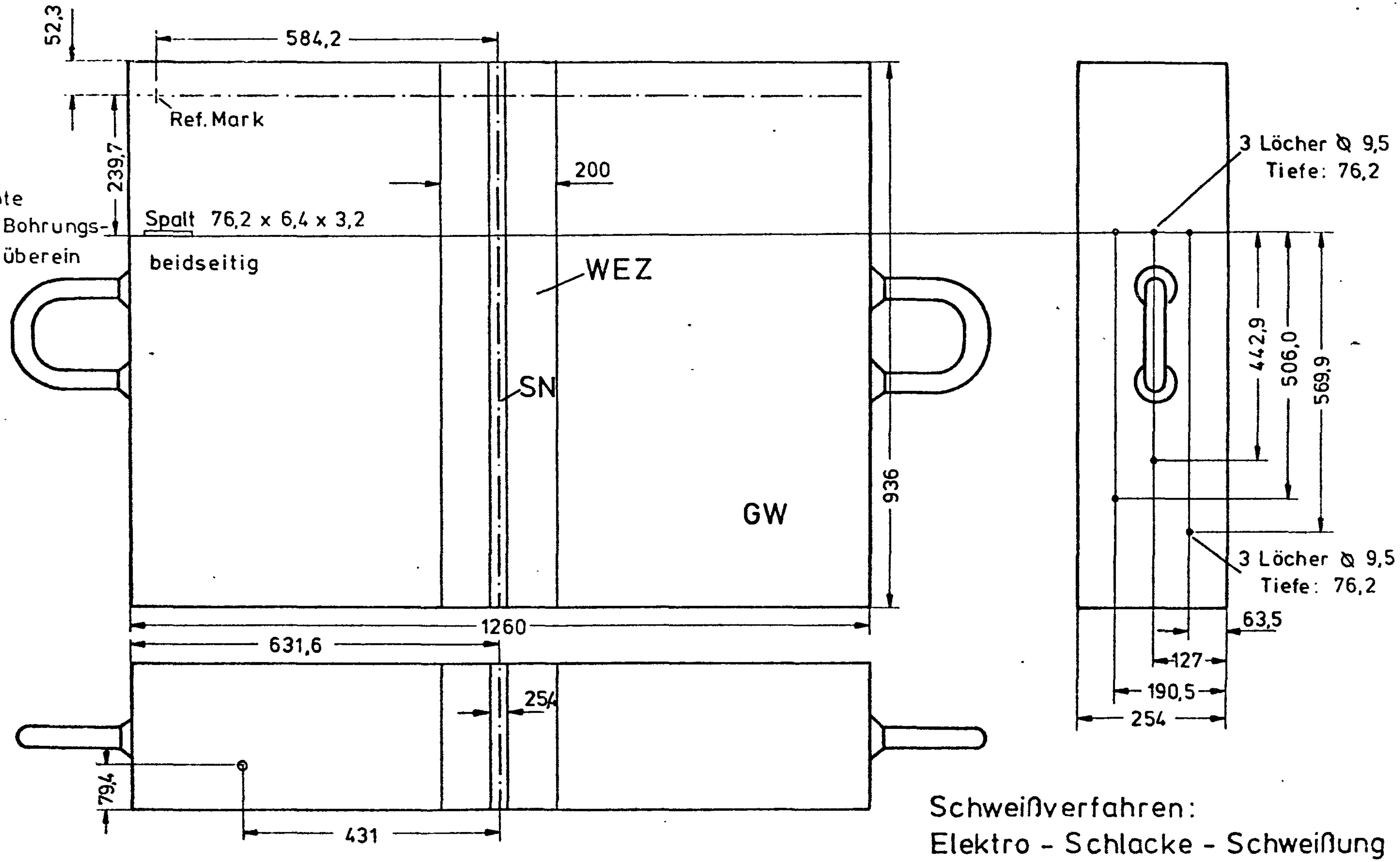
IzfP

TESTKÖRPER 1 (51-53)

Fig. 1

01.01.78 - 31.12.78

RS 247

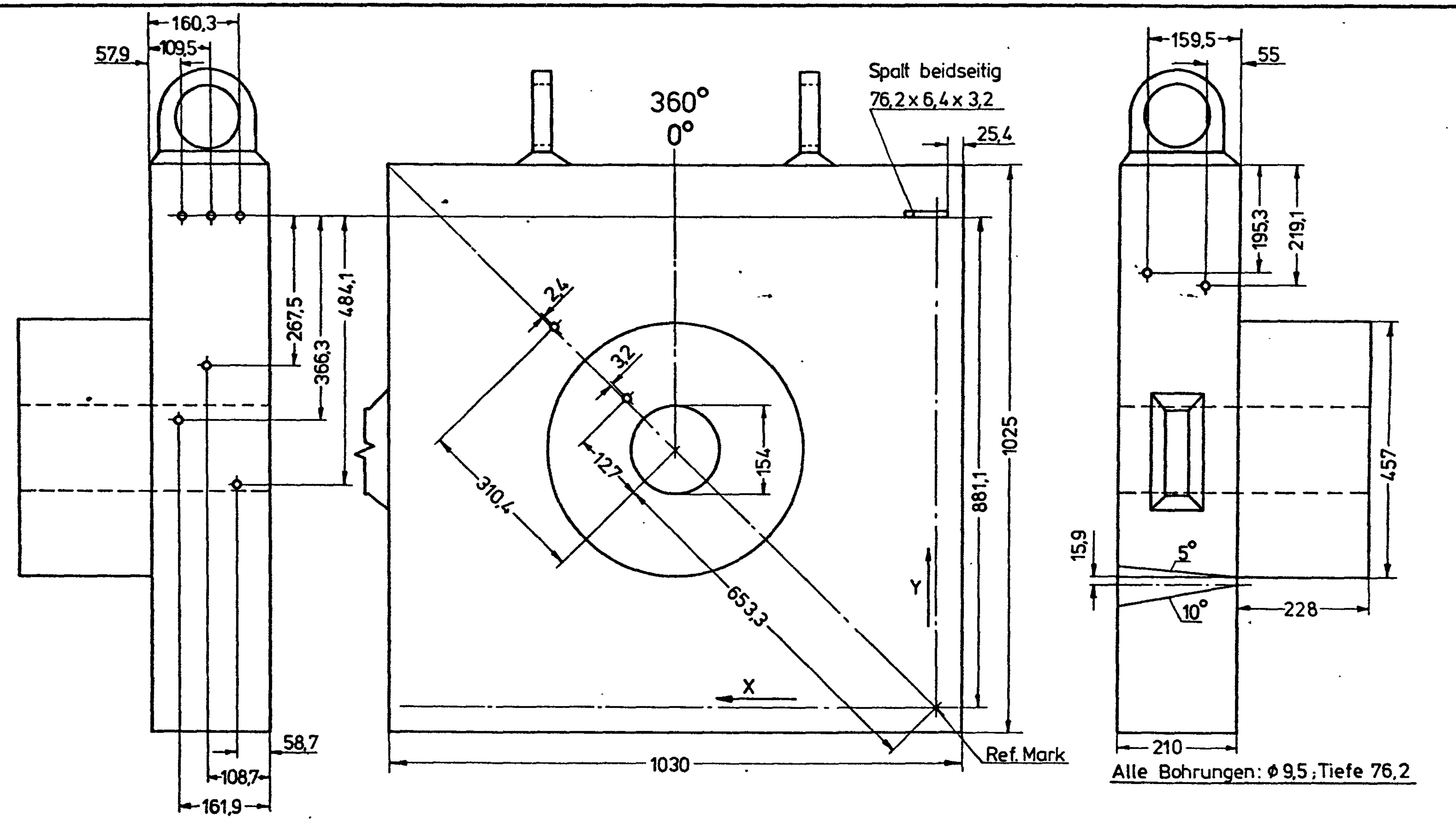


IzfP	TESTKÖRPER 2 (50-52)	Fig. 2

01.01.78 - 31.12.78

-150-

RS 247



Alle Bohrungen: ϕ 9,5 ; Tiefe 76,2

Schweißverfahren: Elektro-Hand-Schweißung

01.01.78 - 31.12.78

- 151 -

IzfP

TESTKÖRPER 204

Fig. 3

RS 247

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78 - 31.12.78	12.3	RS 342

Vorhaben/Project Title	
Zerstörungsfreie Prüfungen an 3 HSST-Platten, Teil II Nondestructive testing of 3 HSST-plates part II	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Fraunhofer-Gesellsch. zur Förderung der angewandten Forschung IzfP-Saarbrücken

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.05.78	31.12.79	Dr. Deuster
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

There were executed nondestructive testings with the 3 HSST-plates on international level. In connection with that there is planned a destructive testing of the found defect regions which renders possible an evaluation of the used procedures.

2. Particular Objectives

2.1. Building of drawing for destructive testing.
2.2. Fixing of regions which shall be tested destructively by the responsible committee PISC and ETF
(PISC - Production Inspection Steering Committee)
(ETF - Evaluation Task Force)
2.3. Comparison of destructive testing with results of nondestructive testing and evaluation of the testing procedures.

3. Research Program

3.1. Dissolving of welding seams together with heat influence surface and base material from the 3 plates (width about 200 mm).
3.2. Additional radiographical and ultrasonic examination of the dissolved welding seam region for control of nondestructive-testing-results.
3.3. Building of transversal and longitudinal grinds within the welding seam region.
3.4. Comparison of destructive and nondestructive-testing results.
3.5. Evaluation of procedures.

4. Experimental Facilities

4.1. Mechanical workshops at Euratom in ISPRA
4.2. Radiography laboratory of ENEL, Laboratorio Centrale DCO Piacenza
4.3. Metallography in the laboratories of MPA/Stgt.

5. Progress to Date

At present Euratom in ISPRA/Italy is responsible for coordination and execution of destructive testing. Also the first cuts were executed here. There was dissolved a region of 100 mm each on both sides of the mid of the two test pieces 50-52 and 51-53 and region of R = 180 to 355 mm of nozzle test piece (204).
Subsequently there were executed radiographical and ultrasonic testings. Thereafter a cross-section (y = 765 mm) with grind was made at a selected region.
Next step was detaching of the both welding seams by transversal cuts (distance: about 50 mm) and of the nozzle welding seam into sections of $\varphi \neq 15^{o}$.
These cuts are executed for about 50 %.
Parallel there will be made destructive testing by MPA on 3 pieces of the welding seams of the pieces 51-53 and 50-52, which shall be finished until March 79.

6. Results

The radiographic examination and ultrasonic-testing with focusing probes up to max. 10 MHz which had been done after dissolving of three welding seams confirmed the detected single defects and defect regions of nondestructive-testing, showed an essential higher defect quantity where defects were found which proved as structure notices.
Before further separating of welding seams a transversal grind (Fig. 1) has been made to a selected point of piece 51-53 which shows the following defects:
1. crack in welding seam up to the surface (depth about 60 mm)
2. voluminuous defect (slag inclusion) of about 5 mm diameter in

grind level of the right weld seam.

3. Defect in the left weld seam (max. extension 1 mm in z-direction) which was not to be registered.

Comparison of the nondestructive-testing results (see final report RS 247) shows good agreement with US-testing. Results of metallographic examinations at MPA/Stgt. are not available at present.

7. Next Steps

It is scheduled to examine all pieces again with focusing probes (f = 10 MHz) and radiographically after closing of cutting in ISPRA. We hope to be able to localize clearly the existing material defects, which render possible thereafter comparison and evaluation of the nondestructive-testing procedure used in the beginning. Parallel there are metallographic testing at MPA/Stgt. on the 3 test pieces from weld seams of test pieces 50-52 and 51-53. The results and comparing evaluation are important for continuing destructive testing on further weld seams which are interesting for the nondestructive-testing.
At the same time there aresome supplement testings in France and Belgium.

8. Relation to Other Projects

Nondestructive-testing of the 3 HSST-plates was made under RS 247

9. References

See report of the project mentioned in point 8.

1o. Availability of Reports

Through GRS resp. realizing institutes of the RS-projects.

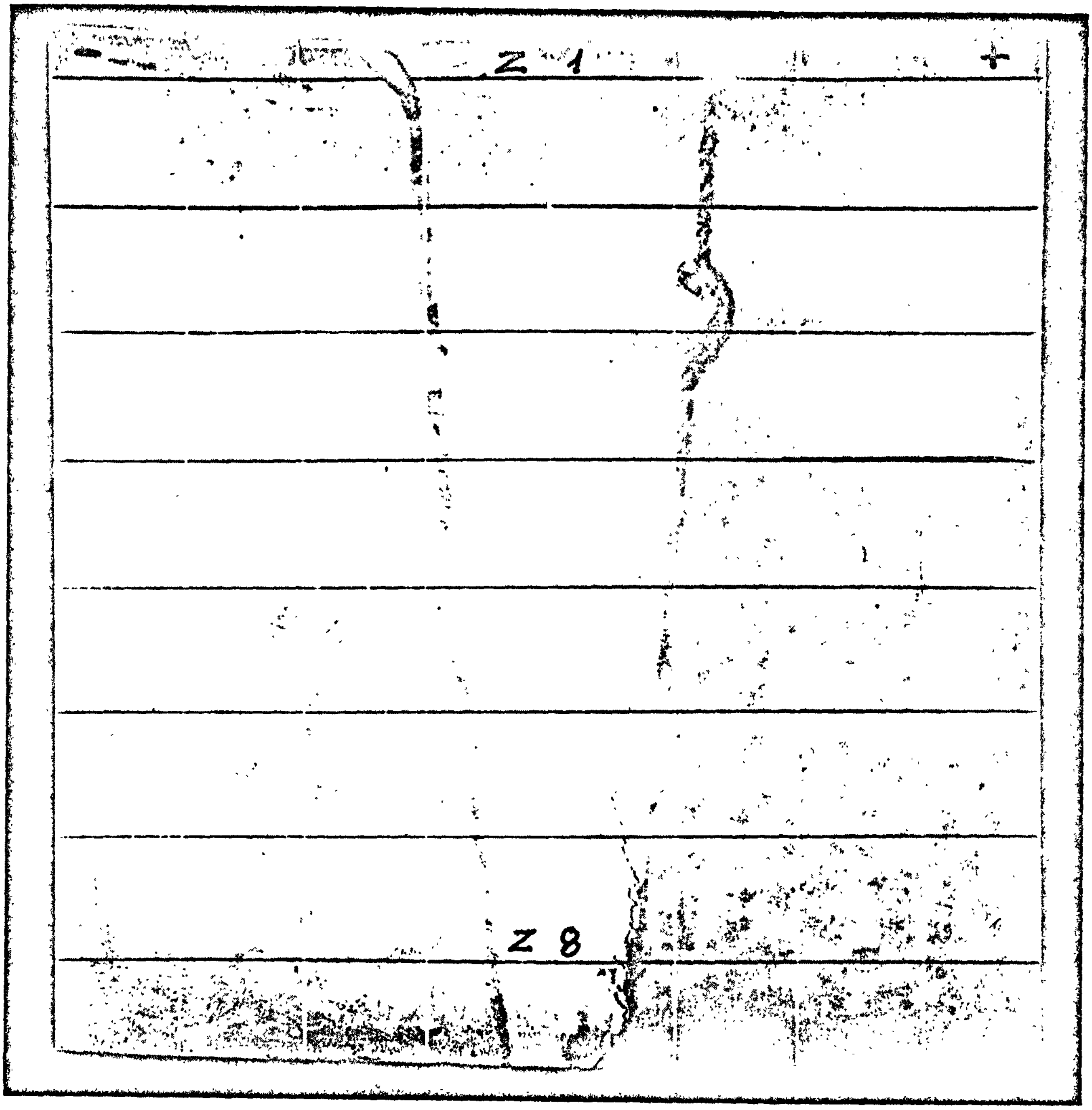


Fig. 1

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78 - 31.12.78	12.3	RS 249/A

Vorhaben/Project Title	Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor
Entwicklung eines elektronisch fokussier- und schwenkbaren Real-Time-Abbildungssystems für die Ultraschall-Werkstoffprüfung Development of an electronically focussed and steered real time imaging system for non-destructive testing	Fraunhofer-Gesellschaft, München IzfP, Saarbrücken

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.01.1978	31.12.1979	Dr. W. Gebhardt
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. GENERAL AIM

   The aim of project RS 249 is to explore the applicability of phased-array techniques in non-destructive testing. Based upon these investigations a real-time imaging system for basic and repetitive inspections of reactor pressure vessels is to be realized.

2. PARTICULAR OBJECTIVES

   While the field characteristics of "conventional" ultrasonic transducers are determined by the geometry of a single transducer, in the case of a phased array a more or less number of transducer elements contribute to the field structure. By variation of the amplitude and phase steering of the transducer elements the ultrasonic field can be influenced definitely: normal and angle probes as well as focussed probes can be substituted by a single array probe.

   From the application of array techniques in non-destructive testing one expects the construction of an inspection system that is applicable both for flaw detection and for flaw analysis. The fast flaw detection could be done with wide opened beam in real-time B-scan, whereas the flaw analysis can be done with fixed and focused beam by A-scan.

3. RESEARCH PROGRAM

3.1 Physical basics

3.2 Measurement and computation of field charakteristics of phased arrays

3.3 Construction of an electronically steered and focussed pulse-echo system

3.4 Form echo elimination by sidelobe suppression

3.5 Array probe optimization

3.6 3-dimensional beamsteering and focussing by use of matrix arrays

3.7 Conception and construction of a smaller, bearable system for hand inspection

3.8 Applications of array systems

4. EXPERIMENTAL FACILITIES, COMPUTER CODES

A first pulse-echo system capable of beam steering, focussing and sidlobe suppression is partly assembled. The time delays of the transmitter signals as well as the receiver signals are realized by CCD-modules. The time delay is determined by a clock generated by a synthesizer module. The system is operating now with eight channels but will be expanted to 24 channels in near future.

The measurements of the sound pressure distribution in the far field and the near field were performed at stepped and at cylindrical steel specimens with electro-magneto-acoustic and piezo-eletric transducers.

A medical phased array system which is modified for non-destructive testing applications can perform electronic beam steering and focussing (real time imaging, sector scanning).

There are computer codes available for

- near field and far field distribution of on array probe
- focussing and beamsteering
- sidelobe suppression
- nonequidstant element configurations
- bulk and shear wave excitation.

5. PROGRESS TO DATE

- further development of the electronically steered and focussed pulse-echo-system,
- theoretical and experimental investigations of the sound field distribution of various array probes (focussing, beam steering, sidelobe suppression)
- bulk and shear wave excitation
- suppression of form echos by amplitude shading
- comparison of electronically steered and amplitude shaded array probes with conventional angle probes
- conception of a smaller, bearable electronically steered pulse-echo system with minor flexibility for hand inspection and for application as stationary probe.

6. RESULTS

Form echos can restrict the inspectability of components. Form echos are produced among others by the sidelobes of normal and angle probes, resp. Suppression of sidelobes and therfore elimination of form echos can be achieved by amplitude shading of an array probe.

To this end theoretical and experimental investigations were made with amplitude shaded probes consisting of 6, 8, 16 elements (aperture 12 mm, 16 mm, 32 mm) and comparisons were made with conventional probes.

The measurement that agree excellent with theory, show that a Dolph-Tchebycheff-weighting brought a mainlobe to sidelobe ratio of up to 60 dB (pulse-echo).

Investigations at a nozzle test piece confirmed, that the amplitude shaded probe is an powerful instrument for formecho suppression. Thurtheron the possibility of sidelobe suppression at oblique incidence (15°-insonification with bulk waves) was investigated. An oblique incidence of the sound beam can be relized by

- the use of an perspex wedge
- electronic beam steering of an array probe.

First, an array probe (2 mm length) was placed upon a perspex wedge and was excited without amplitude shading (this setup is comparable with an conventional angle probe). The far field

characteristic shows a pronounced sidelobe at 10°, which is ca. 11 dB (22 dB pulse-echo) below the main lobe.
To reduce the sidelobe level the probe was shaded by Hamming-weighting. The sidelobe at 10° was reduced, but apart from this the effect of amplitude shading was small.
The reasons for this are the disadvantages of the wedge: unsymmetry because of the variable thickness, multiple reflections.
These disadvantages of oblique incidence can be eliminated, if instead of the perspex wedge the electronic beam steering is used.
Thereby the elements of the array probe are excited with an appropriated time delay. Application of the Dolph-Tchebycheff-weighting now again reduces the sidelobes down to 30 dB (60 dB pulse-echo).

7. NEXT STEPS

- Completion of the pulse-echo system,
- development of various arrays for shear- and bulk waves which are specialized for different inspection modes, array-optimization
- construction of a smaller, bearable electronically steered pulse-echo system with minor flexibilty for hand inspection and for application as stationary probe.
- matrix array.

8. REALATION WITH OTHER PROJECTS

none

9. REFERENCES

A survey of about 126 references is given in a technical report (IzfP, No. 770117). Most of the experiments cited above are discussed in the technical report IzfP, No. 780454.

10. AVAILIABILITY OF THE REPORTS

By GRS, Glockengasse 2, 5000 Köln 1

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	12.3	RS 315

Vorhaben/Project Title	
Real-Time Ultraschall-Holographie Real-Time Ultrasonic Holography	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Inst. f. zerstörungsfreie Prüfverfahren der FhG

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.11.77	31.12.78	Schmitz
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

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.

3. Research Program

3.1 Installation of an acoustical holographic equipment with numerical reconstruction

3.2 Software development for two dimensional numerical reconstruction with C- or 3D-image representation on a graphic display

3.3 Experiments on specimens with artificial or natural flaws and comparison between optical and numerical reconstruction

-2-

4. Experimental facilities, Computer codes

A system with simultaneous scanning of source and receiver is applied. That means, a single probe radiates an ultrasonic longitudinal or shear wave into the material and receives the flaw signal. The probe scanning over the surface measures the phase and amplitude in each position. After mixing the object signal with a reference signal the complex amplitude and phase are stored. A thorough description of the system consisting of holscan 200 and PDP11/34 was given earlier.

5. Progress to date

We habe built up a system with optical and numerical two-dimensional reconstruction which is able to work in the field. Flaw lateral magnification, flaw longitudinal magnification, dependency of frequency, flaw depth, flaw inclination and system parameters have been investigated systematically. The axial resolution is not satisfactory. Therefore we determine the depth by time-of flight measurements with the pulse-echo overlap method. This can be done within one or two minutes. The choice of the probe and the homogeneous beam pattern of such a probe is one of the keys to good imaging. We used focused probes with lenses, focused probes with curved piezoelectric plates in the immersion technique and small probes in the contact technique. For flaws of small depth we prefer probes with planar membranes because of the open directivity pattern and the homogenity beyond the near field length. New probe holding devices have been constructed. They are moved with two cardan shafts along the surface and pressed against it by spring loading.

One of the problems in ultrasonic testing is the deflecting behaviour of large extended flaws. The numerical reconstruction is a well suited tool for quantitative investigation of this phenomena. Experiments in laboratory have been done on horizontal and vertical situated flaws with pulse echo 0^{o}, pulse echo 45^{o} and tandem technique, test specimen with slag inclusions, austenitic weld containing a fatique crack, cladded specimen pipelines, pressure vessel with a nozzle and EPRI-test piece. Experiments in situ: pressure holder Biblis A, acidulous gas vessel.

6. Results

Ultrasonic holography equipment with optical and numerical reconstruction is ready for use in the field. The reconstruction time for one line is done within 10 sec, for a field of 64 x 64 data within 2 minutes and for 256 x 256 within ninety minutes.

7. Next steps

All the experimental work was performed with a slow scanning mechanical system. The use of faster recording devices such as transducer arrays is needed for a more widespread in-service application of ultrasonic holography. This can be done with some tradeoff between multiplexing and parallel processing. In addition the reconstruction time can be diminuished with faster and more parallel working memories.

8. Relations with other projects

None

9. References

V. Schmitz, M. Wosnitza: "Experiences in using ultrasonic holography in laboratory and in the field with optical and numerical reconstruction"
8. int. symp. USA, Miami, 29. May - 2. June 78

10. Degree of availability of the reports

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
Jan. 1 - Dec. 31, 1978	12.3	RS 273

Vorhaben/Project Title	
Erhöhung der Auffindwahrscheinlichkeit von Fehlern durch Verbesserung des Signal-Rausch-Verhältnisses bei der Ultraschallprüfung. Teil II Enhancing the Detection Probability of Flaws by Improving the Signal-to-Noise Ratio During Ultrasonic Testing. Part II	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Battelle-Institut e. V. Frankfurt am Main Kraftwerk Union AG, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
Sept. 1, 1977	Aug. 31, 1979	Dr. R. von Klot
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

The investigations conducted under the above project are aimed at determining the maximum possible signal-to-noise ratio by means of the signal averaging technique during the ultrasonic testing of reactor pressure vessels for flaws in the vicinity of cladding and of austenitic welds. The test speed is to be increased.

2. Particular Objectives

The relatively coarse structure of the austenitic cladding of reactor pressure vessels of light-water reactors and of the austenitic welds in the sodium tank of fast breeders cause strong ultrasonic back-scatter and thus impair the detection of flaws. The signal-to-noise ratio during the search for flaws is to be improved by means of the exponential signal averaging technique.

The signal processing speed of the signal averager is to be increased such that the speed of probe advance can be raised to about 5o mm/s. This must, however, also permit real-time processing of the echo signals at sufficiently narrow pulse spacing, which requires a high repetition frequency.

The maximum possible signal-to-noise ratio is to be determined by variation of the following parameters:

- Probe type (angle, focusing, wide-band and transmitter-receiver probes)
- Test technique (single-probe and tandem technique, inside and outside testing)

- Speed of probe advance
- Signal preprocessing (rectifying, taking the logarithm)
- Weighting factor of the signal averager

3. Research Program

3.1. Purchase and modification of the signal averager
3.2. Extension of the ultrasonic flaw tester
3.3. Laboratory experiments on a clad test specimen with artificial flaws
3.4. Experiments on a test specimen taken from the wall of a pressure vessel at KWU, Erlangen
3.5. Purchase of special probes for testing a specimen with austenitic weld
3.6. Laboratory experiments on specimen with austenitic weld

4. Experimental Facilities, Computer Codes

Re 3.1. Increase in the signal processing speed of the signal averager from 5o to 35o signals per second

Re 3.4. Purchase of probes for testing reactor pressure vessels in the vicinity of claddings.
Flaws in the vicinity of the cladding of a test specimen taken from the wall of a pressure vessel:
(a) At the 6^{o} side wall of the circumferential weld
- No. 2, width 33 mm, depth 3.5 mm
- No. 1o, width 29.1 mm, depth 7.2 mm

(b) At the 1o^{o} side wall of the longitudinal weld
- No. 1o5, width 15.6 mm, depth 7.8 mm
- No. 1o9, width 29.2 mm, depth 14.7 mm

Re 3.5. Fixing the specifications for the probes for testing austenitic welds

5. Progress to Date

Re 3.4. Performance of automatic ultrasonic tests on the test specimen taken from the pressure vessel wall (2 test runs). Recording of the echo ampliutde of the artificial flaws in the vicinity of the cladding and of the scattered reflections from the coarse-grained structure of the austenitic cladding, as a function of the probe path, with variation of the following parameters:

- Test techniques: outside testing by means of the single-probe technique, inside testing by means of the transmitter-receiver technique, inside testing by means of the tandem technique
- Speed of probe advance
- Direction of probe advance relative to the plane of incidence
- Signal repetition frequency and pulse spacing
- Signal preprocessing: HF signal, rectified signal, rectified signal on a logarithmic scale
- Weighting factor

Determination of the signal-to-noise ratio and the half widths from the measured data.

6. Results

Re 3.4. The signal-to-noise ratio as a function of the weighting factor in general has a maximum (Fig. 1). With increasing pulse spacing a , this maximum is shifted towards smaller weighting factors. The following maximum improvement of the signal-to-noise ratio results for the three test techniques:

- Outside testing by means of the single-probe technique: 2.9 dB
- Inside testing by means of the transmitter-receiver technique: 8.6 dB
- Inside testing by means of the tandem technique: 9.1 dB

With decreasing pulse spacing, the maximum improvement in the signal-to-noise ratio in general increases (in the case of all three testing techniques). If the direction of probe advance is normal to the plane of incidence a larger improvement in the signal-to-noise ratio is achieved than if the direction of probe advance is in the plane of incidence (in the case of all three testing techniques).

Averaging over the rectified signal results in a larger improvement in the signal-to-noise ratio (outside testing be means of the single-probe technique) or in approximately the same improvement (inside testing by means of the transmitter - receiver technique) as averaging over the HF signal. In the case of averaging over the rectified signal on a logarithmic scale, the maximum improvement in the signal-to-noise ratio is found to be much smaller (inside testing by means of the transmitter-receiver technique). With increasing transducer dimensions, i. e. decreasing spread angle, the maximum improvement in the signal-to-noise ratio increases substantially (inside testing by means of the transmitter-receiver technique).

The signal-to-noise ratio was evaluated as a function of the weighting factor G for flaws of different dimensions. Flaw No. 1o5 is about half as wide and half as deep as flaw No. 1o9. Without the signal averaging technique, the smaller flaw is indicated with a signal-to-noise ratio which is about 2 dB lower than in the case of the larger flaw (cf. Figs 2a and 2b for $G = 2^{o}$). The signal averaging technique increases the signal-to-noise ratio in the case of the larger flaw by a maximum of 6 dB (Fig. 2a) and in the case of the smaller flaw by a maximum of 11 dB (Fig. 2b).

Since the 6-dB half width of the echo dynamic curves (echo amplitude as a function of the probe path) is necessary to describe and evaluate flaws, it was investigated whether and to what extent the half width is affected by the exponential signal averaging technique. The 6-dB half width is constant for small weighting factors G and increases steeply at large values of G (Fig. 3). For the optimum weighting factor where the signal-to-noise ratio has a maximum (cf. Fig. 1) the 6-dB half width has already increased by the following factors:

- 1.3 for outside testing by means of the single-probe technique (Fig. 3a)
- 1.4. for inside testing by means of the transmitter-receiver technique (Fig. 3b)

- 1.7 for inside testing by means of the tandem technique (Fig. 3c)

If the tests are made with an optimum weighting factor, these factors have to be taken into account when evaluating the half widths.

7. Next Steps

Re 3.4. Performance of automatic ultrasonic tests on test specimen from the pressure vessel wall (3rd test run); outside testing with focusing probes.

Re 3.6. Performance of automatic ultrasonic tests on a specimen with austenitic weld using the following probes:
- Wide-band angle probes for longitudinal and transverse waves
- Focusing probes for longitudinal waves
- Wide-band transmitter-receiver probes for longitudinal waves

8. Relation to Other Projects

RS 19o Enhancing the Detection Probability of Flaws by Improving the Signal-To-Noise Ratio During Ultrasonic Testing
Battelle-Institut e. V., Frankfurt; January 1976 to August 1976

RS 27o3 Onward Development of Nondestructive Testing Methods for In-Service Inspection of Reactor Power Plants
BAM/Berlin, Krautkrämer/Köln, KWU/Erlangen, M.A.N./Nürnberg; February 1976 to December 1978

RS 244 Ultrasonic Testing Techniques for Pre- and In-Service Inspections for Fast Breeder Reactors
BAM/Berlin; January 1977 to December 1979

RS1o2-16/1 Non-Destructive Structure Evaluation by Means of Scattered Ultrasound
IzfP/Saarbrücken, May 1973 to December 1977

9. References

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1o. Degree of Availability of the Reports

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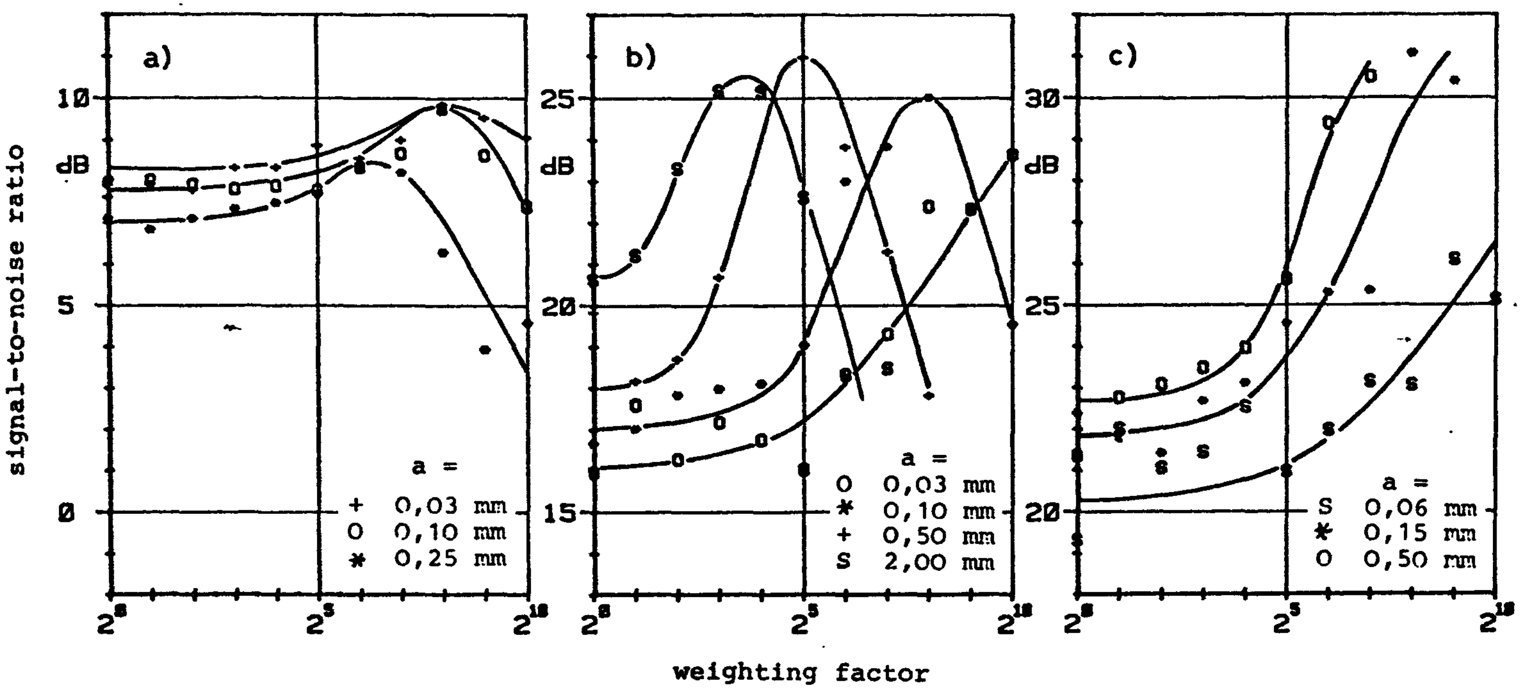

Fig. 1: Signal-to-noise ratio as a function of the weighting factor. Examination of flaw No. 1o in the vicinity of the cladding of the test specimen from the pressure vessel wall. Direction of probe advance normal to the plane of incidence; a = pulse spacing

(a) Outside testing by means of the single-probe technique, averaging after rectification

(b) Inside testing by means of the transmitter-receiver technique, averaging over the HF signal

(c) Inside testing by means of the tandem technique, averaging after rectification

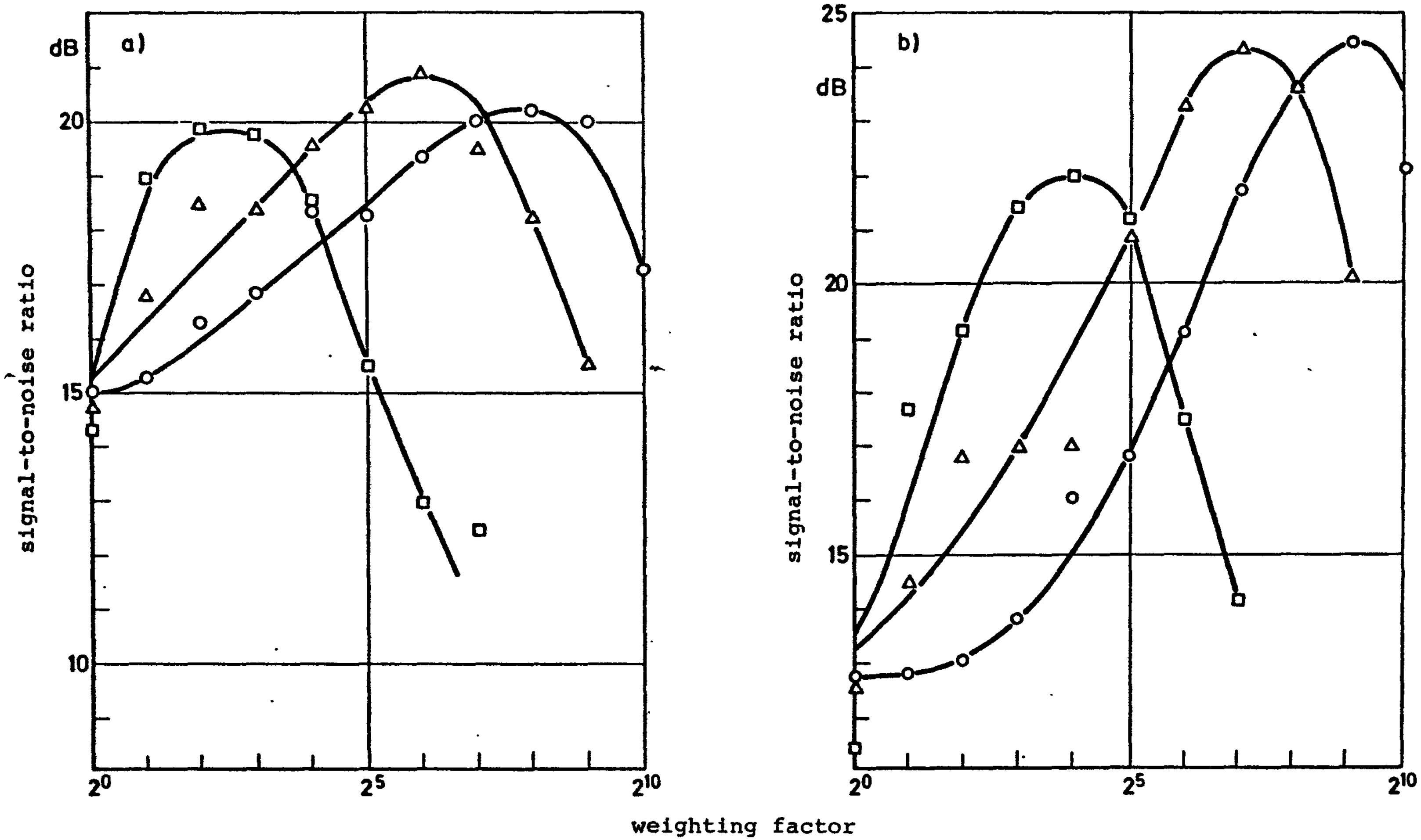

Fig. 2: Signal-to-noise ratio as a function of the weighting factor. Examination of flaws in the vicinity of the cladding of the test specimen from the pressure vessel wall. Inside testing by means of the transmitter-receiver technique, direction of probe advance normal to the plane of incidence, averaging after rectification.

Pulse spacing a = o.o3 mm (o); o.1o mm (Δ); o.5o mm (□).

(a) Flaw No. 1o9 (b) Flaw No. 1o5

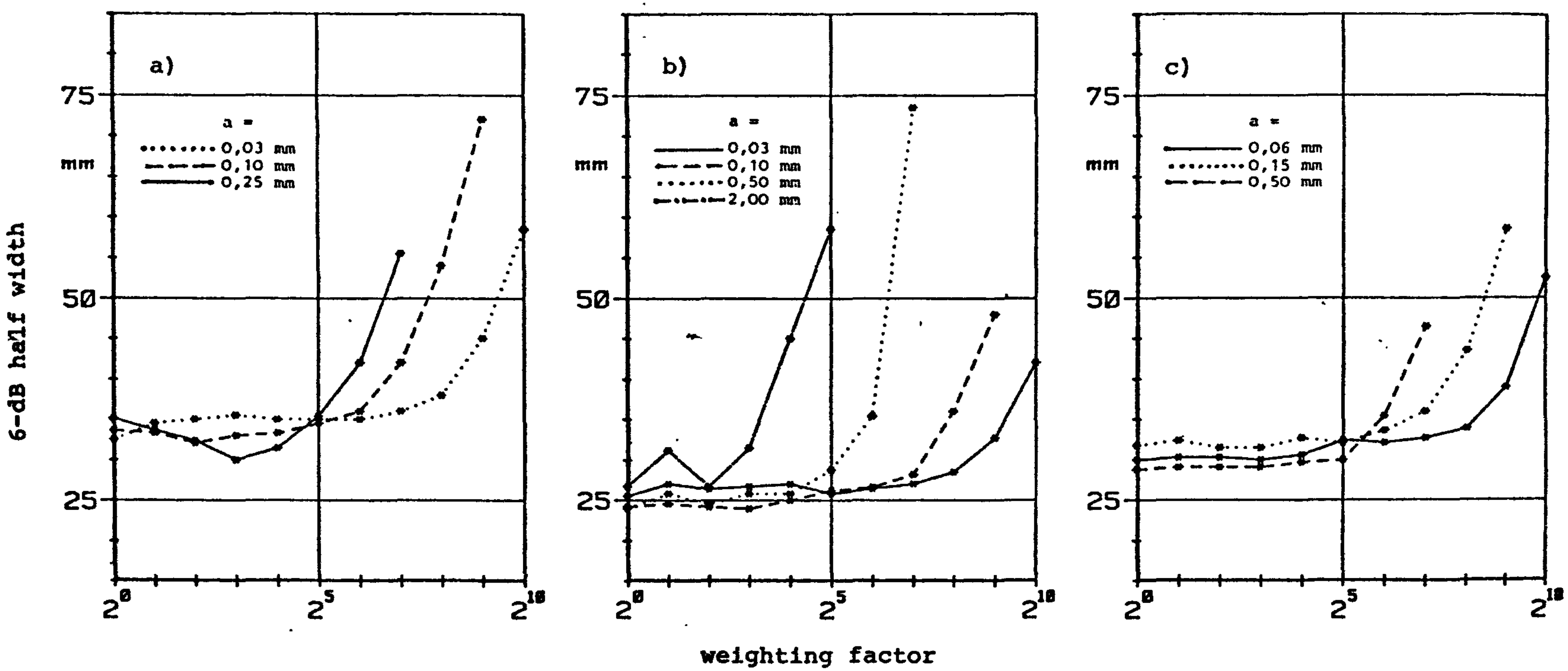

Fig. 3: 6-dB half width of the echo dynamic curves as a function of the weighting factor. Examination of flaw No. 1o in the vicinity of the cladding of the test specimen from the pressure vessel wall. Direction of advance normal to the plane of incidence; a = pulse spacing

(a) Outside testing by means of the single-probe technique, averaging after rectification

(b) Inside testing by means of the transmitter-receiver technique, averaging over the HF signal

(c) Inside testing by means of the tandem technique, averaging after rectification

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.7.78 - 31.12.78	12.3	RS 330

Vorhaben/Project Title	
Methoden zur Messung von Geräteparametern an Ultraschallprüfsystemen für Kernkraftwerkskomponenten Methods for the measurement of characteristic datas from ultrasonic inspection systems for nuclear components	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Bundesanstalt für Materialprüfung (BAM) Labor 6.21, Berlin

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.7.1978	31.12.1980	Dr. Wüstenberg
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	31.12.1978	602.250,-- DM

1. General Aim

Improvement of the reliability and reproduciability of the ultrasonic inspection.

2. Particular Objectives

Construction of a movable laboratory for the examination of equipment- and sound field-parameters. The aim is the automatic and objective recording of all received measurement datas.

3. Research Program

3.1 Procurement and Accommodation of Equipment

Selection of marketed equipments for the registration of important characteristic datas, interfaces for the accommodation of the datas; specially to be regarded: simple operation, serviceability, temperature drift, influences of supply voltages.

3.2 Construction and first experiments of equipment units and testblocks

Determination of kind and number of testblocks needed for the measurements of sound field-parameters and the echodynamic behaviour.

3.3 Combination of the movable laboratory

Choice of a lorry or a container regarding especially the kind of wheel suspension.

4. Experimental Facilities, Computer Codes

5. Progress to Date

to 3.1 Construction of the directivity pattern measurement unit, decision of manipulating possibilities for probes and contactless electrodynamic microphones. Characterization and improvement of the electrodynamic point like microphone.

Choice and procurement of a frequency analyzer and a quick stepless HF-gate for the frequency analysing measurement units.

to 3.2 Conception of a quick and exact peak detector.

to 3.3 Negotiation for the procurement of a lorry with pneumatic suspension of the rear wheels and a small container conception of the space division.

6. Results

to 3.1 Works to the construction of the directivity measurement unit are not yet finished. Only to the characterization of the contactless microphones some concluding results are available.

to 3.2 Works are not yet finished.

to 3.3 Offers and orders concerning the procurement of a lorry are on the way.

7. Next Steps

to 3.1- 3.3 Construction of the directivity pattern measurement unit, of echodynamic measurement and recording device of sound field mapping system (scanning cross section of the sound field with an electrodynamic microphone). Test of the frequency analysing device, amplifier characteristics, longitudinal resolution.
Use of a microprocessor for data acquisition and a mini-

processor for present the datas.

8. Relation with other Projects

RS 27; RS 169

9. References

H. Wüstenberg
Bestimmung der Richtcharakteristik von Winkelprüfköpfen für die Ultraschallprüfung am Kontrollkörper DIN 54120
Materialprüfung 11 (1969) Nr. 9 S. 311 - 315

H. Wüstenberg
Untersuchungen zum Schallfeld von Winkelprüfköpfen für die Materialprüfung mit Ultraschall
Dissertation D83 TU-Berlin 1972

M. J. Whittle; H. Smallman
The Assessment and Specification of Ultrasonic Probes
Report; ISPRA, Italy, June 1 - 3, 1977

H. Wüstenberg
Characteristical sound field data of angle probes
VII International conference on NDT; 4. - 8. Juni 1973, Warschau

E. Mundry; H. Wüstenberg
Untersuchungen zur Richtcharakteristik von Winkelprüfköpfen
Vortrag; DGZfP-Tagung in Mainz, 11. - 12. Mai 1977

J. A. Baily
Ultrasonic Pulse-Echo Instrument Linearity Measurement
Materials Evaluation; May 1977, Vol. 35 Nr. 5 S. 64/68

DGZfP-Richtlinie
Eigenschaften von Prüfeinrichtungen mit Ultraschall-Impuls-Echo-Geräten und ihre Kontrolle
September 1971

H. Wüstenberg; E. Schulz; W. Möhrle; J. Kutzner
Zur Auswahl der Membranformen bei Winkelprüfköpfen für die Ultraschallprüfung
Materialprüfung 18 (1976) Nr. 7, S. 223 - 230

Bericht der Fachgruppe 6.2 "Zerstörungsfreie Materialprüfung"
Laboratorium 6.21 "Mechanische und thermische Prüfverfahren"
Untersuchungen zum Einfluß der Plattierung und der Geometrie auf die Prüfbarkeit von dickwandigen Reaktorkomponenten mit Ultraschall bei der Tandem- und der Einkopftechnik
Oktober 1972; BAM-Berlin

10. Degree of Availability of the Reports
GRS, Köln

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	12.3	RS 299

Vorhaben/Project Title	
Untersuchungen zur Anwendung des Impuls-Wirbelstromverfahrens für die zerstörungsfreie Prüfung von kerntechnischen Komponenten Investigations on the application of the pulsed eddy current method to nondestructive testing of nuclear power components	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Bundesanstalt für Materialprüfung, Berlin (BAM)

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.1978	31.12.1979	Dr.-Ing. G. Wittig
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

Starting from the present problems in nondestructive testing of austenitic surfaces and components within nuclear power plants fundamental investigations are to fulfil with the general aim, to make the pulsed eddy current method available for these applications. The inspection of thin-walled irradiated and non irradiated cladding tubes for fuel elements practised in USA by this method is to extend to thick-walled components or thicker layers.

2. Particular Objectives

Development of an equipment for investigations on calibration standards and specific test specimens. This is an independent instrument which contains all the required electronic circuits and signal outputs for signal evaluation and recording.

Development of coil systems adapted to present test problems. The particular consideration is directed to a high local defect resolution with a sufficient penetration depths.

Determination of the conditions for an optimal signal evaluation. Connected with these problems the influence of noise parameters is to hold of and provisions for their suppression are to develop. Statements about the efficiency of the pulsed eddy current method are to deduce.

3. <u>Research Programm</u>

3.1 Design and construction of experimental apparates
- pulsed eddy current equipment
- measuring and control assemblies
- mechanized scanning system
- connection of the test equipment to a system for digital signal processing.

3.2 Development of coil systems.

3.3 Investigations about the signal evaluation for applications to calibration standards with defects and specific test specimens.

3.4 Constructing of a pilot model.

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

3.1 The pulsed eddy current test equipment contains the following assemblies: Function control generator, power stage to deliver a current pulse into the field coil, signal pre- and main-amplifier, four parallel channels for sampling the signal at adjustable delay times.

Measuring amplifier with probes to survey the pulsed magnetic field of coils.

Interface assembly for the connection of the test equipment to a digital signal processing system.

3.2 Design of software for signal evaluation.

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

3.1 Design, construction and trial of the pulsed eddy current test equipment.

Development of the assembly to survey the pulsed magnetic field of coil systems.

Design of interface circuits for connection of the test equipment to a data processing system.

3.2 Design and manufacturing of probe coil systems with screenings to shape the generating fields. Measurement of the pulsed field distribution.

6. Results

3.1 Specifications of the pulsed eddy current test equipment:
- half-sinusoidal pulse current through the field coil with adjustable amplitude up to 6 A and pulse width 5 to 50 μs
- signal gain 30 dB
- four adjustable sampling points, delay times 1 to 200 μs
- four analog signal outputs corresponding to the sampled amplitude of the measuring coil voltage.

7. Next steps

Further work on the development of interface circuits and of coil systems.

Investigations about the signal evaluation for applications to calibration standards with defects and specific test specimen.

Constructing of a pilot model.

8. Relation with other projects

RS 89 and RS 229
Cooperation within the working group LA RS 255ff for the projects RS 255, RS 256, RS 257, RS 258.

9. References

Final Report RS 229: Study on the application of pulsed eddy current methods to quality control of nuclear power plant components.

10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	12.3	RS 255

Vorhaben/Project Title	
Mehrfrequenz-Wirbelstromprüfung Multi-Frequency Eddy Current Testing	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 214, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 3. 77	31. 3. 80	H. Jacob
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 78	

1. General Aim

To design, construct and test a multi-frequency eddy current test unit (prototype) based on the currently available test results. This test unit is to meet the requirements of the automatic in-service inspection test of the pressure-containing reactor component walls (plated, unplated). For this reason it must produce an on-line record while being used in the pertinent control area.
It is the aim of the venture for the unit to be as broadly applicable as possible; it is to be universally used for both LWR types, i.e. BWR and PWR as well as for LMFBR.

2. Particular Objectives

To extend the test findings for S.G. tube testing, in addition to the free tube area, especially for the rolled-in sections of the tubessheet, spacers and flow distribution sheet as well as the necking down on both sides of the tube bend.
Test of ferritic tubes used in the intermediate heat exchangers of Na-cooled reactors.
Comparison of results obtained from external tests of straight tubes and internal tests of U-tubes in order to find a relationship between manufacturing test and in-service test.

Test of the multi-frequency method as applied to surface testing in order to improve failure detection by better sub-surface resolution.
To design, construct and test a data acquisition system for eddy current testing for master data storage, on-line evaluation and test data recording.

3. Research Program

3.1 Coordination of developmental work

3.2 Tube test
- to erect a S.G. model
- To apply multi-frequency eddy current method to the model
- to improve current manipulator system
- to apply method to PWR S.G.
- System analysis

3.3 Surface testing
- prepare basic test data
- optimization of characteristic data for test scans
- applicability of current manipulating systems
- multi-frequency eddy current test method applied for HDR

3.4 Data acquisition
Acquisition, combining and recording of test results.

4. Test Facilities

Measurement equipment, provided by KWU, will be used. This operates on 4 frequencies from 1 kHz to 1 MHz, which will be simultaneously applied to the test coil.

The preparation of the pertinent algorithms as well as the required soft ware package (master data storage, on-line evaluation, recording) as well as specification, concept and technical arrangements of the data processing plant will be performed under the scope of this task.

5. Progress to Date

To 3.1
- Bilateral discussions with partners
- Conduct of 5th Steering Committee Meeting
- Coordination talks to extend the scope of project RS 256 (Interatom) with partners
- Conduct of the 6th Steering Committee Meeting
- Assembly of eddy current test blocks.

To 3.2
- Examination of the influence of the coil diameter regarding the "filling level" in the U-tube s.G. test
- Comparision measurements with commercial multi-frequency unit (procedure qualification tests under laboratory conditions) and with eddy current test techniques employed by KWU and of the single frequency method with automatic evaluation.

To 3.3
- Manufacture of test blocks for covered cracks
- Fabrication of test probes for basic tests on ferromagnetic materials
- Preparation of test blocks for basic investigations on ferromagnetic materials
- Basic investigations on ferromagnetic materials
- Trial run of testing equipment on 2 threaded holes of a PWR pressure vessel (original conditions).

To 3.4
- Provision and discussion of a test flaw catalog for the eddy current examination set-up
  Preparation of detail drawings for locating test flaws
- The specification for the multi-frequency eddy current examination system has been supplemented regarding revision "a", specification of the interface between the eddy current unit and the data processing system.
- Participation in a demonstration of the single-frequency eddy current examination unit of Messrs. Förster at KWU Erlangen
- A program description has been completed. A start was made with the evaluation of signals recorded on magnetic tape
- Analysis of computing program for processing eddy current signals in laboratory tests. Draft of a concept for modifying and extending the program with a view to

applicability for in plant eddy current examinations. Preparation and encoding of a multi-task FORTRAN program

- The possibility of using averaging units and Fourier-type analyzers was examined at the same time to reduce the time needed for processing and classifying signals. An FFT processor was available on loan for a short period of time. Original data were analyzed.
- One original SG tube with an overall length of approx. 7300 mm was provided with a total of 20 different test defects both in its straight and bend sections. It is thus available for computing program verification and the development and testing of the planned interface.
- Work was continued for improving the signal noise ratio in the evaluation of magnetic tapes. At present, tapes with test signals are being evaluated.

6. Results

To 3.1 The Steering Committee agreed in principle to Interatom's extension of the project scope. The work should also encompass the applicability of alternative methods for Na-wetted tubes e.g., the magnetic leakage flux examination.

To 3.2 The S.G. U-tube examination should be carried out as close to a water level of $\eta > 0.80$ as possible, as otherwise the detection threshold of materials separation with flaw depths $\leq$ 20 % can be met only to a limited extent. When the multi-frequency units made available by IFR ("Institute for Nuclear Engineering") and IzfP ("Institute for Non-destructive Materials Examination") were compared with the single-frequency test method used by KWU, no improvement was found as regards the resolution and detection of flaws.

To 3.3 Flaws originating on the contact surface can be tested with an interpretability comparable to that with nonferromagnetic materials.

The fundamental investigations of ferromagnetic materials showed that the threads on the RPV can be examined, for material separation using the eddy current method and that a statement on the depth of such flaws is possible. In principle, the thread bottoms of blind holes in the RPV can be examined. The changes in noise level compared with the test block are negligible. A continuous test rate must be ensured.

To 3.4 The following correlation independent of material constants and test frequency of the eddy current examination unit was confirmed for the time response of the complex signal between the amplitude maximum values:

$$t_{pp} = \frac{\max\{l_F, l_s\}}{v} = \frac{N}{B}$$

t_{pp}	= duration of signal (peak to peak)	(mm)
l_F	= length of flaw in axial direction	(mm)
l_s	= distance between coils of probe	(mm)
v	= rate of probe travel	(mm s^{-1})
B	= scanning rate (constant)	(s^{-1})
N	= number of scans between peaks	

The requirement of the eddy current units to be compatible with present evaluating units (US data acquisition systems) is deemed to be fulfilled by supplementing the specification for the multi-frequency eddy current examination unit by the interface between the eddy current unit and the data processing system.

7. Next Steps

To 3.1 Preparation and conduct of the next Steering Committee Meeting with a critical assessment of the stage reached in the completion of the overall project.

To 3.2 - commissioning of the new KWU multi-frequency eddy-current unit
- comparison measurements.

To 3.3 - Completion of the test blocks and first trial measurements

To 3.4 Program parts shall be rewritten and tested in assembler language to increase the processing speed in the selection and filing of eddy current data on magnetic disks.
The evaluation algorithm shall be modified to the effect that influences from disturbances on original eddy current data no longer affect classification and interpretation. Furthermore it is planned to design the interface to be prepared so that the noise mentioned of programmable electronic modules is already suppressed.

8. Relation with Other Projects

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9. References

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78 - 31.12.78	12.3	RS 257

Vorhaben/Project Title	
Anwendungstechnische Versuche zur automatischen Prüfung der Einbauten von Reaktorkomponenten und deren Wandungen mit Mehrfrequenzwirbelstromverfahren mit ON-LINE-Dokumentation Investigations on the application of the multifrequency eddy current method on the automatic testing of installed reactor components and their walls with on line document.	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Fraunhofer-Gesellsch., München IzfP, Saarbrücken

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.03.1977	29.02.1980	Dipl.-Phys. R. Becker
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

The aim of the project is the application of the results obtained from the preceeding projects (RS 231, RS 102-18) for the development and improvement of a multifrequency eddy current test equipment (prototype). This test device has to meet the requirements of the automatic recurrent testing of the installed components of reactors and their walls. For this purpose an on-line documentation as well as the employment in a control range is necessary. The application field of the test device is aspired to be as large as possible.

2. Particular Objectives

Realization of the test method, development of the needed software, adaptation of the test device to the different test situations, improvement of the test method at models and real reactor components.

3. Research Program

3.1 Optimization of the multifrequency prototype for the testing of steam generator tubes with 3 frequencies.

3.2 Improvement of the prototype hardware concerning an automatic calibration process.

3.3 Interpretation of the defect signals obtained after the suppression of the disturbing signals,

a) in the case of pick up coils

b) in the case of coaxial inner coils

4. Experimental Facilities, Computer Codes

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4.1 A multifrequency test equipment built up in the scope of RS 231.

4.2 A program computing the coil impedance as a function of the frequency, the material characteristics and the coil dimensions.

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

Relation to 3.1:

The following disturbing situation was considered: tube sheet tube support plates, variations of the electrical conductivity, of the inner and the outer diameter of the tubes. Three test frequencies must be applied to suppress the corresponding disturbing signals. First these frequencies can be choosen arbitrarily out of a given series of 9 frequencies. Eighty-four combinations are possible at all.

By means of a computer program a frequency combination was selected which gives the optimal defect sensitivity after the suppression of the disturbing signals.

Relation to 3.2:

A μ-processor takes the organisation of the calibration step and computes and tunes constants. During the testing process the actual measuring values are multiplied with these constants, so that the signals caused by the disturbing parameters are suppressed and only the defect signals are indicated. The corresponding software concept was developped, as well as the hardware interface between the processor and the eddy current prototype was realized.

Relation to 3.3:

After the multifrequency algorithm is applied the defect signals usually are obtained in a scalar form. This proceeding is not able to identify uniquely the type and the size of an indicated defect. A method which solves this task was developped.

By means of adding at least one measuring value (this can be the real part or the imaginary part of the coil impedance at a new frequency) it is possible to apply the multifrequency algorithm two times with the condition that in the two cases the same disturbing parameters are suppressed. The two read-out values are given on the horizontal and the vertical deflection of an oszilloscope. By this procedure one obtains a two-dimensional description of the defect signals, so that one has a higher information content to interpret the defect signal.

6. Results

Relation to 3.1:

After the simultaneous suppression of the above named disturbing signals, the signals of saw cuts at the outside of the tubes, axially orientated and with a depth of 30 % of the wall thickness could still be detected.

Relation to 3.2:

The automatic calibration procedure yields an improvement in reducing the remaining disturbing underground because of a more exact measuring and handling of the disturbing parameters especially in the case of small signals.

Relation to 3.3:

The two-dimensional description of the defect signals after the suppression of the disturbing signals gives a phase splitting corresponding to the defect types, the indication peak is correlated to the defect depth. The following defect types can be discerned;

a) pick up coil for testing plates, welded joints and plated surfaces: different depths of surface- and subsurface defects.
b) coaxial inner coils for testing heat exchanger tubes: slits, reduction of wall thickness and holes at the inner and outer side of the tubes.

7. Next Steps

7.1 Continuation of the interpretation model for the defect signals, consideration of other defect types.
7.2 Construction of a heat exchanger model, proving and comparison of several multifrequency systems.

8. Relation With Other Projects

RS 231, 255, 256, 258.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78 - 31.12.78	12.3	RS 258
Vorhaben/Project Title Experiments regarding automatic inspection of the built-in parts of reactor components and their walls by means of multi-frequency eddy-current method with on-line documentation for repetitive inspection Anwendungstechnische Versuche zur automatischen Prüfung der Einbauten von Reaktorkomponenten und deren Wandung mit Mehrfrequenz-Wirbelstromverfahren mit ON-LINE-Dokumentation für die wiederkehrende Prüfung		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Institut Dr. Förster
Arbeitsbeginn/Initiated 01.03.77	Arbeitsende/Completed 28.02.80	Leiter des Vorhabens/Project Leader Dr. W. Stumm
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

It is aimed at the application of the knowledge gained in previous projects for design, construction and testing of an eddy-current test installation using the multi-frequency method, considering available test experience. This test installation shall meet the requirements of the repetitive inspection of the built-in parts of reactor components and their walls (plated and unplated). For this purpose is necessary an on-line documentation as well as an in-situ operation of the test installation. A maximum range of application of the test installation is aimed at; it ought to be universally applicable for light water reactors of the boiling water and pressurized water design as well as for sodium-cooled fast breeders.

2. Particular Objectives

Inspection of austenitic plated surfaces on cracks originating from the surface as well as from the joint face to the basic material. Application of the multi-frequency method for improvement of the signal-to-noise ratio by a - compared to the single-frequency method - better suppression of characteristic disturbing influences as distance variations of the probe and variations in ferrite content. Possibility to classify depths of defects by means of eddy-current signal.
Application of multi-frequency method for inspection of steam generator tubes over the whole tube range, especially at the rollings at the bottom of the steam

generator tube, the support grids, the flow distributor plate as well as the whole tube bend.

3. Research Program

3.1 Conclusion of the study about the applicability of measuring the probe spacing by an electrostatic method.

3.2 Design, building up and testing of the transmission line between probe and test instrument.

3.3 Theoretical study about the influence of the transmission line on the signal character.

3.4 Beginning of the development of a monitor facility for the whole test unit, including the probe and the transmission line.

3.5 Design and testing of different coil arrangements for the inner diameter of S.G. tubes. Study of the possibility to suppress the signals of the spacers.

3.6 Testing of a rotating coil configuration, especially for the rolled-in sections of the tube sheet.

3.7 Experimental study of the signal amplitude and phase as a function of frequency and defect shape.

3.8 Design of several probes for the plated walls of reactor vessels and their testing on a test wall and on plated test specimens.

3.9 Theoretical and experimental study of the possibility to determine the depth of defects in plated walls by means of calibration curves, obtained on test specimens.

3.10 Design and fabrication of an electronic device for the fast measurement of the angles of signals in the complex plane.

4. Experimental Facilities

See annual report 1977.

5. Progress to Date

re 3.1 Analysis of the influence of different parameters (temperature, surface conditions, composition of water) on the distance signal.

re 3.2 Procurement of a suitable cable for the transmission line. Production of pre-amplifier, fit to be used inside the tubes. Modification of the test unit to adapt the transmission line. Testing test unit to adapt the transmission line. Testing the line.

re 3.3 Design of a simple computer program to calculate the distortion of test signals by the transmission line. Parameters are signal amplitude and phase, test frequency, length of the cable.

re 3.4 Development of an electronic unit to feed a sythetically produced defect signal into the test coil and to control its correct processing by the whole test unit. Finishing of the signal generation unit. Matching the unit to the test coil. Design and fabrication of the evaluation unit.

re 3.5 Fabrication of segmented coils and their testing on S.G. tubes with spacers. Attempt to minimize lift-off signals by variation of field coil geometry.

re 3.6 Modification of a rotating jig. Adaption of special rotation probe. Study of the possibility to suppress signals of spacers and rolled-in sections.

re 3.7 Manufacture of a test tube with different artificial defects. Use of a well guided coil. Measurements and records of defect and spacer signals with variation of parameter. Excitation with two frequencies simultaneously.

re 3.8 Development and production of 5 different probes with variing field and measuring coil geometry. Comparison between the probes by means of records, obtained on a test wall and on plated test specimens with both artifical and natural defects.

re 3.9 Production of austenitic test specimens with saw cuts of different depth and width. Records of defect signals with variation of frequency and probe geometry. Evaluation of the test results.

re 3.10 Design of the different components (detection of signal maximum etc.) Providing the necessary signals from the test unit. Manufacture of a suitable kit for the electronic unit with an indication panel.

6. Results

re 3.1 The tested method is not applicable because of the strong influence of the water composition on the measuring signal.

re 3.2 The employed cable for the transmission line shows a good noise rejection. Because of the signal damping there exists the necessitiy of a pre-amplifier.

re 3.3 At given conditions (cable lenght, frequency) the cable produces resonances which can distort the defect signal both in amplitude and phase. If the changes of impedance of the coils, produced by a defect are, however, small in relation to the absolute value of the coil impedance, these effects can be neglected.

re 3.4 As far as now no resulty available.

re 3.5 With segmented coil there can be reduced the signals from the spacers, on the other hand the defect signals are thereby reduced with an increase in lift-off effect. The test results are worse than those from normal cylindrical coils.

re 3.6 The test results with a rotation probe are, with spacers as interference factor, in some degree better than those obtained with conventional coils. There exists,however, no conception of suitable mechanics for the in-situ inspection of long tubes. As for the testing of the rolled-in parts, there the mechanics are less complicated and the results much better than with normal coils.

re 3.7 There exists a frequency range where the combination of two frequencies yields the best defect resolution. Under these test conditions a 30 % of wall thickness deep longitudinal saw cut can be automatically detected in the region of a spacer.

re 3.8 The smaller the measuring coil of the probe the better the defect resolution. With a 10 mm diameter coil a 1,5 mm deep saw cut can be detected on the test wall and on the test samples.

re 3.9 Defect width affects very little both signal amplitude and phase. In the region of small defect depths exists linear relation between depth and amplitude.

re 3.10 No results yet.

7. Next Steps

re 3.1 Use of the two-frequency test set to obtain a measuring signal of the distance probe to surface.

re 3.2 Providing a cover for this part of the cable, which passes the cable driving mechanics.

re 3.4 Development of the evaluation electronics of the monitor facility. Putting into operation of the unit.

re 3.8 Finishing the experimental studies on the test specimens. Development of a test method to find sub-surface defects. Use of the specially manufactured test samples.

re 3.9 Checking of the predicted defect depth in the test specimens by mechanical destruction.

re 3.10 Putting into operation of the unit, especially during two-frequency testing. Attempt to predict defect dangerousness by simultaneous automatic evaluation of defect amplitude and phase.

re 3.11 Test unit preparation for the comparison test on a steam generator model.

8. Relation with other projects

RS 255, RS 256, RS 257

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1978 - 31.12.1978	12.3	RS 230

Vorhaben/Project Title	Land/Country
Untersuchungen zur Leistungsfähigkeit der akustischen Holographie, vor allem im Vergleich zu fokussierenden Prüfköpfen bei der ZfP Investigations of the efficiency of acoustical holography especially in comparison to focused beams in NDT	FRG **Fördernde Institution/Sponsor** BMFT **Auftragnehmer/Contractor** Bundesanstalt für Materialprüfung (BAM) Berlin Laboratorium 6.21

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.10.1976	31.3.1979	Dr.Kutzner,Dr.Wüstenberg
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	31.12.1978	

1. General Aim

Development of simplified methods of
- acoustical holography with numerical reconstruction
- flaw sizing by scanning with focused beams.

Application of the developed methods to the analysis of ultrasonic indications under practical circumstances.

2. Particular Objectives

-

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 indicence, oblique incidence, tandem-technique, distance of scanning points), development of the reconstruction- and interface-software (AD-converter, plotter) for scanning in one- and two-dimensions.

3.3 Holography in tandem-technique

Investigations of the efficiency of this method, experiments on test-blocks.

3.4 Construction and testing of focusing probes

Checking of the available construction method, manipulator for focusing probes, adaption of the focusing probes on the curvature of the surface.

3.5 Experiments with the probe-systems developed for holography- and focusing systems.
Testing of the developed systems on the test-blocks, which were produced in other research programs of reactor security, estimation of the efficiency of the developed techniques.

4. Experimental Facilities, Computer Codes

To 3.1 Manipulator system, electronic equipment for acoustical holography

To 3.2 Reconstruction software (FORTRAN), driver-software for AD-converter, CRT-terminal and plotter (Assembler)

To 3.3 Tandem-manipulator, electronic-equipment

To 3.4 Device to the production of acoustical lenses, digital equipment for positionning.

To 3.5 Whole equipment

5. Progress to Date

To 3.1 Some new probes (2 MHz and 4 MHz types) for perpendicular and inclined incidence are available as well as design specifications and criteria for probe dimensioning. Experiments on a focusing probe with a parabolic mirror were done.

To 3.2 Theoretical investigations and practical experiments on the choice of parameters and probes for the numerical acoustical holography.

For the two-dimensional holography (perpendicular incidence, inclined incidence, tandem-technique) a software package has been developed and tested. A driver module for producing a raster-type scan of the probe manipulator system is available.

In 1978 there are added several program parts, e.g. gate positioning by a microprocessor in order to obtain a more comfortable system handling.

To 3.3 Investigations under laboratory and practical conditions on natural defects on several test objects (see 3.5).

To 3.4 Preparing of work sheets for the application of focusing probes.

To 3.5 Test of the numerical acoustical holography (perpendicular incidence, inclined incidence, tandem-technique) under laboratory and practical conditions on several test objects (nozzle weld of a pressurizer, circumferential welds of thin walled tubes (14 mm - 24 mm), EPRI specimen, thick walled (360 mm) procedure quality control specimen, vessel wall).

Examination of the possibility of the acoustical holography to discriminate between defect indications and indications caused by the geometry of the specimen.

5. Results

To 3.1 The experiments with the 4 MHz probe on thin walled components were satisfactory. Defects in a depth of 15 mm could be analyzed. Furthermore, a crack in a depth of 110 mm and an extension of 2.5 mm x 6 mm (shown by a metallographic examination) has been measured with an accuracy of ± 0.5 mm.

For measurements on objects with bad surface conditions, e.g. cladded components, the applicability of probes with a higher frequency (e.g. 4 MHz) is restricted. In such cases the application of TR-probes may be an alternative solution. The concept of the focusing probe with a parabolic mirror has not been proved to be reliable due to too many reflections from the wedge shape.

To 3.2 The work on the choice of parameters and probes for the numerical acoustical holography has been completed. In the second technical report a detailed discussion of the results is given.

The software package for the two-dimensional holography has been proved to be correct from logical point of view. Due to memory limits of the computer system it was not possible to test the package on real reflectors.

To 3.3 Acoustical holography in tandem-technique requires no special probes. Tandem-technique means here: stationary transmitter and movable receiver. In case of moving both probes the resolution decreases.

In order to obtain satisfactory results with tandem-technique an accurate movement of the time gate is necessary. Another problem is the distortion of the illuminating sound field by cladded surfaces. For a better solution further investigations are necessary.

To 3.4 Theoretical and experimental determination of DGS-diagrams for the sensitivity setting. Adaption of focusing probes on curved surfaces.

To 3.5 The practical experiments in 1978 have shown that all important defect parameters, especially extensions in depth and length can be determined with an accuracy of ± 1 mm.
The wall thickness of specimen to be examined can vary between 15 mm and 360 mm.

An important advantage of the acoustical holography in comparison with conventional techniques is the better ability to discriminate between defect indications and indications caused by the object geometry.

The examination of near surface cracks is a further problem. In order to analyze defect agglomerations in detail it seems to us that the surface holography may in some cases be more efficient than the line holography.

7. Next Steps

To 3.1 Third technical report

To 3.2 Surface holography on artificial defects (after memory expansion)

To 3.3 -

To 3.4 Elaboration of a comparison of numerical acoustical holography versus focusing probe technique.

To 3.5 Further practical experiments on cladded components

To 3.1 - Final report
3.5

8. Relation with other Projects

Real-time ultrasonic holography (IzfP-Saarbrücken) (RS 102) reactor security research program FB 2703

9. <u>References</u>

1. technical report; 2. technical report

H. Wüstenberg, E. Mundry, J. Kutzner: Experience with Flaw Size Estimation by Ultrasonic Holography with Numerical Reconstruction; Vortrag auf der Konferenz "Nondestructive Evaluation in the Nuclear Industry; Salt Lake City, Febr. 1978

V. Schmitz, J. Kutzner: Holographische Rekonstruktion von Werkstoff-Fehlern; Vortrag beim Seminar "ZfP in der Kernreaktortechnik", Saarbrücken, März 1978.

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

GRS, Köln

Berichtszeitraum/Period 1.3.1978-31.12.1978	Klassifikation/Classification 12.3	Kennzeichen/Project Number RS 327
Vorhaben/Project Title Investigations on the Defect Evaluation of Reactor Structures by Means of Holographic Interferometry Untersuchungen zur Fehlerbewertung an Reaktorbauteilen mit Hilfe der holographischen Interferometrie		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Inst.f.MiM, Hannover BIAS, Bremen
Arbeitsbeginn/Initiated 1.3.1978	Arbeitsende/Completed 31.8.1979	Leiter des Vorhabens/Project Leader Dr.-Ing. W. Jüptner
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

0. General Remarks

The projects RS 327 and RS 329 were approved as work package with common work plan. For the execution of the investigations the following institutions cooperate:

- Dornier System (DS), Friedrichshafen
  Dr. K. Grünewald, Dipl.-Phys. Wachutka
  RS 329
- Institut für Meßtechnik im Maschinenbau (MiM), Hannover
  Dr. H. Kreitlow, Dipl.-Ing. Geldmacher
  RS 327
- Bremer Institut für Angewandte Strahltechnik (BIAS), Bremen
  Dr. W. Jüptner, Dipl.-Phys. B. Fischer
  RS 327, subcontract RS 327-2

The measurements carried out with Messrs. Krupp were looked after by Dipl.-Ing. A. Nölker and R. von Lanken. Messrs.Krupp places the tank, the personnel, and the material free of charge at their disposal.

For the theoretical investigations, a subcontract has been submitted to the

-Institut für Festkörpermechanik (IFKM), Freiburg
 Dr. E. Sommer, Dipl.-Ing. H. Kordisch

1. General Aim

The preceding activities in the scope of the research orders RS 102-22 and RS 148 indicate that, additionally to fault detection and localization, the holographic interferometry is also suitable for fault qualification. An essential reason for this is to be found in the nature of this testing method, since with it deformations covering an area can be determined.
The objective of this research project resides in the determination of the possibilities and limits of a fault qualification at reactor component parts by continous laboratory and field studies. Moreover, it aims at a further development of the holographic-interferometric measuring technique to enable measurements at reactor component parts that cannot be insulated from vibrations in the holographic-interferometric sens.

The result of the complete investigations shall thus prove, if a fault assessment can be achieved with sufficient security by holographic-interferometric measurements.

2. Particular Objectives

The research activities consist of the following main fields:

- The holographic test assemblies and methods are to be further developed with respect to their application at installed reactor component parts that are not insulated from vibrations. It is planned to test the results at the HDR after a successful conclusion of these activities. Additionally, the conditions for a simplified transfer of the interferometric data to EDP systems are to be elaborated.

- The studies for the qualification of material faults by means of the holographic interferometry include not only a theoretical elaboration of surface deformation in function of defects and stress intensities but also experiments to determine the stress intensity prior to cracks from the measured surface deformation.
  In this connection, the crack lengths and the applied stresses shall be varied.
- The above investigations are to be extended to the detection and qualification of structural anomalies within the range of welded joints.
- An application of the knowledge gained from the preceding activities at the HDR is intended.
- The holographic test assemblies and methods were to be further developed with respect to their application at installed reactor component parts that are not insulated from vibrations. For this purpose, also measurements were carried out at a pressure tank of Messrs. Krupp, Forschungsinstitut, Essen.
- The studies for the qualification of material faults by means of the holographic interferometry should include in the period under review above all the theoretical elaboration of surface deformation in function of defects and stress intensity. Experiments to determine the stress intensities from the measured surface deformation follow in 1979.

3. Research Program

In the scope of the research project the institutes participating prepared or started to prepare the following points in 1978:

3.1 Further development of holographic test assemblies and methods for the measurement at reactor pressure vessels

3.1.1 Further development of holography with test assemblies that are not insulated from vibrations

3.1.2 Design and construction of a working platform for the measurement at the pressure tank of Messrs. Krupp and at the HDR

3.1.3 Testing of the developments according to 3.1.1. and 3.1.2 at the large-sized tank of Messrs.Krupp and at the HDR

3.1.4 Testing and assessment of recording and evaluation techniques with increased measuring accuracy
For this purpose, the differential pressure interferograms recorded for RS 148 are evaluated by means of the contour projector to determine the amount of sensitivity with which the existing graded local inhomogenities can be proved. Moreover, it shall be decided if the Dändliker method is to be included in the experimental program.
Handled with priority by: DS

3.2 Further development and practical test of evaluation methods with respect to a comparative stress analysis of pressure vessels

3.2.1 Adaptation of the programs to the special measuring task for the determination of deformations and strains

3.2.2 Considerations about the graphic representation of three-dimensional deformation zones

3.3 Stress analytical investigations with local inhomogenities

3.3.1 Calculation of surface deformation for defects of different position and size by the finite element method for tensile test and component part defect

4. Experimental Facilities, Computer Codes

ad 3.1 A tank with the dimensions 0.9 m Ø x 1.25 m was procured for pretesting. With this tank the designed working platform for the pressure vessel of Messrs. Krupp was tested (MiM). A working platform, constructed for these measurements, essentially consistes of a very rigid mount base to which the optical components are flanged (BIAS, MiM).
The evaluation techniques with increased measuring accuracy require the expansion of existing test systems and evaluation programs.
The present evaluation program "displacements" must be expanded for the calculation of strains and stresses as well as adapted to the application to curved object surfaces (reactor

pressure vessel). Moreover, a plot program was prepared for the three-dimensional representation of holographically measured deformation zones (DS).
The programs for the evaluation of fracture analysis are to be modified.

ad 3.2 KWU and KFA Jülich, respectively, were approached in order to get program parts for the three-dimensional representation and stress-analytical calculations (DS).

ad 3.3 IFKM obtained a subcontract for the calculation of surface deformation with a program prepared by them. Personal contacts with IFKM ensure that the existing FEM program is adapted in the best possible way to the problems to be anlaysed.

5. Progress to Date

a) Dornier System

ad 3.1 Further development of holographic test assemblies and methods for the measurement at reactor pressure vessels.
In cooperation with MiM and BIAS, a work platform was designed and tested for measurements at the Krupp pressure vessel. This platform is described in detail under b). With it, the measurements with Krupp were successful. Moreover, laboratory experiments served to clear up detailed problems with the quantitative evaluation of interferograms of objects that are not insulated from vibrations (particularly the phase reference mirror method). The evaluation of the measurements performed with Kurpp was started and the problems which appeared in this connection were anlaysed theoretically and experimentally. Since phase reference mirrors rigidly connected with the object were used in the measurements for the compensation of object vibrations, the usual evaluation equations are to be corrected.
In simple laboratory experiments the corresponding interferograms were simulated and the evaluation was compared in case of a stationary reference light source and using a phase reference mirror. Moreover, additional experiments to explain the difference between these two methods were performed with identical object displacements. Basing on this, an equation

system corrected against the original evaluation formula was set up with which it should be possible to calculate the true displacements for all object points also in the general case.

ad 3.2 Further development of the evaluation methods with respect to a comparative stress analysis of pressure vessels.
An improved program for the calculation of circumferential strains and changes of diameter was prepared from holographic-interferometrically measured displacements at cylindrical tubes. The accuracy to be achieved was tested experimentally.

The evaluation programs have been adjusted to the special test methods with the holography of objects that are not insulated from vibrations. Moreover, a program was developd and tested for the perspective graphic representation of three-dimensional deformation zones.

ad 3.3 Stress-analytical investigations with local inhomogenities
A tensile testing device was conversed and improved according to the requirements of the holographic-interferometric stress analysis. The procurement of test objects of reactor steel was initiated. The experiments will be started after receipt of the reactor steel test specimen. The experiments to be performed will be arranged according to the theorectical results from the calculations according to the finite element method.

b) <u>Institut für Meßtechnik im Maschinenbau</u>

ad 3.1 In cooperation with DS and BIAS, a mobile test facility was designed for the preparation of holographic interferograms of pressure vessels. This facility was constructed at the Insitut für Meßtechnik im Maschinenbau together with BIAS and the three institutions (MiM, BIAS, DS) used it for the simulation of deformation measurements at a pressure vessel having the dimensions: length 1,250 mm and Ø 900 mm.
Since the test facility should directly be connected with the object for the planned tests at the pressure vessel with Messrs. Krupp, a rigid basic mount was constructed of x-95 profiles which could easily be fixed to the pressure vessel and accommodate the optical components of the test facility.

The facility allows the application of two different recording methods:

- single-beam holography for the determination of deformations along a surface line near the object cover (BIAS)
- holography with phase reference mirror for the determination of deformations along a cover line (diamter) (MiM).

Perturing motions of the object are compensated when using a phase reference mirror at the object. Pretests cleared up the point on the measuring surface at which the phase reference mirrors were to be placed as well as the influence of the perturbed phase relationship of the reference beam cuased by ambient vibrations on the reference beam reflector surface.
The investigations at the pressure vessel with Krupp were performed together with DS and BIAS.
To fix the test facility (frame and optical components), mounting elements had to be welded to the object. Basing on the experience gained in the pretests, the mobile test facility connected with the object was constructed, fig. 1.
The object was deformed by increasing the water pressure by means of a hand pump. In this connection, the deformations resulting at the pressure vessel were measured by the holographic method, by means of strain gauges and by an inductive displacement pick-up.

c) Bremer Institut für Angewandte Strahltechnik

ad 3.1 The activities mentioned under a) and b) were coordinated with Dipl.-Ing. A. Nölker of Messrs. Krupp.
The design, set-up, and testing of the mobile test facility were carried out in cooperation with MiM and DS. Details are mentioned under b).
The method of the single-beam holography in which the object is illuminated through the hologram plate rigidly connected with the specimen was developed in pretests in their own laboratory.
Using this method, the object is illuminated through the hologram plate rigidly connected with it. The advantages of this method are the simple design and the insensitivity against perturbed object motions. It is above all disadvantageous

that only a small surface can be illuminated and the coherence length of the laser restricts the distance between object and hologram.
The laboratory experiments were subsequently tested at a tank accomodated in the test facility commonly developed.
The theoretical conditions were revised for the evaluation of the holographic-interferometric measurement using a phase reference mirror. Corresponding laboratory tests will be carried out by Dornier System.
The conditions for the evaluation of the interferograms recorded by the single-beam holography will also be revised.

ad 3.3 The order to the IFKM for the theoretical determination of surface deformations above defects was specified in its details. IFKM has meanwhile set up the dot screen for the FEM calculations and submitted the first results.

6. Results

a) Dornier System

ad 3.1 The quantitative evaluation of the tests with Krupp have been initiated, fig. 2. Final results are not yet known, since the evaluation equations are to be corrected for the use of phase reference mirrors. For this purpose, laboratory experiments have been carried out by means of which the corrections terms should be determined.
The laboratory experiments established with identical object displacements distinct differences in the interferograms recorded with phase reference mirrors or stationary reference illumination, respectively. Using a phase reference mirror the strip density is for instance lower and the fringe Position depends with fixed point of observation on the hologram range used for reconstruction. The total phase position in the point of observation results from phase displacement due to a change of the object and from phase displacement due to a change of position of the phase reference mirror. This means the introduction of a correction term depending on position and displacement against the original evaluation equations. The influence of the phase change of the reference beam can be eliminated on the

assumption of a linear course of the displacement in the ambiency of the apparent zeroth order.

ad 3.2 The evaluation with increased measuring accuracy of the interferograms established a considerably improved recognition of lacal defects against a purely visual evaluation. A complete automatic control or the method seems to be possible. As an example, fig. 3 shows the interferogram of an internal pressure loaded steel tube with a crack of 0,28 mm depth in the inside wall (11.5 % of the wall thickness). With unprejudiced visual judgment of the interferogram, the defect can practically not be located. The computer printout for the variation in slope of the data N (x), i.e. interference order against the coordinates already shows a distinct fault indication. Considerable improvements of the fault recognition will result by the approximation of the data by polynomials of different degrees; for instance fig. 4: difference between the data and the approximated straight line. More serious faults will be indicated with adequate clearness.

With the application of the program for the determination of circumferential strains at cylindric bodies from holographically measured displacements, calculation and experiment established a mean inaccuracy of the radius change of ± 0.2 µm, i.e. a fault of 4 % for dr = 5 µm, due to the holographic uncertainty in measurement of ± 0.05 µm with respect to the displacement measurement of a single point.

For the perspective representation of three-dimensional deformation zones a grid is put on the object surfache which will turn into a deformed grid due to the object deformation. In this connection, the orientation of the point of observation can be freely selected relatively to the object. The program then projects all points of a grid on a level square with the connection line between center point of the object surface and point of observation and supplies a format-filling drawing. The display is made on a graphic screen with hard-copy output. The program was as yet tested in idealize deformation zones. In fig. 5, a plane surface is deformed according to the function of $dz = a \cdot \frac{\sin\beta r}{r}$. Fig. 6 represents the barrel-shaped deformation of a quarter of a tube with fixed ends.

b) Institut für Meßtechnik im Maschinenbau

ad 3.1 Measurements were performed for an optimum setup of the work platform. The following result was achieved:
With given size of the measuring surface on the face of the test object the position to attach the reference mirror essentially depends on the mutual phase relationship as well as on the amplitude of the object vibration. With a measuring surface of abt. 400 mm Ø the testes established a distance of abt. 200 mm between reference mirror and measuring line. This leads to a simplified test assembly, as the reference beam could be deffected from the object illimination; i.e. a common source point can be used for object and reference wave. The use of a separated object and reference beam source point established no improvement. No improvement was achieved either in focusing the reference beam on the phase reference mirror to eliminate the influence of natural vibrations of this mirror. Considering all optimization results, the mobile test assembly was established for deformation measurements at pressure vessels.
The investigations at the pressure vessel of Krupp were performed in cooperation with Krupp, DS and BIAS in the third quarter of 1978. The measurements were carried out with a change of pressure of 0.5 bar and base pressures of 10, 20 and 30 bar. Fig. 7 shows the interferograms achieved.
The object section along the given and marked measuring line has a diameter of abt. 500 mm. The interferograms can be interpreted that the deformation increases relatively symmetrical against the radius of the cover in direction to the cover center: to be recognized by the concentric and nearly equidistant interference fringes of about the same distance.

Due to the theoretical fundamentals of this method all achieved deformations of the cover refer to the position of the reference mirror or of the common line through the position of the reference mirror.
The measurements showed that by means of the developed and

optimized mobile test facility using the reference mirror established also at test assemblies and objects that are not insulated from vibrations.

c) Bremer Institut für Angewandte Strahltechnik

ad 3.1 Since the tests described under b) were performed in co-operation with MiM and DS, the results are also to be found under b).

The evaluation equations are to be corrected for the evaluation of the interferograms achieved by the phase reference mirror method, since the reference mirror carries out a rotation of the surface on the position of attachment and thus simulates a rotation of this quantity for all points of the surface. The new evaluation equations which are, however, not yet revised experimentally, are:

$$n_I \cdot = \nu_{zI} \cdot t_z + \nu_{yI} \, (t_y + \alpha_I \, (Z-Zp))$$

$$n_{II} \cdot = \nu_{zII} \cdot t_z + \nu_{yII} \, (t_y + \alpha_{II} (Z-Zp))$$

n: interference fringe order

: wave length

ν: geometric function, sensititivity vector

t: translation

α: rotation of phase reference mirror

z: coordinate in the surface

y: coordinate perpendicular to the surface

Zp: coordinate of the phase reference mirror

In the ambiency of the mirror, the following equations can be applied:

$$n_I \cdot = \nu_{zI} \cdot t_z + \nu_{yI} \, (t_y + t_y \, (Zp))$$

$$n_{II} \cdot = \nu_{zII} \cdot t_z + \nu_{yII} \, (t_y + t_y \, (Zp))$$

These equations must still be checked. They enable however the evaluation in two steps, the first step defining the angle α.

ad 3.3 IFKM meanwhile established the structure for the FEM calculations, fig. 8, and performed first calculations. In this respect, it was assumed that the specimen consisted of the material 22 NiMoCr 37 and had the dimensions 500 x 100 x 30 (length x width x thickness in mm). Moreover it was defined that the specimen is clamped at the bottom and pulled upward. For the first computer runs a central fault with different diameters was assumed.

Structure	Fault Ø	a/W
L 10	14	0.07
L 11	20	0.1
L 12	40	0.2

W: specimen width
a: fault radius

The fault itself was assumed as being round. (Exentricity $e_L/W = e_R/W = 0.5$).
The surface displacements were calculated according to the finite element method with the program system SOLID-SAP [1] on the UNIVAC 1100/80 of the computing center of the Freiburg University. The structures were partly automatically divided into isoparametric triangular elements with three corner nodes by the mains generator TOPONET [2]. The preparation of the structure and the subsequent evaluatión of the results were made by the computer Up 2100 S with IFKM by means of paricularly processed evaluation programs.

With respect to expenditure, calculation cost, and the required accuracy, the initial structure was at first optimized using the example L 11. Additionally to the control of the marginal conditions, the comparison of the calculated stress superelevation at the edge of the hole with analytical results acc. to PETERSON [3] presents an accuracy test (with $\alpha_K = \delta_{max}/\delta_o$) :

structure	α_K analyt.	α_K FEM	F %
L 10	3.07	3.07	0.
L 11	3.14	3.12	-0.6
L 12	3.74	3.72	-0.5

Experimental values for the surface displacements are not yet known. But, since the program system SOLID-SAP works according to the displacement variable method, it can be assumed that the error in the calculation of the surface displacement will be smaller or, at maximum, equal to that of the stresses.

The result shows that for a centered round defect the maximum of the stress intensity always remains in the same position, while the amplitude increases according to the diamter, fig. 9. With this, it seems to be possible to point to the size of the defect. The experiments in 1979 must establish the proof for this fact. Theoretically it is to be tested how the amplitude of the stress intensity and the position of the maximum change with varied defect position. The first calculations in this respect are being carried out at the moment. Since however the nodal network for the calculations is, at the same time, put into a clearer and optically more informative shape, the results are not yet complete.

7. Next steps

ad 3.1 The investigations at test facilities not protected against vibrations are finally treated so that a complete flow chart will bi available for later measurements at large-size objects, e.g. the HDR. For this purpose, it might be possible to carry out additional tests with Krupp.
The evaluations will be checked and corresponding programs be prepared that will be interlaced with other evaluation programs.

ad 3.2 The works for this program point will be continued with concentrated efforts and tested and completed in experiments in the laboratory and in measurements at large-size object.

ad 3.3 The FEM calculations will be performed by IFKM. The possibilities of a support by the program "FEABLE" existing with BIAS are to be discussed. In this connection it must be pointed out that this program has not yet been applied to the present problem.
The preparations for the experimental investigations with respect to stress intensity measurement are taken. The performance of these activities is however concentrated on 1979.

8. Relation with other Projects

RS 102-22, RS 148, RS 327, RS 329

9. References

[1] E.L. Wilson: "SOLID-SAP. A static analysis program for three dimensional solid structures", California University, BErkeley, PB-209 949 (1972)

[2] H. Führing: "Finit-Element-Lösungen von Kerb- und Rißproblemen mit Hilfe automatischer Netzerzeugung", Heft 24 (1973), Inst. für Statik und Stahlbau TH Darmstadt

[3] R.E. Peterson: "Stress concentration design factors", John Wiley and Sons, (1966)

10. Degree of Availability of the Reports

-

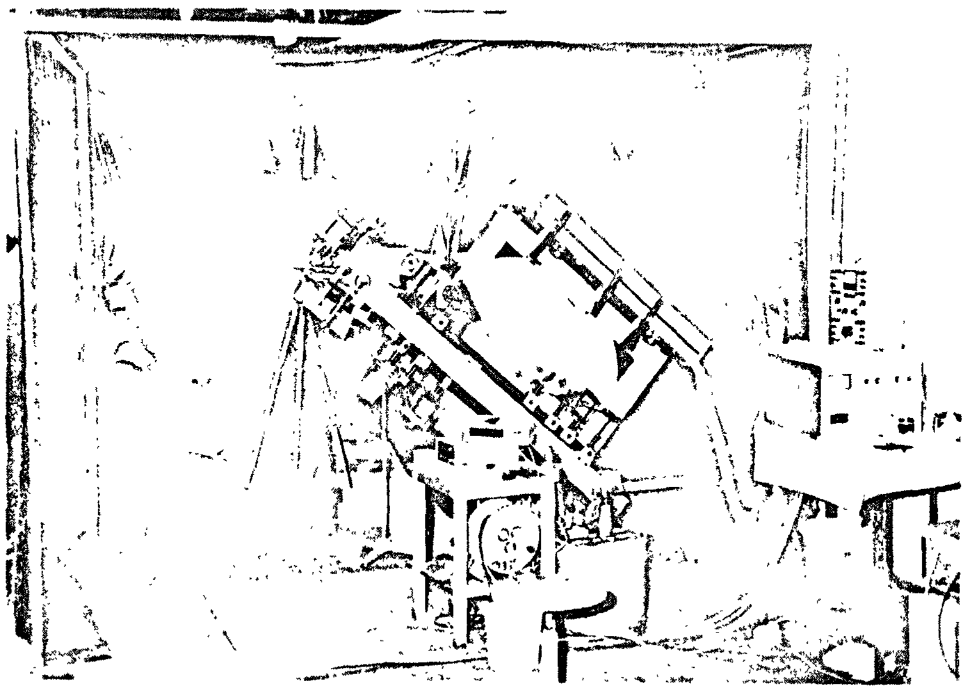

Fig.1: Mobile Test Setup in Operation at Pressure Vessel of Krupp

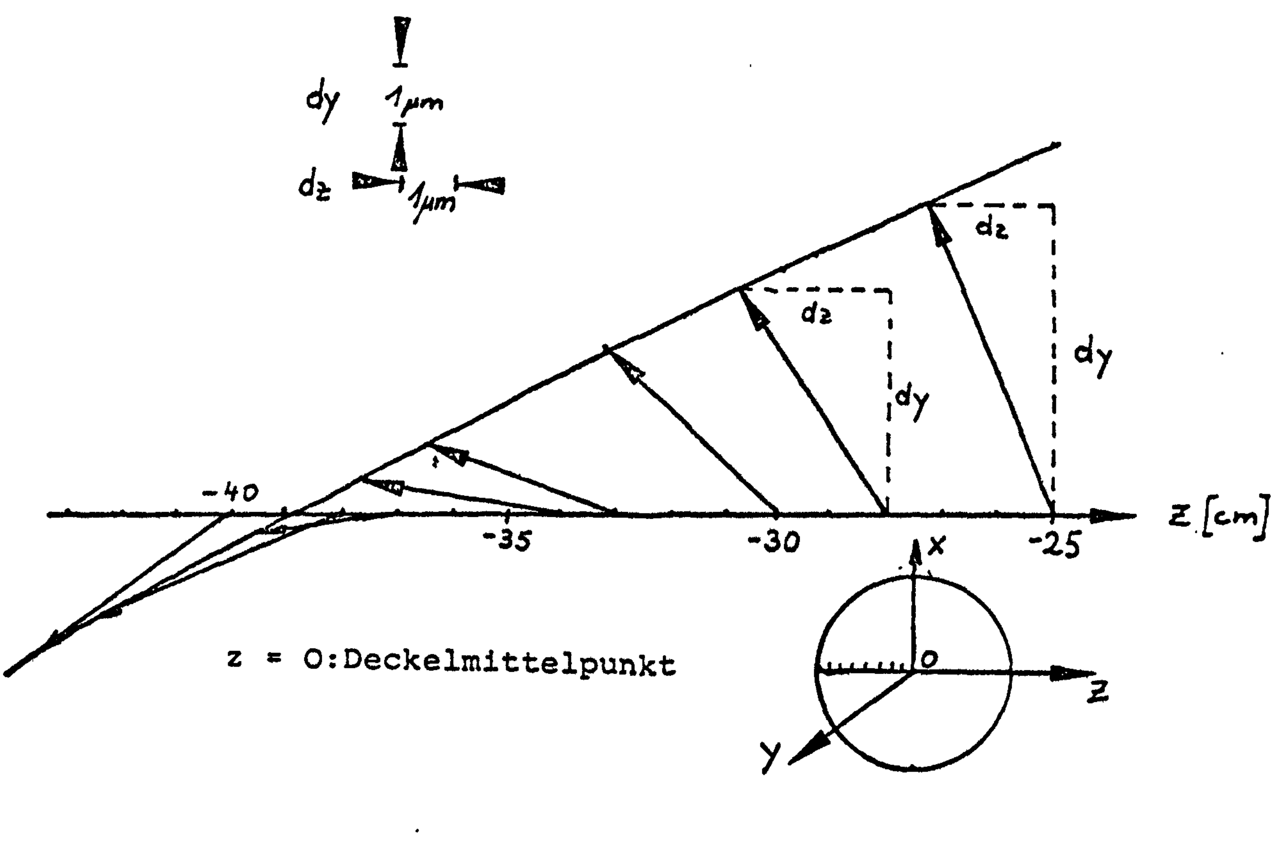

Fig.2: Displacements of Single Cover Points after Evaluation of of Measurements with Krupp

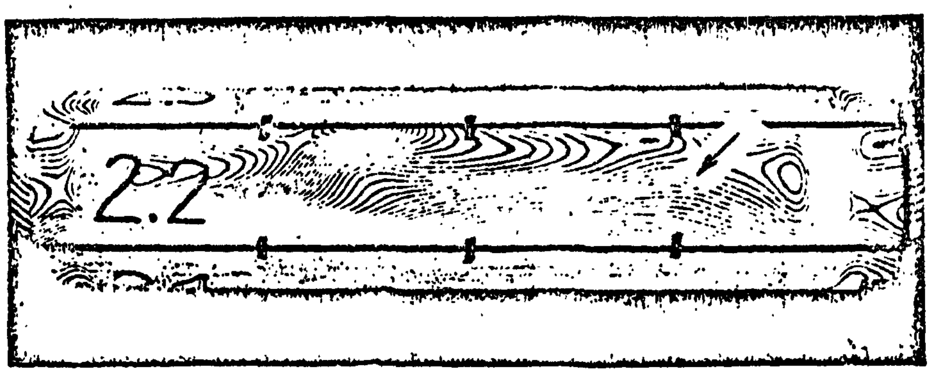

Fig. 3: Differential Pressure Interferogram of a Tube with Local Defect, arrow

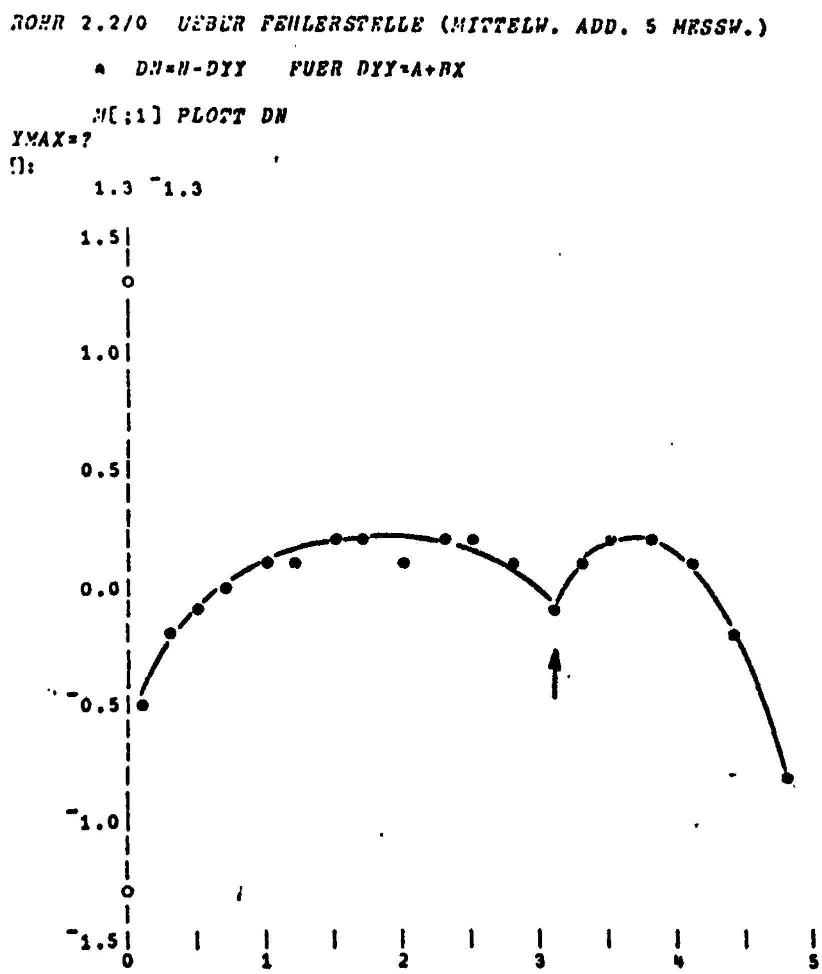

Fig. 4: Difference between Data and best fitted straight line

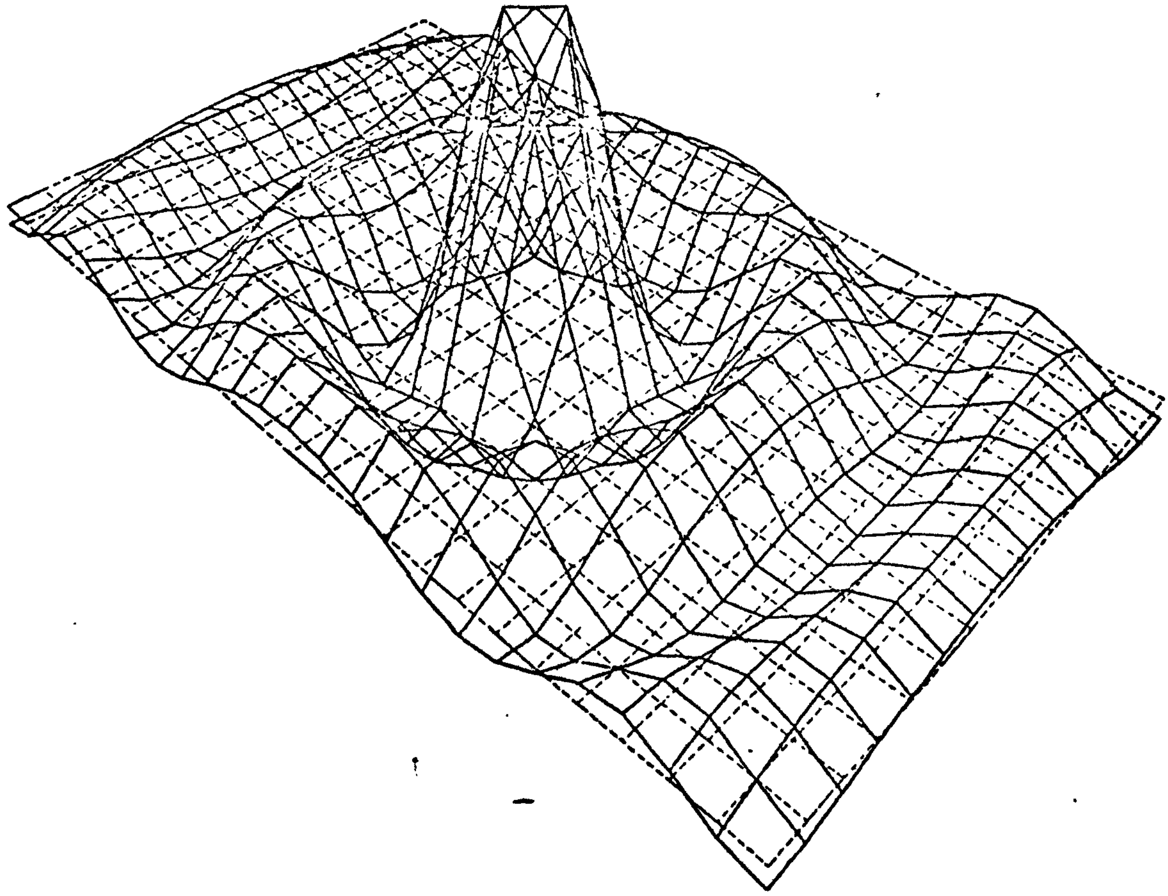

Fig. 5: Deformation of a Plane Surface acording to the Function
$dz = a \cdot \frac{\sin\beta r}{r}$

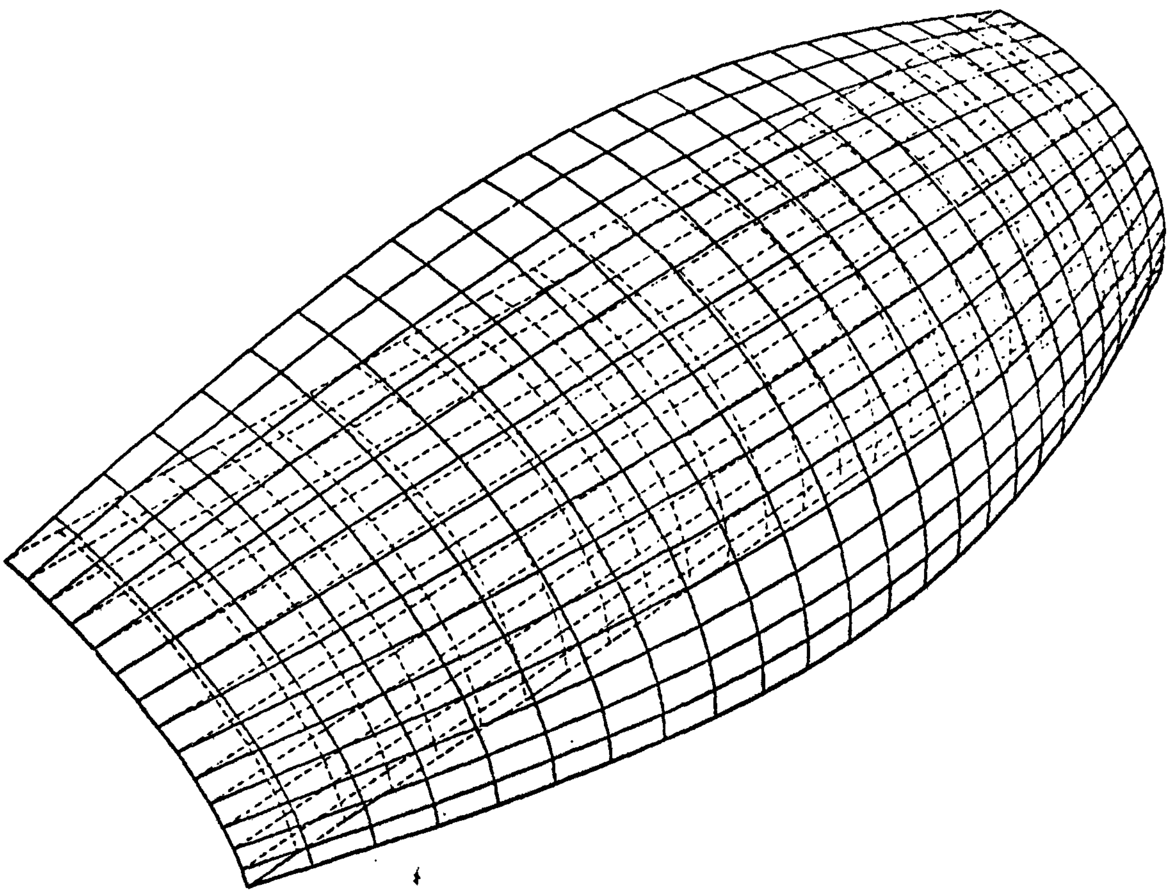

Fig. 6: Barrel-shaped Buckling of a Quarter of a Tube
with Fixed Tube Ends

Fig. 7: Holographic Interferogram Recorded by Phase Reference Mirror Method at Large-size Tank of Krupp

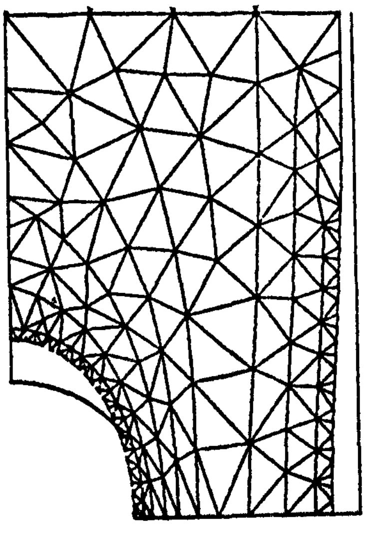

Fig. 8: Nodal Structure to Calculate Stress Intensity according to Finite Element Method

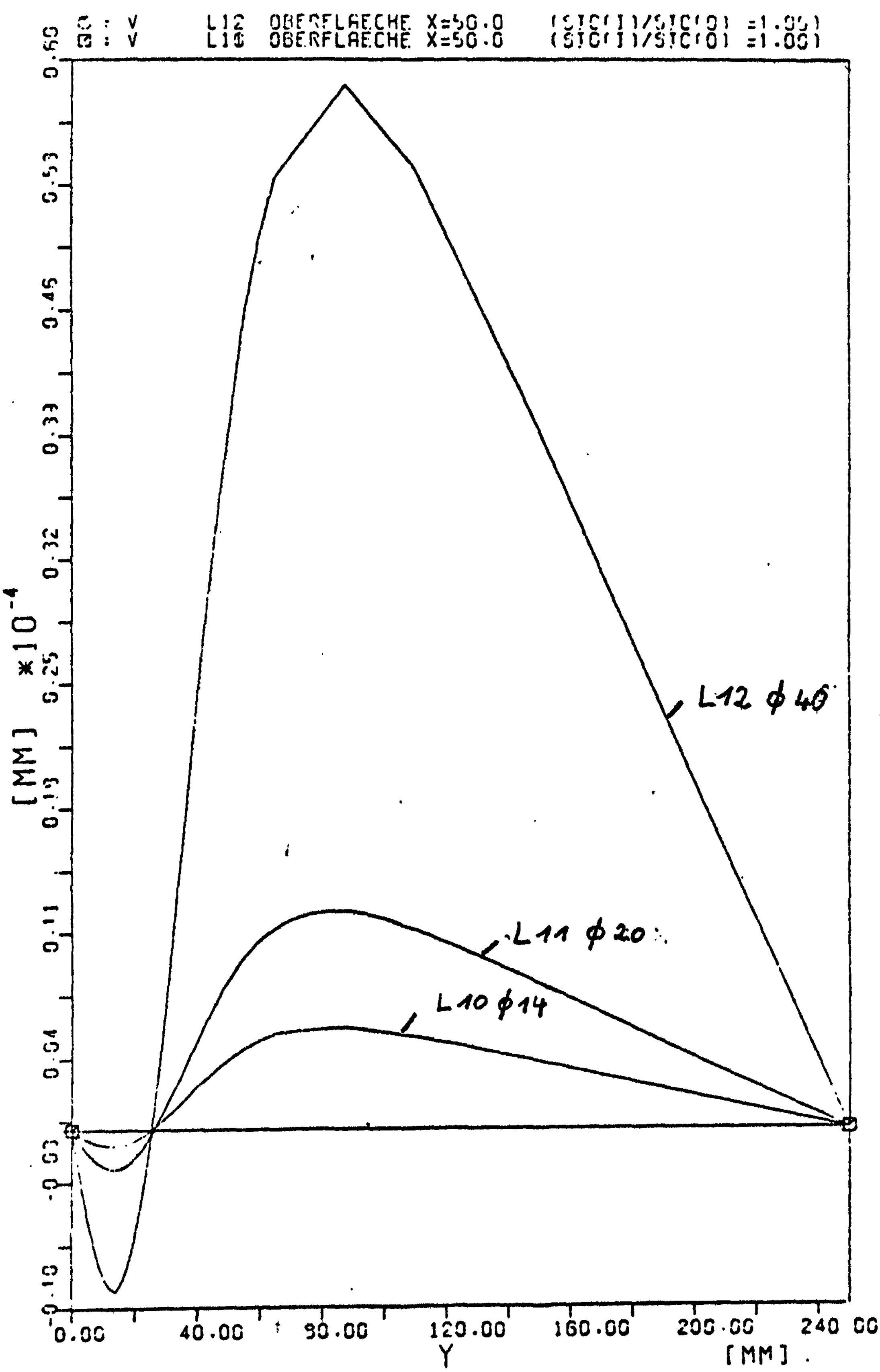

Fig. 9: Amplitude of Stress Intensity for a Round Defect in Center Position. Variation of Defect Diameter

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.3.1978-31.12.1978	12.3	RS 329

Vorhaben/Project Title	
Investigations on the Defect Evaluation of Reactor Structures by Means of Holographic Interferometry Untersuchungen zur Fehlerbewertung an Reaktorbauteilen mit Hilfe der holographischen Interferometrie	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Dornier System Friedrichshafen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.3.1978	31.8.1979	Dr. K. Grünewald
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

The report concerning RS 329 is contained in the RS 327 report.
ook there.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
Jan.1 - Dec. 31, 1978	12.3	RS 191 A

Vorhaben/Project Title	Land/Country
Schallemissionsmessungen an bruchmechanischen Proben, Phase 2: Messungen an Fertigungsfehlern Acoustic Emission Measurements on Fracture Toughness Specimens, Phase 2: Measurements on Natural Defects	FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Battelle-Institut e.V. Frankfurt Materials Behaviour Section

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.10.1977	31.7.1979	Eisenblätter/Jöst
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31.12.1978	

1. General Aim

These investigations are aimed at establishing correlations between the measured values of acoustic emission (AE) and the stress at natural defects, on the basis of results obtained in Phase 1 of the project. The reulsts of the investigations performed under Phase 2 are to make AE suitable for testing the pressure vessels of nuclear reactors.

2. Particular Objectives

The investigations will be conducted mainly on large fracture toughness specimens and plates made from the materials used for light-water reactors, i.e. the steels 22 NiMoCr 37 and 20 MnMoNi 55. Several types of defects occurring in practice will be investigated, which - unless available in large parts - will be generated by simulation and welded into the large specimens. The defects to be investigated will include both defects in the base material, such as segregations. non-metallic inclusions and shrinkholes, and defects in weld joints, e.g. reheat cracks (cracks in the area adjacent to welds or undercladding cracks), hot cracks, cold cracks, slag inclusions, fusion defects and pores. To obtain the desired correlations between the measured acoustic emission and the defect sizes, comprehensive non-destructive and destructive tests will be carried out. In addition, measurements of the progagation of structure-borne sound will be carried out on a reactor pressure vessel by transmitting test signals from the two surfaces (inner and outer surface).

3. Research Program

3.1 Creating the conditions for carrying out the experimental phase. Supply of flawless and defective materials as well as preparation of specimens and construction of devices

3.2 Performance of experiments with large fracture toughness specimens and plates, including acoustic emission measurements

3.3 Evaluation of experimental results, establishing correlations between measured AE values and the quantities characterizing the defects

3.4 Measurement of acoustic wave propagation on a reactor pressure vessel

3.5 Elaboration of a final report and detailed planning of the possible continuation of the investigations

4. Experimental Facilities, Computer Codes

Design and construction of a pit with covering, which is to be used for the experiments with the large specimens, taking into account functional and safety aspects

Selection and installation of a hoist (portal crane): for handling the specimens and other experimental facilities

Design of a loading mechanism for the large-area fracture toughness specimens (LFTS) with a maximum force of 6 MN and placing a subcontract for the production of this mechanism

Installation of the pipings and fittings to which the loading mechanism, which has meanwhile been completed, is to be connected, and construction of the specimen holders for the LFTSs. Additional installation of hydraulic fittings for the space formed between pairs of large plates (LP) and generating internal pressure between the two LPs

Design of an improved data aquisition and evaluation system (DA system) for on-line evaluation of the measurements by means of a computer, including the establishment of correlations between measured quantities of acoustic emission and the quantities characterizing the defect (e.g. size and location, local stress, stress intensity factor). Collection and comparison of offers, discussions with offerers and final specification and placing of orders for the DA system.

5. Progress to Date

Re 3.1

Establishing contacts and carrying discussions with producers of materials, with manufacturers of components for nuclear power plants, and with representiatives of the Materialprüfungsanstalt (Materials Testing Office) in Stuttgart in order to be supplied with the necessary materials with and without defects

Calculation and preliminary dimensioning of the large-area fracture toughness specimens (LFTS) and the large plates (LP) as well as production of scale-model specimens of the two specimen shapes for stress analysis. Application of strain gauges and performance of strain measurements at different static loads, and rough evaluation of the results.

Final dimensioning of the two specimen types and design of an appropriate sealing element for sealing the space formed between the pairs of LPs, as well as drawing up the working drawings (Figs. 1 and 2). Placing orders for the first three flawless large specimens (1 LFTS and 2 LPs) for testing the experimental facility and for measuring the noise of the facilitiy. After competion of test measurements, defective parts will be welded into these specimens.

Re 3.4

Measurements of acoustic wave propagation on an empty reactor pressure vessel of a power pressurized-water reactor at the plant of the manufacturer. Acoustic pulses were introduced from inside and outside to simulate signals from defects, and the AE signals received were recorded.

Preliminary evaluation of part of the measured signal shapes.

6. Results

Re 3.4 (a)

Generation and Recording of a Test Signal at the Outer Surface (Simulation of an External Defect)

It is known that in addition to the dominating wave mode, the surface wave, other wave modes (probably volume waves propagating in a zigzag pattern) are excited. The percentage of these modes depends on the type of excitation (breakage of a pencil lead, transmission of an acoustic signal by means of different transducers connected either direkt or via a cone) or, more generally speaking, on the radiation characteristic of the source concerned.

The signal of the surface wave generated by the breakage of a pencil lead, as detected by the measuring cascade, reaches its maximum after a rise time of $t \approx 10\ \mu s \,\hat{=}\, \frac{1}{\Delta f}$ (Δf = filter bandwith of the measuring cascade) and then decreases by an average value of

$$A = A_0 \cdot \exp(-\ ^{t}/_{t_0}),\quad t_0 \approx 45\ \mu s \text{ (receiver of type S 140B)}$$

Re 3.4(b)

Generation of a Test Signal at the Inner Surface , Recording at the Outer Surface (Simulation of an Internal Defect)

The previous finding was confirmed that volume waves propagating in a zigzag pattern are recorded in the near field, while a wave packet appearing more and more homogeneous and propagating at a velocity of v $\approx$ 3 mm/ μs is recorded in the far field. The dominating volume waves in the near field were identified for the investigated case of spontaneous (relief) of a normal point force (caused by breakage of a pencil lead or by the test signal of a US transmitter). It is primarily the transverse wave T with its reflections $T^3, T^5, \ldots, T^n$ (n = odd = number of refelctions) and to a smaller extent the longitudinal wave L with the reflections L^2T, $L^3T, \ldots$, $L^{n+1}T^n$, where the longitudinal wave L coming from the interior was converted into a transverse wave T at the outer surface and back into a longitudinal wave at the inner surface.

7. Next Steps

Re 3.1 Completion of the experimental facility, supply of materials and preparation specimens

Re 3.3 Continuation of the evaluation

8. Relation to Ohter Projects

The project ist based on Phase 1 (measurements of fatigue cracks, RS 191). The investigations are being conducted in close cooperation with the IzfP, Saarbrücken, the IFAM, Bremen, and the IFKM, Freiburg (RS 196).

9. References

(1) R. von Klot, Ausbreitung von simulierten Schallemissions-Impulsen in dickwandigen Bauteilen Report BF-R-62.945-1 (Dec. 1977)

(2) H. Jöst, J. Eisenblätter, Schallemissionsmessungen bei bruchmechanischen Versuchen an Werkstoffen leichtwassergekühlter Reaktoren. Report BF-R-62.945-2 (April 1978)

(3) T. Fischer, Schallemissionsmessungen bei bruchmechanischen Versuchen an Werkstoffen des schnellen Natrium-gekühlten Reaktors. Report BF-R-62.945-3 (April 1978)

10. Degree of Availability of the Reports

The reports are available through GRS-Forschungsbetreuung.

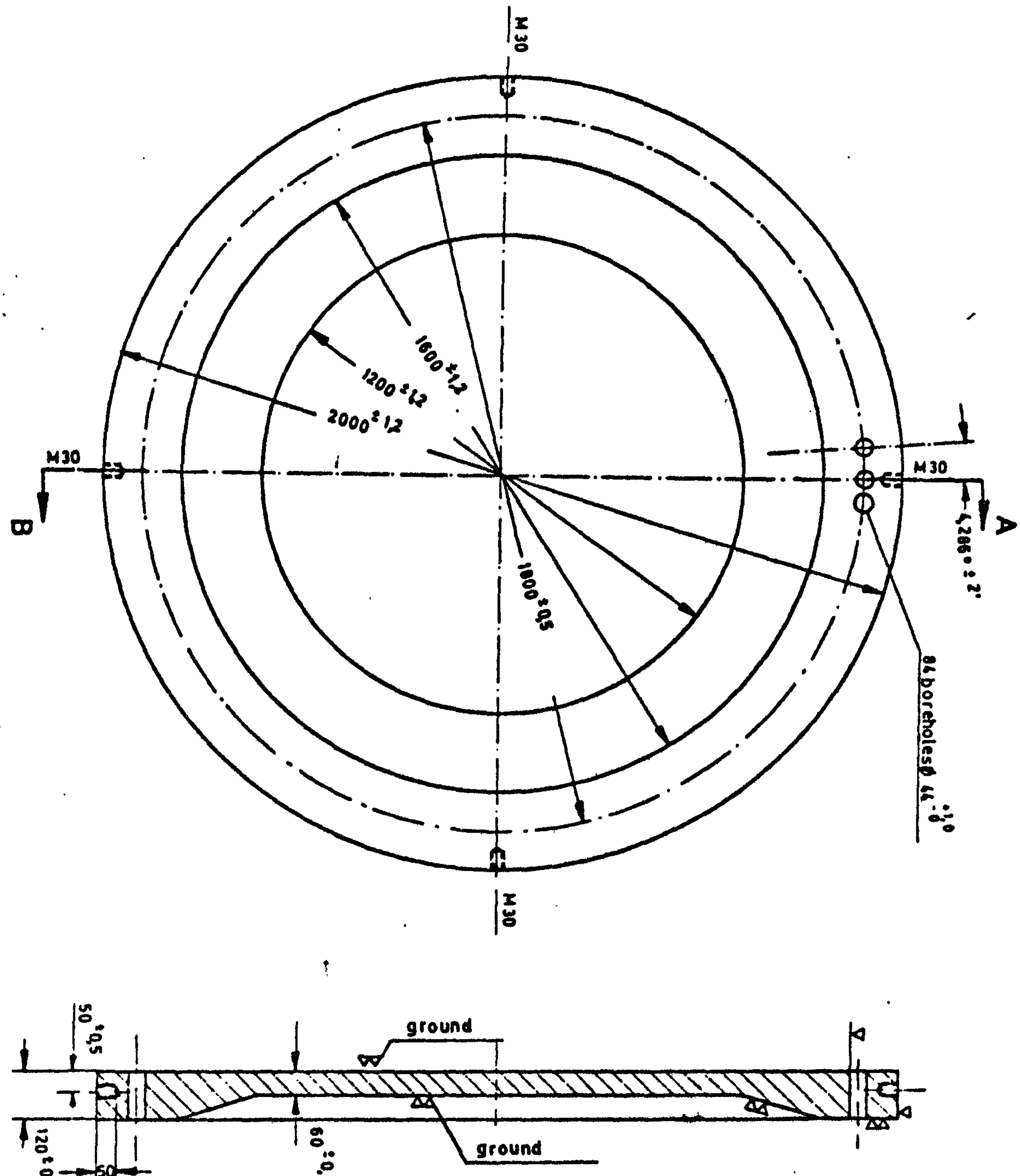

Fig. 2: Large plate (LP)

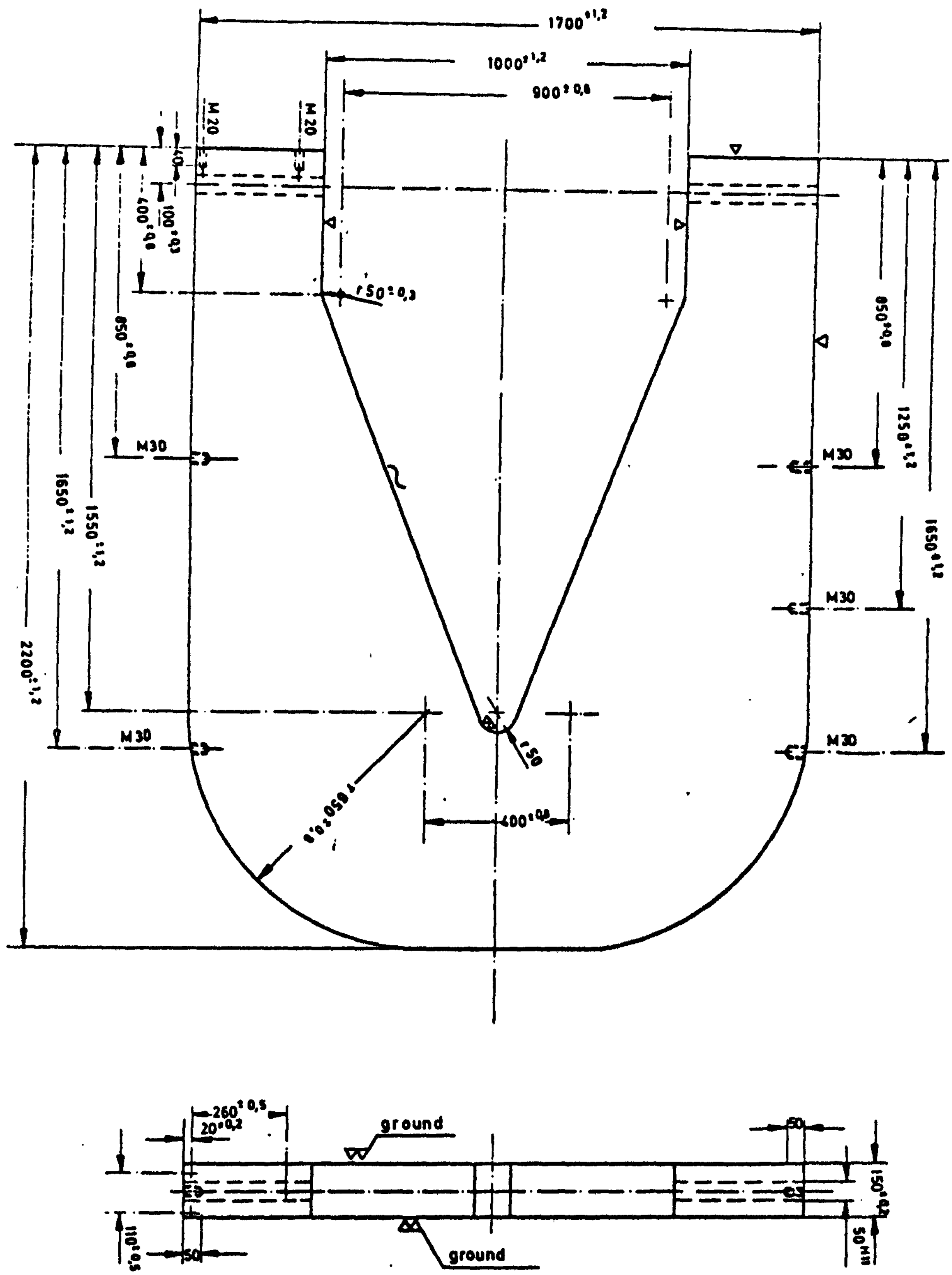

Fig. 1: Large-area fracture toughness specimen (LFTS)

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78 - 31.12.78	12.3	RS 196

Vorhaben/Project Title	
Verbesserte phänomenologische Beschreibung von Schallemissionssignalen und ihre Analyse auf eine bessere Fehlerbewertung Improved phenomenological description of acoustic emission signals and their analysis with respect to a better evaluation of defects	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: FraunhoferGesellsch. München IzfP Saarbrücken IfaM, IFKM

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.02.1976	30.04.1979	Dr. J. Lottermoser
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	31.12.78	

1. General Aim

- Development of an acoustic emission testing method for the application on pressure vessel tests on nuclear reactor power plants.
- Detection and evaluation of acoustic emission signals.

2. Particular Objectives

- Development of models for the evaluation of acoustic emission signals, firstly for laboratory experiments on CT specimens with fatigue crack and/or natural defects on ferritic and austenitic steels.
- Determination of the laws for progagation of acoustic emission signals in thick-walled work pieces (plates).

3. Research Program

- Measurements of acoustic emission signals on modified CT specimens.
- Signal analysis and determination of the correlation between signal parameters and the physical phenomena.
- Development and experimental verifications of models.
- Theoretical and experimental investigations on the propagation of acoustic emission signals in thick plates.

4. Experimental Facilities, Computer Codes

-

5. Progress to Date

5.1. Derivation of correlations between acoustic emission parameters with stress intensity factor K and crack growth (IzfP).

5.2. Measerements of directivity patterns of acoustic emission sources and comparison with models (IzfP).

5.3. Development of a new concept for acoustic emission testing with a local application (IzfP).

5.4. Welding tests for reproducably applying definit faults into welded seams (IfaM).

5.5. Experimental derivation of laws for the propagation of AE-signals in thick plates as well as theoretical investigations (IzfP).

6. Results

6.1. For tension-tests on 50 mm CT-specimens of the steel 22NiMo Cr 37 with different heat-treatments a roughly linear correlation

$$n_G \sim (\overline{\Delta a})^m \ , \ m \approx 1$$

was found; n_G is the number of high-energetic events ($\geq 10^{-12}$ Joule) and $\overline{\Delta a}$ is the corresponding crack propagation (Fig. 1). The value of $\overline{\Delta a}$ was measured on the crack-surface after the test, marking the crackfront by applying a fatigue load after stopping the tension-test.

For the same tension-tests the following correlation with the stress-intensity-factor K was found:

$$n = \overline{n}_1 \cdot (\sigma^2_{SE} \ K)^4 \ , \ \overline{n}_1 = (6,7 \pm 1,3) \ 10^{-33} mm^{22} N^{-12}$$

n = number of events from the beginning of the test (without friction noise).

σ_{SE} = acoustic-emission-yield point = stress-level, with the maximum of the event-rate in a tension-test on a specimen without notch.

This result is shown in Fig. 2 on the basis of 20 specimens with different heat-treatments ($400 = \sigma_{SE} = 750 \ N/mm^2$) (IzfP)

6.2. On modified CT-specimens the directivity patterns of real acoustic emission sources are measured (only longitudinal component). Fig. 3 shows the result for the first load cycles of fatigue (0^{o} correspnds to the crack-propagation-direction). The dotted line shows the results of calculations for a two fold force pair - corresponding to a dislocation loop. Fig. 4 shows the results for micro-crack-formation and macroscopic crack-growth. The dotted line shows the results of calculations for a simple force pair (cleavage). (IzfP)

6.3. Using the above mentioned correlation between the number of events and the stress intensity factor an AE-testing-method is proposed: If the locus of a defect is known from other ndt-methods or AE-triangulation an AE-transducer may be placed very near to this defect. In this case the low-amplitude events can be measured and there should be less problems with transfer-corrections. A mathematical discription of the problem is derived.

6.4. The task has been to produce definite welding faults within the welded seam. To do this reproducable, extensive preliminary tests had to be performed.
To produce slag-inclusions which were reproducable with regard to their size and their local situation, it proved to be necessary to cut a slot of definite dimensions into the weld metal and to fill it up with a suitable prepared piece of slag. Then the next layer has been applied into the welding joint giving a slag inclusion of definite size. This has been proved by nondestructive testing methods as ultra sonic and X-ray inspection as well as by cutting off the test pieces. The slag inclusion contains a volume of about 250 mm^3, as may be seen from a cross section, Fig. 5a, as well as from a section made into welding direction, Fig. 5b.
To produce faults of cold cracking type, the following technique has been chosen:
during welding process powder with different moisture content has been used to give hydrogen cracking within a selected layer. During the cooling phase a tension stress transvers to the welding direction has been applied for more than 72 hours The metallographic examinations following revealed the existence of cold cracks within the layer considered, Fig. 6.

To produce faults of the type hot crack the wire for the submerged arc welding has been coated by Cu of the whole wire weigth. The amount of Cu which is not solved within the crystals will migrate predominantly to the grain boun- deries and cause the initiation of hot cracks along these sites. In Fig. 7 a micro section through the submerged arc weldment taken in the welding direction and containing a hot crack may be seen. Using this technique discribed above during preliminary tests, hot crack could be found sporadic. Therefore, additional tests are planned to use S-containing filler material to produce a hihger amount of hot cracks within the weld material.

In additional tests we succeeded in producing faults of the type pores (IfaM).

6.5. The propagation of simulated AE-signals was measered on a 250 mm thick plate; Fig. 8 shows the measured energy of an event after travelling a path of X mm. The zick-zack-pro- pagation of the acoustic pulse is clearly seen and the mea- sured energies are following a law between the law for a free ware ($E \sim \frac{1}{x^2}$) and a cylindric wave ($E \sim \frac{1}{x}$). The length of the near-field (in the far-field we are observing guided waves with the law $E \sim \frac{1}{x}$) is depending on the frequency-spec- tra of the acoustic pulse: for an acoustic pulse of 0,5 μs duration it is approximately 15-times the plate thickness, for an acoustic pulse of 10 μs duration this value is approx. 6 times the plate thickness (IzfP).

7. Next Steps

7.1. Continuing the preliminary tests to produce other types of welding faults like lack of adhesion at the fusion line, stress relieve cracking within the heat affected zone and under plate cracking.

7.2. Continuing the welding and specimen machining.

7.3. Starting of AE measurement on specimens with natural defects.

7.4. Further investigations of the propagation of AE-signals in thick-walled components.

8. Relations with Other Projects

RS 191 A

9. References

1. J. Lottermoser, E. Waschkies, P. Zenner, B. Voss, J. Götz
   Laboruntersuchungen zur Erarbeitung von Interpretationsmodellen für die Bewertung von Fehlstellen bei der Schallemissionsprüfung am Kernreaktor, Bericht Nr. 780236-TW, der Fraunhofergesellschaft IzfP, Saarbrücken (1978).
2. T. Fischer
   Rißausbreitung und Schallemission in Schweißnähten des Stahls 22NiMoCr37, Bericht T 1 der Fraunhofergesellschaft, IFAM Bremen (1977).
3. K. Wolitz
   Rißausbreitung und Schallemission in Schweißnähten des Stahls x6CrNi1811, Bericht T 2 der Frauhofergesellschaft, IFAM Bremen (1978).

10. Degree of Availability of Reports

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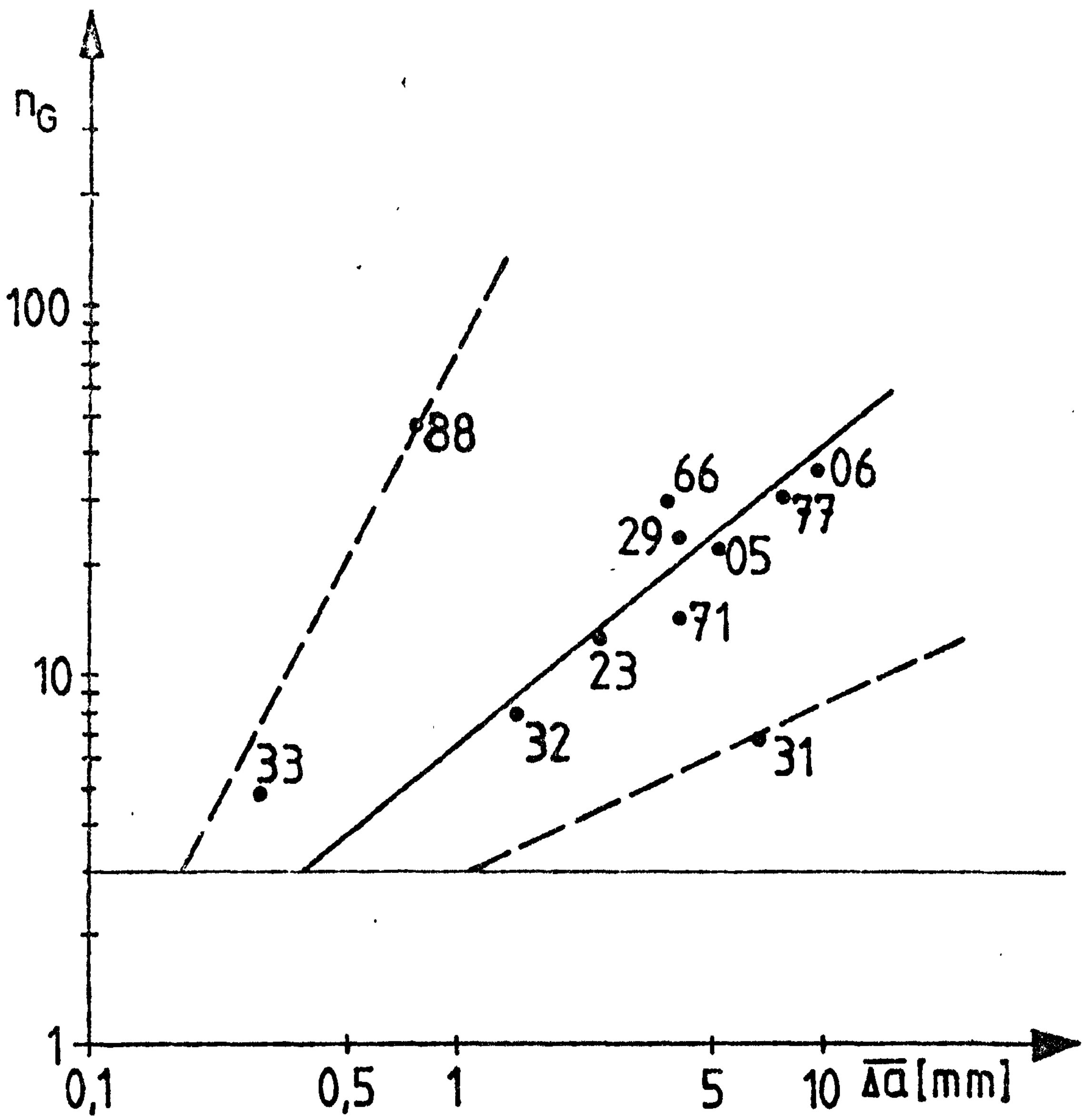
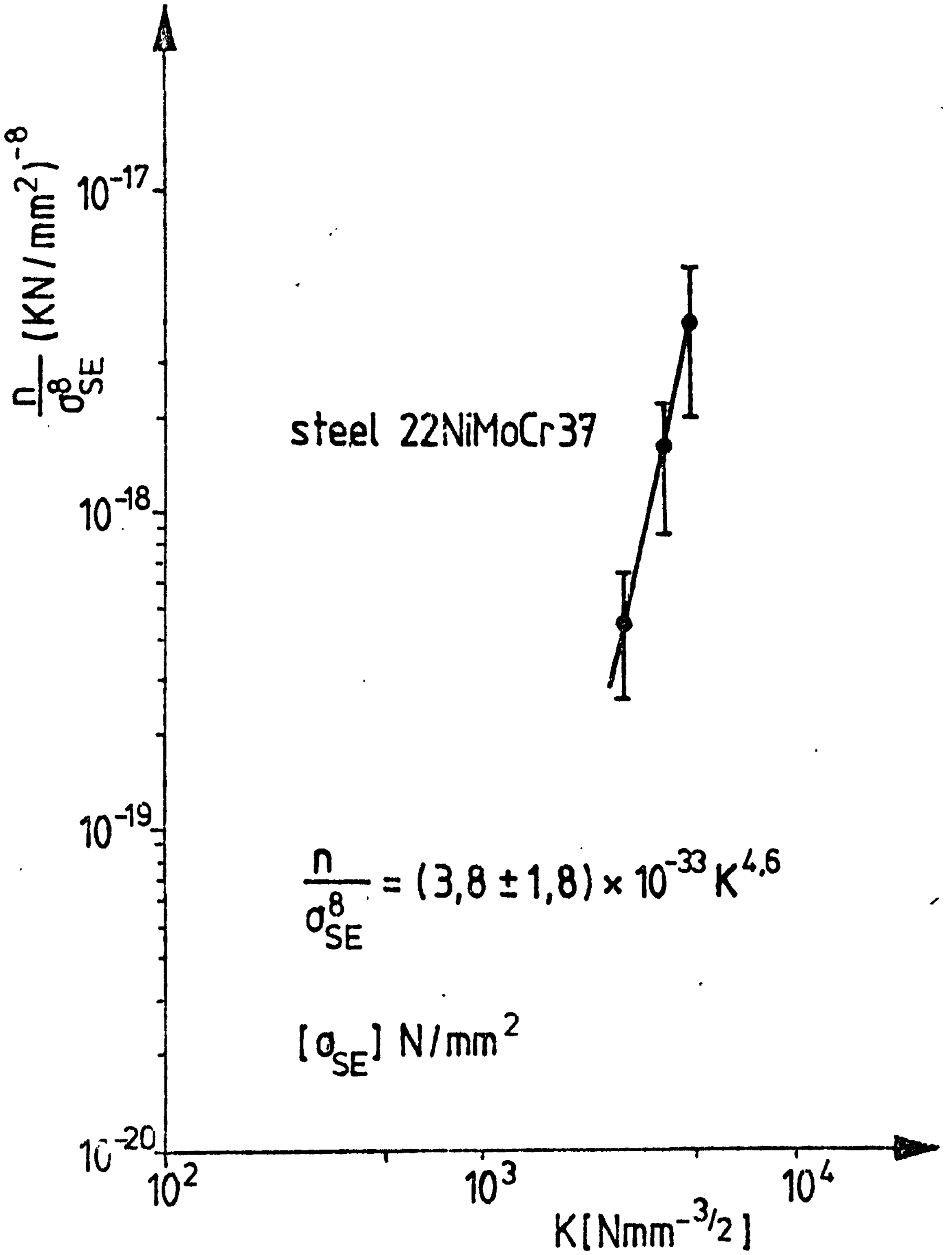


$n_G = f\,\overline{\Delta a}^{\,m}$

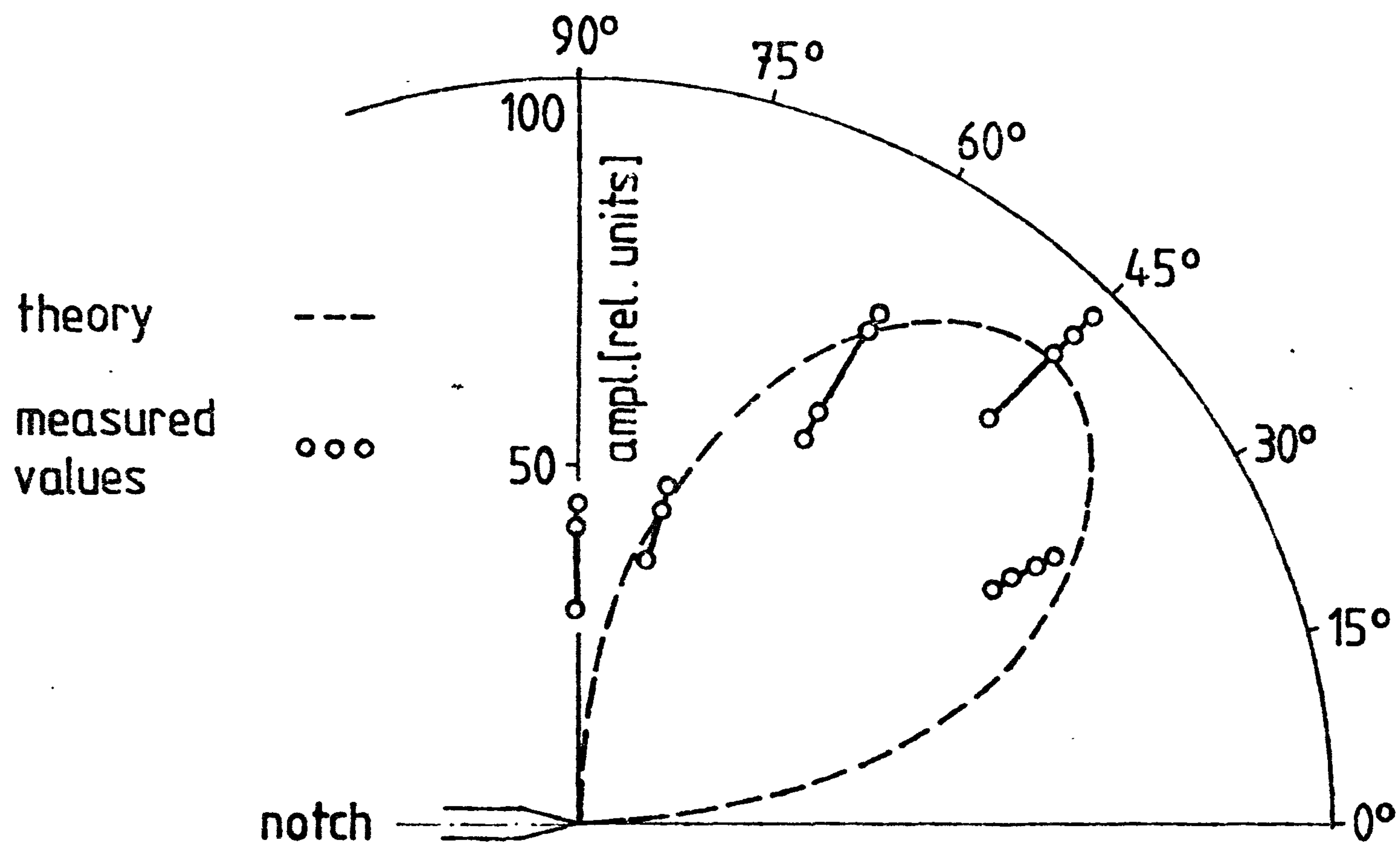
Number n_G of high-energetic events ($E > 10^{-12}$ Joule) as a function of the macroscopic crack growth $\overline{\Delta a}$ (50 mm CT-specimens)	IzfP
	Fig. 1



Number of events relative to the AE-yield point as a function of the stress intensity factor K

IzfP

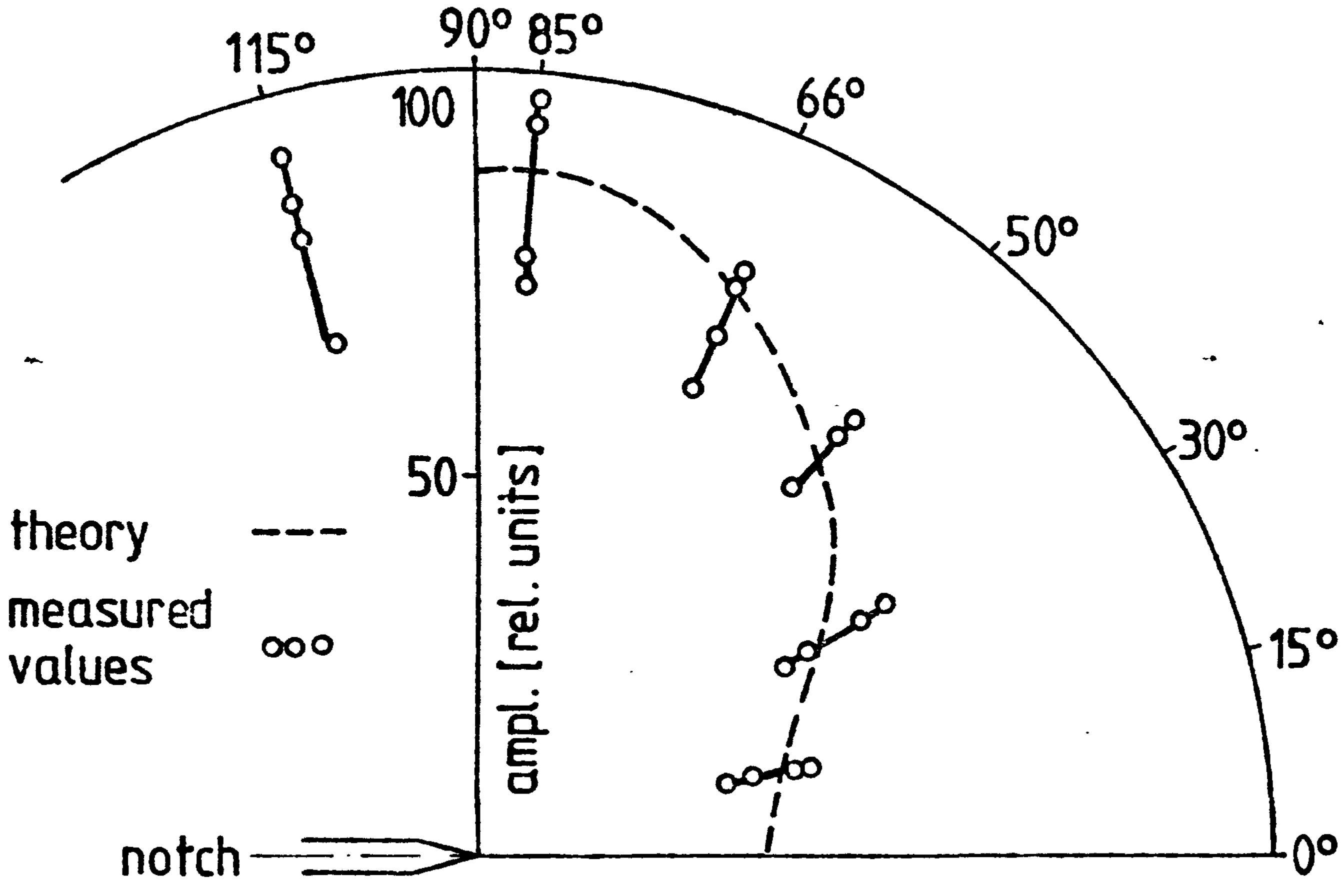
Fig. 2



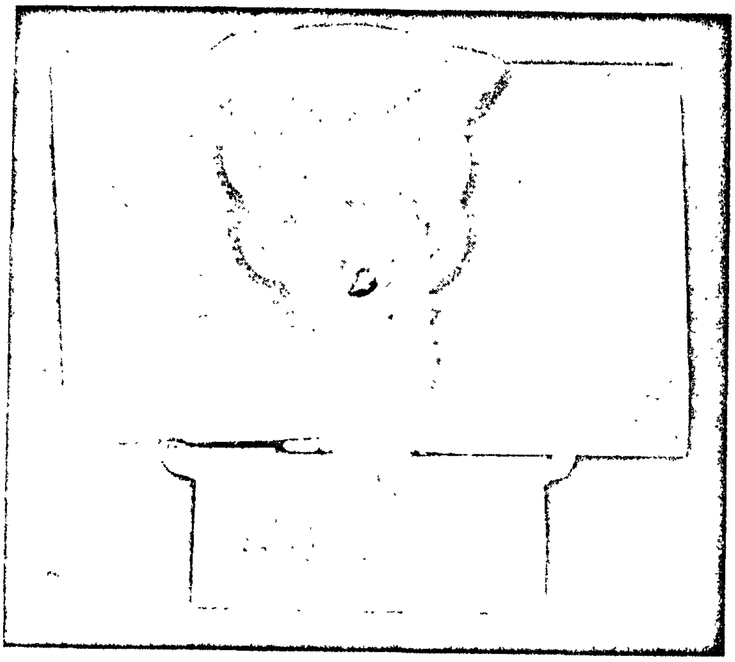
IzfP

Directivity-pattern for AE - source at the initiation of the fatigue crack

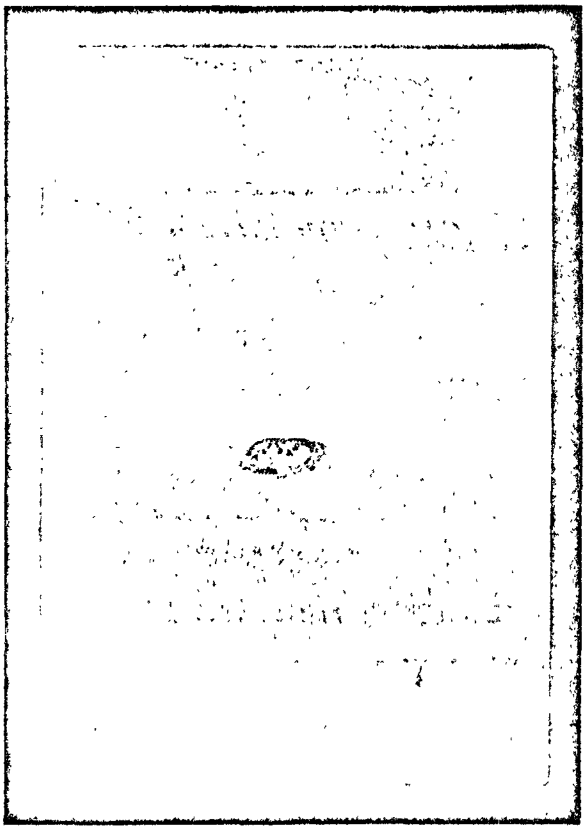
Fig. 3



IzfP | Directivity - pattern for AE - source of microcracking and macroscopic crack propagation | Fig. 4



a) Cross section
1 : 1



b) Longitudional section
1 : 1.25

Fig. 5

Slag inclusions within the submerged arc welded seam
2 % -HNO_3-etching (IFAM)

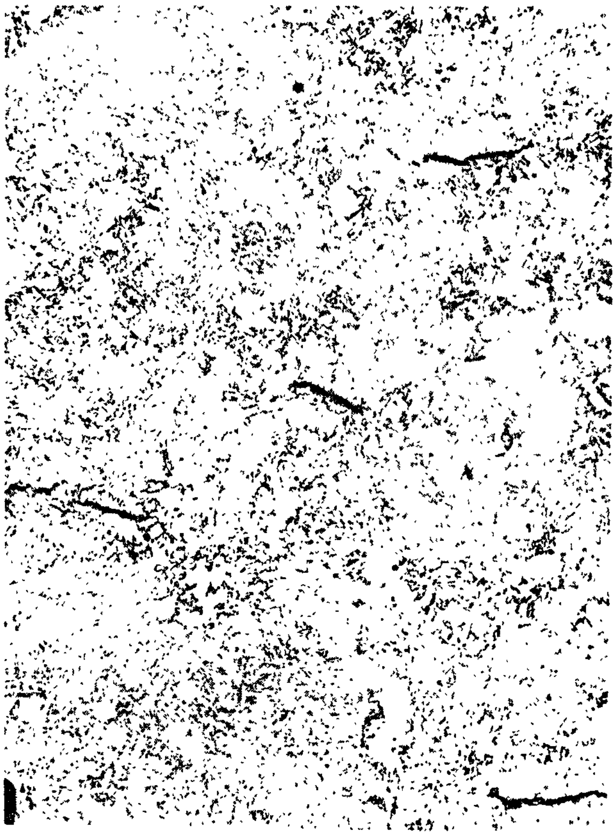


Fig. 6

Cold crack within the submerged arc welded seam

2 % HNO_3-etching 50 : 1 (IFAM)

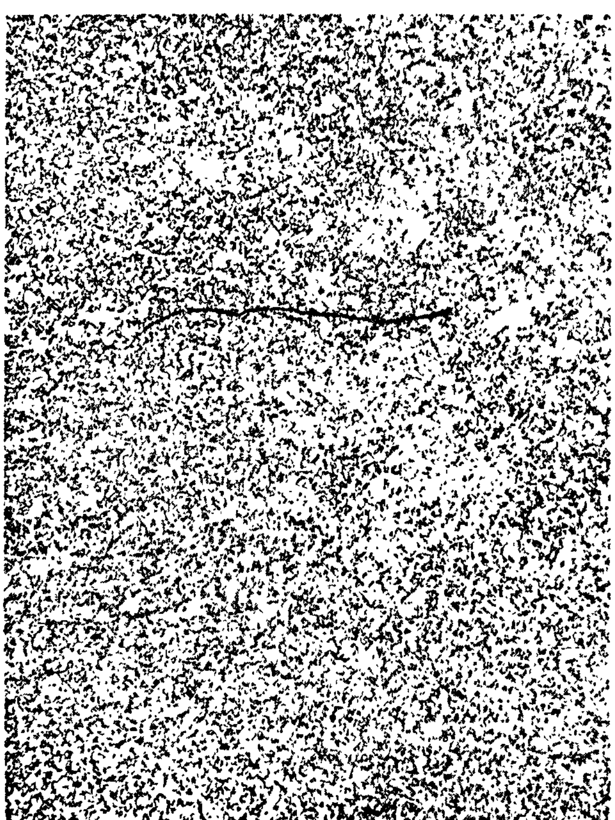
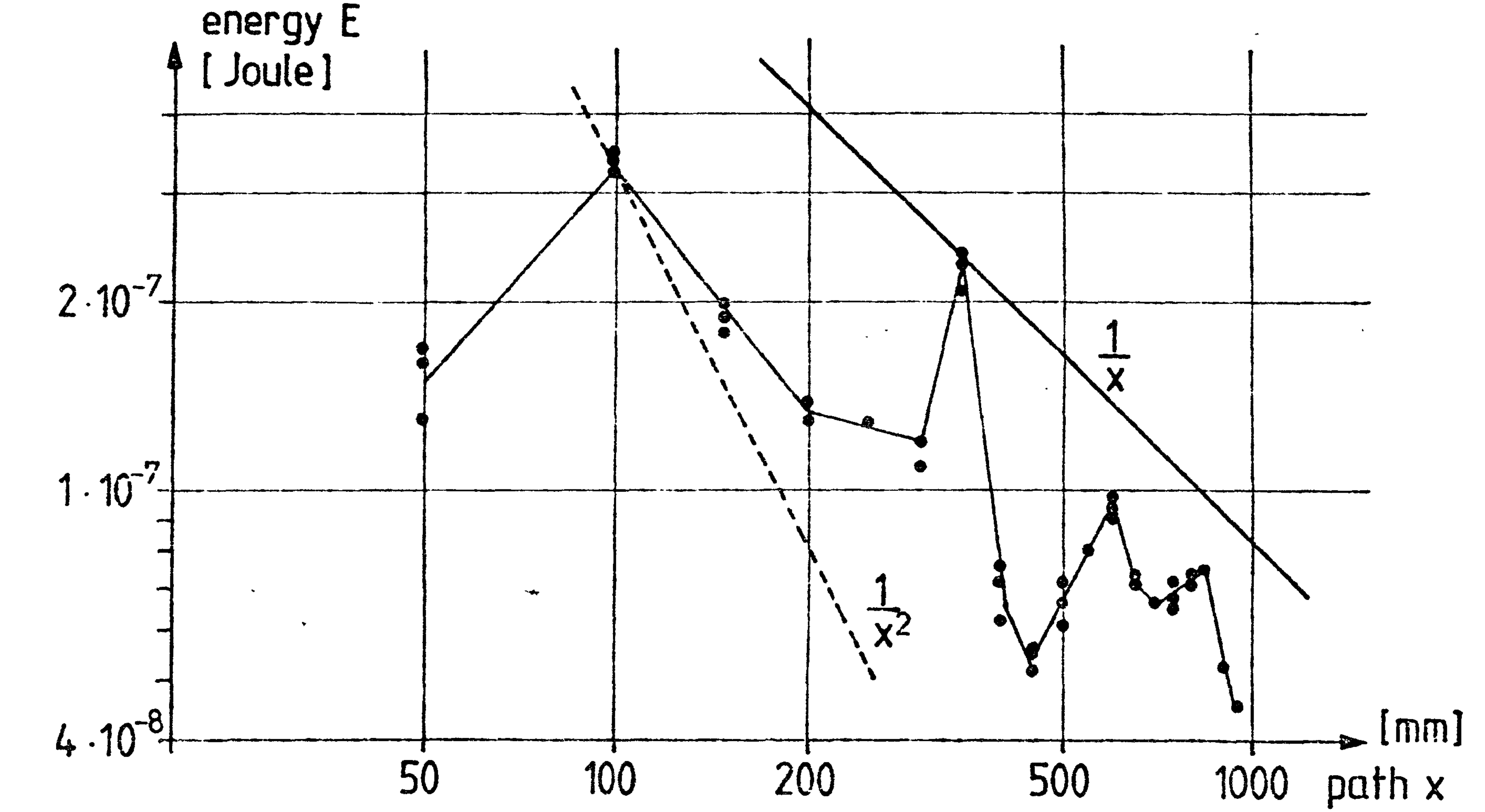


Fig. 7

Hot crack within the submerged arc welded seam

2 % HNO_3-etching 200 : 1 (IFAM)

RS 196



IzfP	Energy E of a signal as a function of the travelled path x	Fig. 8

01.01.78 - 31.12.78

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78 - 31.12.78	12.3	RS 322

Vorhaben/Project Title	
Development and application of signal averaging techniques for the ultrasonic testing of coarse grained austenitic materials and welded joints Signalmittelungsverfahren für die Ultraschallprüfung von grobkörnigen austenitischen Werkstoffen und Schweißnähten	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Fraunhofer-Gesellschaft, München IzfP-Saarbrücken

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.01.1978	31.12.1980	Dipl.-Ing. S. Kraus
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

The general aim is the improvement of the testability of coarse grained materials with high coherent noise level as for example austenitic steel and welds, ferritic and austenitic castings and claddings.

2. Particular objective

Techniques used for the improvement of the ultrasonic testability, are the signal averaging technique in form of the spatial-, frequency- and directional averaging, combined with conventional methods like focusing or testing with broad-band-pulses.

3. Research program

The research program consists of two essential parts, to be done in two phases. In the first one, within a short time, a device for the application of the spatial averaging is to be built up. This apparatus will be combined with a conventional UT-device. Whether the signal processing occurs in analog or digital manner was decided in preliminary tests. The essential aim of this first phase is the increase of the data processing rate to allow on-line testing with about 50 mm/sec testing velocity without loss in information content.
In the second stage of the project, an apparatus should be developped and tested for spatial-, frequency- and directional avera-

ging by means of phased arrays. Here, especially the developments, done in RS 249, are used.

4. Experimental facilities, computer codes

The experimental facility consists of an Ultrasonic-Flaw-Detector USIP 11 (Krautkrämer) and the digital averaging system, built in our electronic laboratory. Compared to the faster commercial averaging unit, this one will be 50 times faster; the performance time for adding 1024 A-Scans in the 8K by 18 bit memory at a sample rate of less than 0,05 μ s per adress is 0,4 sec. The apparatus is microcomputer controlled and suitable for practical application.

5. Progress to date

After preliminary tests concerning the kind of data processing (analog or digital) and a market research about averager, we could see that no fast enough apparatus is available. Therefore the signal averaging unit was outlined, built up and tested in our electronic laboratory.

6. Results

Our digital averaging system consists of a commercial UT-device, an ADC and the averaging unit. The ADC has a sample rate of 30 megacycles per second and 8 bit resolution. The on-line-averaging unit has a 8K by 18 bit memory and works with 20 megacycles per second and adress. The number of sweeps is selectable from 1 to 1024. All functions are microcomputer controlled.

7. Next steps

Complete testing of the signal averaging device.

8. Relations with other projects

Exchange of experience and specimen with RS 244, RS 249 and RS 273.

9. References

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10. Degree of availability of reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.07.78 - 31.12.78	12.3	RS 349
Vorhaben/Project Title Rekonstruktion aus Signalortskurven Reconstruction by signal locus curves		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Fraunhofer Gesellsch., München IzfP, Saarbrücken
Arbeitsbeginn/Initiated 01.07.1978	Arbeitsende/Completed 30.11.1980	Leiter des Vorhabens/Project Leader O.A. Barbian
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General aim

The aim of this project is the development, the demonstrations, and the test of an automatic system for ultrasonic testing and reconstruction of the defects, which

- appends to the former development and to the present technique of the testing methods and extensivly the probe systems, too;
- consider the present requirements of the fully automatic in-service inspection, but as not limited by these requirements;
- consider in addition the problem for the automation of the inspection during manufacturing
- uses for the reconstruction of defects the amplitudes and transit times respectively the corresponding locus curves and the mathematical models of diffraction;

2. Particular Objectives/3. Research Program

2.1 Completion of some aspects of the RS 102-17 project such as
   - extension of the two-dimensional to the three-dimensional evaluation
   - expansion of the mathematical models for reconstruction
   - improvement of the suppression of noise as during the data acquisition as during the data processing.

2.2 Examination of the possibilities for the application of new testing techniques, the development of which will be terminated in other RS-programs in the duration of this project.

2.3 Recognition of defects near the surface by considering the structure of both surfaces, for the outer and inner testing.

2.4 On line coloured monitoring of the results of each test method for the whole wall in the length of the probe system.

Off-line monitoring with suppression of perturbations.

4. Experimental Facilities, Computer Codes

For the development of the software-program the PDP 11 Computer system with peripheral devices is used. For the testing of the programs in situations simular the reality an automatic ultrasonic testing system with data acquisition is available, consisting of two manipulators, an ultrasonic-hard-ware and a data acquisition system.

5. Progress to Date/6. Results

Theoretical investigations concerning the requirements for the accuracy of the transit-time measurements as for the two-dimensional as for the three-dimensional cause were conducted.
We have found that the accuracy of the recontruction depends on the following parameters

- accuracy of the transit-time mearurement of the ultrasonic hard-ware
- depth position of the defect
- distance of the correlated probe positions.

The last mentioned parameter can be adjusted that with maximal inaccuracy of the transit-time mearsurement of $\pm \lambda/2$ sufficient reconstruction can be reached for each depth position. Software components like data sorting, fault-clearing and monitoring of partial results were accomplished and tested by data of the automatic inspection of the pressure vessel test specimen.

7. Next Steps

Improvement of the fault-clearing- and reconstruction-programs. Other tests of the existing programs with the data of the inspection of the pressure vessel test specimen. Specification for the improvement of the ultrasonic hardware.
Development of a software program for the calculation of transit-time-locus-curves of known reflectors, such as geometrical variations.

8. Relations with Other Projects

This program is partly based on the results of the reactor safety research projects RS 102-17, RS 27-1, 2, 3 and will be carried through in coordination with the still running projects RS 27-4, RS 169, RS 249, RS 231 and RS 257.

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78 - 31.12.78	12.3	RS 323

Vorhaben/Project Title	Land/Country
Magnetic Particle Pseudo-Crack Indications Quantitative Interpretation of Magnetic Leakage Flux Testing	FRG
	Fordernde Institution/Sponsor: BMFT
Magnetpulverscheinrißanzeigen Quantisierung der magnetischen Streufluß-prüfung	Auftragnehmer/Contractor: Fraunhofer-Gesellschaft München; IzfP, Saarbrücken; MPA, Stuttgart

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.01.78	31.12.80	Dobmann
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31.12.78	

1. General Aim

Basic investigations about the reasons of pseudo-crack indications, their interpretation and discretization from real surface-cracks.

2. Particular Objective

Investigations to:
- metalurgical and physical reasons
- description of the leakage flux topography
- criticality of pseudo-crack areas
- possibilities for acting in practice for recognization and interpretation.

3. Research Program

3.1 Mechanical-technological Investigations
- Materials selection: TSB 370, StE 36, StE 51, 20 MnMoNi 55, 22 NiMoCr 37, Welmonil
- Welding-Simulation: between 700° - 1.300° C
- Welding-Joints: UP-Welding, electr.-hand-welding, electr.-beam-welding, with and without preheating, heat input energy between 15 KJ/cm - 35 KJ/cm
- Annealing: stress relieve, normalization
- Metalurgical Investigations: polishments, hardness, chemical analysis, micro hardness, REM, STEM,
- Fracture- and Fatigue Measurements:
- Stress-Measurements

3.2 Physical Investigations

- Variation of the Influence-Parameters: field-amplitude, static-alternating fields, premagnetization, powder concentration.
- Leakage-flux-Measurements in Vector-Components: magneto-probing, magnetography.
- Physical Materials Properties: Conductivity, permeabilty, koercive-force, austenit-content.
- US-Tests, Liquid Penetrant Tests

4. Experimental Facilities, Computer Codes

to 3.1 construction of a joke magnetization and magnetography-unit

to 3.2 Use of a computer program for computing the leakage-flux-topography

5. Progress to Date

to 3.1 Mechanical Technogical Investigations

Welding and Simulation of the TSB-370 specimen is finished

to 3.2 Physical Investigations

- the construction of the magnetization-unit with probe-scanning instrumentation is ready
- the magnetography-unit is realized in hardware electronic, for data handling a μ-processor is used,
- first numerical investigations for construction and calibration of the eddy-current-micro-probe for conductivity and permeability measurements are made.
- the leakage flux theory with alternating field-excitation is formulated.

6. Results

to 3.1 until to 3.2 A prexamination of a simulated specimen shows pseudo cracks with a structure gradient.

7. Next Steps

to 3.1 until to 3.2 First all investigations with TSB 370 specimens.

8. Relation with Other Projects

finished: RS 102 - 18

continued: RS 334

9. References

none

10. Availability of the Reports

none

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.04.78 - 31.12.78	12.3	RS 334

Vorhaben/Project Title	
Bestimmung der Mikrogefüge von Druckbehälterstählen mit magnetisch induzierten Meßgrößen Determination of the microstructure from pressure vessel steels with magnetic induced measuring quantities	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Fraunhofer-Gesellsch. München, IzfP Saarbr. MPA-Stuttgart

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.04.78	31.03.82	Dr. Theiner (IzfP)
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim
   - Production of different micro-structures by heat treatments, tinsile test, time fracture test. Material: Different reactor pressure vessel steels.
   - Description of the micro structure by mechanic-technological-, metallographical- and electron microscopical investigations and by nd methods as Barkhausen noise measurements and macroscopic magnectic quantities to be measured.
   - Separation of the different micro structures (grain size, precipitations, discocations) by nd quantities to be measured.

2. Particular Objectives
   - 2.1 Supply of materials (22NiMoCr37, 20MnMoNi55, STE51, TSB370)
   - 2.2 Fabrication of the whole sample set.
   - 2.3 Mechanic-technological, metallographical and electron microscopical investigations of the states - as delivered, cycle 13, cycle 13 + tempering (22NiMoCr37, 20MnMoNi55)
   - 2.4 Choose of model materials with non-ferromagnetic precipitations and production of sample set.
   - 2.5 Preliminary experiments and supply of experimental set ups for reversible and irreversible magnetization processes.

3. Research Program
   - 3.1 Examination of material homogeneity.
   - 3.2 Production of the sample sets. (removel, heat treatments)
   - 3.3 Investigation of the steel states (22NiMoCr37) as delivered + cycle 13+tempering,
     - fracture mechanic quantities; a_k-T curves; hardness HV 10;

light microscopic determination of structure; electron-microscopic investigation (extraction, thin films) only 22NiMoCr37.

3.4 Theoretical investigation of the actual knowledge of micromagnetisme and of metallographical discription of steel properities.

3.5 Experiments to the generation of acoustical Barkhausen events, development of nd quantities to be measured. Variable: Microstructure, stress state.

4. Experimental Facilities, Computer Codes

ad. 3.1 Experimental set up for homogeneity examination (Klöckner-Werke, Thyssen AG)

ad. 3.2 3.3 MPA Stuttgart: Facility for sampling, sample machining, heat treatment; mechanical-technological-, metallographic and electron microscopic examinations. MPI Düsseldorf: Facilities for TTT diagrams and isothermal continual cooling. University Saarbrücken/IzfP: facilities for decarbonization, heat treatment, metallographic + electron microscopy (→ model material).

ad. 3.5 3.6 IzfP: Experimental set up for magnetic and acoustic Barkhausen noise measurements; hystrometer for rod shaped samples (Ø 5, Ø 8 mm); coercimeter; magnetostriction; e.m.a. exitation; US scattering; thermopower; electrical conductivity on rods.

5. Process to date

ad. 3.1 Supply of material (22NiMoCr37, 20MnMoNi55). Examination of homogeneity (22NiMoCr37, 20MnMoNi55) and texture with X-Ray + Barkhausen quantities (22NiMoCr37).

ad. 3.2 Sampling of the forged piece 22NiMoCr37 - as delivered; as delivered + tensile test; cycle 13 + tempering 610°C; cycle 13 + time fracture test. Production of model material Fe-Cu (0,5-; 0,8-; 1,3-; 1,8- vol. % Cu; C < 10 ppm + tempering).

ad. 3.3 Investigation sample set 22NiMoCr37 - as delivered; cycle 13 + tempering 610°C; cycle 13 + time fracture test 610°C.

ad. 3.4 Choose of Fe-Cu, Fe-Au as precipitation hardening model material

ad. 3.5 Proof of the origin of acoustical Barkhausen events. Preliminary experiments on three different states of 22NiMoCr37

+ 3.6 and supply of experimental set ups.

6. Results

ad. 3.1 Little scattering of chemical composition; of fracture mecha-nic quantities and of grain size over the cross section. Texture negligible small.

ad. 3.2 As delivered: Fine grain micro structure (ASTM 8) with veined micro segregations parallel to the main deformation direction.

+ 3.3 Hardness HV 10 = 205. Globulare cementite precipitations (~250 nm); needle shaped Mo_2C precipitations (100-200 nm long, 10 nm thick).

Cycle 13: Coarse grain structure (ASTM 1-2), martensile-bainite hardness HV 10 = 400 - 420. Fine rod shaped $Fe_{2-3}C$ precipitations (100-200 nm long, 15-30 nm thick). Widmannstätten sturcture. Grain boundary without precipitations.

Cycle 13 + tempering 610°C: Hardness decrease with termpering time (150h - HV 10 = 250). After one hour there exists a plateau in hardness (secondary-precipitations). Transformation $Fe_{2-3}C \rightarrow Fe_3C$. Simultanus moulding and production of globulare Fe_3C and needle shaped Mo_2C precipitations inside the grains. Globulare Fe_3C precipitations on the grain bounderies. During tempering time the size and number of Mo_2C needles increase.

ad. 3.5 The three states - as delivered; cycle 13; cycle 13 + 610°C could be distinguished in the Barkhausen noise as well as in

+ 3.6 the macroscopic mag. quantities. The electrical conductinity and the thermopower showed different values for cycle 13 and the two other states.

The amount of the magnetostriction is great enough (22NiMoCr37) for developping quantities to be measured.

Patent: Verfahren zur zerstörungsfreien Feststellung von Werkstoffzuständen unter Ausnutzung des Barkhausen-Effektes.
30.08.1978 / P 2837 733.5

7. Next Steps

ad. 3.1 Supply of material STE51, TSB370

ad. 3.2 Fabrication of the rest sample set 22NiMoCr37 ad 3.3 Electron microscopic investigations (22NiMoCr37 + 20MnMo Ni55)

ad 3.3 Supply of image evaluation equipement; ouput units; el. mag. receivers; power supply + frequency analyser.

+ 3.6 Examination of the new material state ndt quantities.

8. Relations to other projects

RS 323; RS 102-16; RS 102-18

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.07.78 - 31.12.78	12.3	RS 332
Vorhaben/Project Title Echtzeit-Schallemissions-Prüfsystem Real-Time Acoustic Emission Triangulation System		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Fraunhofer-Gesellsch., München IzfP, Saarbrücken
Arbeitsbeginn/Initiated 01.07.1978	Arbeitsende/Completed 30.06.1980	Leiter des Vorhabens/Project Leader Dr. J. Lottermoser
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 31.12.78	Bewilligte Mittel/Funds

1. General aim

Development of a triangulation system for locating acoustic emission sources (ESEP-system) which can handle the events parallel compared with conventional systems which can process data only sequential. An ESEP-system should be more suitable than sequential processing systems for locating crack growth during hydrostatic test of a reactor vessel. Conclusions from RS 191 and RS 196 have to be considered while working out an ESEP-system.

2. Particular objectives

In phase I the ESEP-system will be simulated on a computer and the results (from simulated and available real data) will be compared with those from sequential working locating systems.
It is the aim of this project to demonstrate the capabilities and limits of the ESEP-system. During the second phase of the project a hardware ESEP-system will be realized for one transducer array so that the system can be tested under realistic conditions.

3. Research program

3.1 Working up a concept for the ESEP-system
3.2 Working up the mathematics for the triangulation on any spherical or plane surface
3.3 Development of a computer program for the software simulation of an ESEP-system
3.4 Test of the system with simulated and real AE-data
3.5 Outlining the capabilities and limits of the sytem
3.6 In phase II the system will be realized in hardware. A complete list of the works to perform will be given after the successful

termination of the first phase.

4. Experimental Facilities, Computer Codes

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5. Progress to date

   5.1 Working up the mathematics for the allocation of a location from two time differences on any spherical or plane surface.

   5.2 We have started the programming

6. Results

   6.1 The shape of the location areas for different shift register lengths was compared. The optimum shape was found to result from an even number of storage locations in each shift register.

   6.2 A concept for the storage of the location coordinates which permits to reduce the necessary size of the storage area for the locating as far as possible was developed. The programming work has not finished yet.

7. Next steps

   Continuation and termination of phase I

8. Relations with other projects

   The KWU (RS 339) makes analog tapes from hydrostatic test of reactor vessels available. These data do not include any critical action like crack growth.

   In RS 344 (Battelle) test measurements were performed. The data shall be evaluated with a sequential working system.

   The experiences from RS 31 with a serial working system and the results concerning signalanalysis and estimation of AE signals from RS 191/RS 196 will be taken into account.

9. References

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10. Degree of availability of reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1. 1. 78 - 31. 12. 78	12.3	RS 339

Vorhaben/Project Title	
Untersuchungen zur Aussagefähigkeit eines neuen Schallemissions-Prüfsystem-Konzeptes (ESEP) für die Ortung von Fehlern in Reaktorkomponenten Investigations concerning the information value of a new concept of acoustic emission test system (ESEP) for the location of flaws in reactor components	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: KRAFTWERK UNION AG Reaktortechnik R 214, Erlangen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1. 5. 78	30. 6. 79	Dr. Votava
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	31. 12. 78	

1. General Aim

A new acoustic emission test system (ESEP) for location of flaws in reactor components shall be developed. This system is meant to provide a possibility to cover a larger number of acoustic emission (AE) signals. For this purpose a detailed work proposal was submitted by IzfP (Saarbrücken).

2. Particular Objectives

The advantages and disadvantages of the ESEP system, as compared with the location system used previously, will be determined partly by means of computer simulation of AE signals which both have been obtained in the course of AE measurements on reactor pressure vessels (carried out by order of KWU), partly by simulation of signals on a reactor pressure vessel. The results of this project shall serve as a basis for making a decision on the construction of a prototype system.

3. Research Program

3.1 Providing analog tapes of AE measurements carried out on reactor pressure vessels. Part of the AE signals stored are due to flaws, the location and nature of which are known in part (surface cracks, brittle surface layers, fretting). These tapes are copied, as required.

3.2 Providing a reactor pressure vessel for simulation tests of signals in a plant hydrostatic test.

In 1978/79 KWU carries out a hydrostatic test. The following extreme cases are scheduled:

- a discrete sequence of high amplitude signals, as they occur in connection with macroscopic crack propagation;

- a high rate of low amplitude signals, as they occur in the case of plastic deformation;

- a sequence of partly overlapping signals of varying amplitudes from one or two AE-sources;

- signal injection from inside with and without water in the reactor pressure vessel.

3.3 Including simulation tests in the time schedule of a reactor pressure vessel and plant-specific planning, arranging for installation work and its execution, preparing the measuring points.

3.4 Reporting on project as a whole and final conclusions in cooperation with the participating partners.

4. Experimental Facilities

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5. Progress to Date

To 3.2 After consultation with the partners, the Battelle Institute (Frankfurt/Main) and IzfP (Saarbrücken), a measuring program was prepared with the aim to carry out this program on the Grohnde reactor pressure vessel. The plans were to fix four receiving probes (type S 140 B) on the outside surface of the cylindrical part in the corners and in the centre of gravity of a triangle (1 locating area) and to generate test signals in various places. Variations are to be made as to the number of transmitters, the height and type of pulse sequence, the location of the transmitter, the transmitter power, and inside and outside sound injection. Two questions were given particular emphasis in connection with this program:

a) Which transmitter is best in simulating an AE source?
b) Does water in reactor pressure vessels affect the signal transfer on account of radiation into the water

On a)
Battelle (Frankfurt) conducted extensive preliminary tests concerning this question. They used:
- the breaking of pencil lead
- transmitting with S 140 B
- transmitting with S 750 B
transmitting with S 750 B and cone
- transmitting with angle beam search unit
- transmitting with lithium sulphate probe
- the breaking of glass
- mechanical twinning in indium.

It has been found that signal injection with a cone badly reproduces an AE source. Any other procedure is better suited for this purpose.

On b)
It has been decided to conduct a test series with a reactor pressure vessel filled with water. Further planning will have to be done in this respect as the Grohnde reactor pressure vessel is at present on roller bracket.

6. Results

To 3.2 Between September 4 and 13, 1978, the scheduled tests were performed without any water filling on the Grohnde reactor pressure vessel at Gutehoffnungshütte Oberhausen. The first result is that a signal from inside is clearly much longer and more structured than one generated on the outside surface.

7. Next Steps

To 3.2 Analysis of test measurements on the Grohnde reactor pressure vessel and planning for the purpose of measurements on a reactor pressure vessel filled with water.

8. Relation with Other Projects

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9. Literature

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10. Degree of Availability

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.6.78-31.12.78	12.3	RS 344

Vorhaben/Project Title	
Investigations into the Informative Value of a Novel AE-Based Method (ESEP) for Locating Defects in Reactor Components Untersuchungen zur Aussagefähigkeit eines neuen Schallemissions-Prüfsystems (ESEP) für die Ortung von Fehlern in Reaktorkomponenten	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Battelle-Institut e.V. Abt. WerkstoffVerhalten

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.6.78	31.3.79	Dr. P. Jax
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

The general aim of the project is to develop a new acoustic emission measurement system (ESEP) for the location of faults in reactor components. With this system it should be possible to record high-rate acoustic signals as they occur, e.g., during the burst experiments performed within the superheated steam reactor and component safety program.

2. Particular Objectives

The particular objective is to determine, by computer simulation, the advantages and drawbacks of the ESEP system compared with conventional location systems. Signals emitted during burst experiments and simulated signals will be recorded for this.

3. Research Program

3.1 Generation of simulated signals in a reactor pressure vessel under variation of the place of signal generation, the signal rate, and the amplitude statistics. Determination of the reception times of the simulated signals and the signals recorded during the burst experiments and subsequent digitization of these time values. Transfer of the digital data carriers to the IzfP, Saarbrücken (see below).

3.2 Location of the signals described in 3.1 with the existing method.

3.3 Comparison of the results obtained according to 3.2 and with the ESEP method, and determination of the merits and limits of the ESEP system compared with the conventional fault locating system.

Step 3.3 will be carried out in cooperation with the agencies listed in 8.

4. Experimental Facilities, Computer Codes

Re 3.1

The simulated signals will be produced with an emitter or with reproducible signals sources (e.g. breaking pencil lead) and stored on analog tape. To be able to determine the reception times and to digitize these values a measuring unit is required which will record the beginning of the individual acoustic groups within a signal and transmit these values to a computer. The end of a signal group is characterized by a delay time (20 - 100 μs) during which no exceeding of amplitudes is observed on a preset discriminator. A computer code will be develop with which the time values of 4 measuring channels will be classified and transferred to digital tape.

5. Progress to Date

Re 3.1

Signals have been simulated in an empty reactor pressure vessel and recorded on analog tape. Signals were generated inside and outside, using single signals and sequences of signals (periodic and random rates, constant amplitudes,preselection of an amplitude distribution) A measuring unit for recording and digitizing the signal reception times has been designed and is largely completed.

Comparison with the research program: the various steps will all be continued according to schedule.

6. Results

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7. Next Steps

Re 3.1

Digitization of the reception times of the simulated signals and the signals emitted from faults.

Re 3.2

Location of faults with the existing system

8. Relation with Other Projects

The project is carried out in close cooperation with the IzfP, Saarbrücken (RS 332) or Nukem (subcontract from IzfP), and the KWU, Erlangen (RS 339). Ap"t from cooperating in step 3.3, the KWU and the IzfP are respon ible for the following work:

- Provision and preparation of suitable pressure vessels in which the simulated signals can be produced (KWU);
- Provison of magnetic tapes on which signals from defects have been recorded during AE mesurements on pressure vessels (KWU)
- Location of signals with the new ESEP method, using computer simulation (IzfP und Nukem)

If the ESEP system is assessed favorably a prototype will be constructed (IzfP and Nukem).

9. References

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10. Degree of Availability of the Reports

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13. SYSTEMS OPTIMASION, STANDARDISATION, NEW CONCEPTS

	CLASSIFICATION: 13
TITLE (ORIGINAL LANGUAGE): Développement des Cuves Intégrales pour Réacteurs Nucléaires PWR.	COUNTRY: Belgium
	SPONSOR: Ministry of Economic Affairs
TITLE (ENGLISH LANGUAGE): Integral Vessel Development for PWR Nuclear Reactors.	ORGANISATION: S.A. COCKERILL.
	PROJECT LEADER: J. WIDART
INITIATED: 1974. / COMPLETED: 1983	SCIENTISTS: Alain SCAILTEUR, Philippe LAURENT, Raymond HENIN, J.Paul COLLETTE
STATUS: In progress / LAST UPDATING: May 1979	

GENERAL AIM : To supply, mainly to PWR (Pressurized Water Reactors), reactor vessel having numerous advantages at the manufacturing, inspection (during fabrication and mainly in service inspection) and mechanical behaviour points of view. This is obtain by conceiving the geometry of a large forging (500 T ingot) that includes the vessel flange, the nozzle shell course, the major part of the nozzles and of the support pads.

PARTICULAR OBJECTIVES : Development of the design by :

2.1 Conceiving a new geometry of the PWR vessel,
2.2 Optimizing this new geometry by 3D (three dimensional FEM (finite element method) calculations for pressure, thermal and external loads conditions.
2.3 Discussing the new design with other Manufacturers and Safety Authorities to maximize the Integral Design advantages.
2.4 Studying adequate means and schedule for manufacturing.
2.5 Checking calculational results by instrumented hydrotest.

EXPERIMENTAL FACILITIES AND PROGRAMME : As the vessel is a static equipment and as the integral design induces lower stress levels in the vessel than those existing in a conventional vessel, ans as Cockerill has to supply two Integral Vessels in 1980 and 1981 respectively, no exprimental facility has been foreseen.

The programme includes :

3.1 Design development of Integral RPV (Reactor Pressure Vessel) for Doel 4 and Tihange 3 three loop nuclear power stations (1000 MWE)
3.2 Design development of an Integral three loop RPV with support pads obtained without any weld (900 and 1000 MWE).
3.3 Design development of an Integral four loop RPV with support pads obtained without any weld (1200-1300 MWE).
3.4 Development of Integral designs for RPV of small dimensions (100-150 MWE).

PROJECT STATUS : 4.1 Integral RPV has been extensively developped at the metallurgical, stress analysis, manufacturing and in service inspection points of view.
Main materials for Doel 4 and Tihange 3 RPV have been supplied without any problem, those vessels are in manufacturing sequences.

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4.2 An Integral Vessel is currently examined for a 125 MWE project.

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) NEXT STEP : Check of the computional results by instrumented hydrotest of the three loop RPV for Doel 4 and Tihange 3.

) RELATION TO OTHER PROJECTS AND CODES : RPV are designed according to the Code ASME Sections 2, 3, 5, 9 and 11.

) REFERENCE DOCUMENTS :

(1) Cambien R.B; New Design of PWR Reactor Vessel Using Large Forging, ASME Conference, Mexico City, September 1976.

(2) Reynen J, De Windt P, Widart J, et al; A Novel Design for LWR Pressure Vessel Nozzles and Corresponding Stress Analysis, 3rd International Conference on Pressure Vessel Technology at Tokyo, April 1977.

(3) Widart J; Design of Reactor Pressure Vessel Considering Easier I.S.I, I.A.E.A Technical Committee, Kobe (Japan) April 1977.

(4) Onodera S, Nagata M, Tsukada H; Large Size, Integrated Tr Steel Forgings As Intended for Easier I.S.I, IAEA Technic Committee, Kobe (Japan), April 1977.

(5) ASME Boiler and Pressure Vessel Code, Sect.XI, 1977 Editi

14. PROBABILISTIC METHODS OF SAFETY ANALYSIS

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78-31.12.78	14	RS-189

Vorhaben/Project Title	
Rechenprogramm für Zuverlässigkeit und Verfügbarkeit von Gesamtkernkraftanlagen mit Hilfe der Zustandsanalyse Computer Program for Reliability and Availability of Nuclear Power Plants using the Method of State Analysis	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: INTERATOM 5060 Berg. Gladbach 1 Abt. Zuverlässigkeit und Strukturdynamik

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.01.76	31.12.79	Dr. Zeibig
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1 General Aim

Development of quantitative methods for the assessment of safety, availability and risk of nuclear power plants.

2 Particular Objectives

It is planned to use the method of state analysis as an aid in estimating the risk of nuclear power plants. For this purpose the relevant states of the plant have to be found. Possible transitions between these states have to be identified. The mathematical methods for calculating the state probabilities have to be developed. The chief objects of the project are:

- a computer program for this purpose
- the performance of an exemplary state analysis for a specific plant
- documentation in order to permit general use of the method

3 Research Program

The project consists of the following tasks:

3.1 Definition of states
3.2 First calculations
3.3 Extension of methods
3.4 Revision of computer programs
3.5 Improvement of details
3.6 Test runs, variation of parameters

3.7 Documentation

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

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5 <u>Progress to Date</u>

(3.1) A basic state model of a reactor plant was developed which is defined by binary characteristics in such a way that the plant is at any time in exactly one of the states. By means of the characteristics the exclusiveness and the completeness of the states can be proved. State transitions are caused by alteration of characteristics and described by a logic. The characteristics of the state model are not necessarily independent, but to some extent caused by common primary events. Hence it is necessary to define the maximal independent units ("degree of freedom", generally subsystems) and to assign them logically to the characteristics.

(3.3) The analysis of a state model by the previously developed computer code ZUSTA requires the preceding evaluation of the transition-logics (analytical or simulative). Using transition-rates for describing the changes of the states the necessary correlation of the characteristics and the memory of the process - i. e. the process not being Markovian - is lost. Therefore a computer code MISIM (<u>mic</u>ro<u>sim</u>ulation) is being developed which can handle the degrees of freedom in their time-dependence, thus avoiding the described deficiency. Taking into account the redundant structure of the systems the transition-probabilities are reproduced correctly.

For specific application different operational strategies with revealed failures of safety systems were defined. They were checked with regard to the possibility of optimisation and prepared for the realistic determination of parameters by the computer code MISIM.

(3.4) The flow charts for the rough structure of the code MISIM were carried out.

(3.5) To demonstrate the feasibility of the developed method of state analysis a model of a specific plant was constructed. With respect to the knowledge of the systems KKW-Kalkar was chosen as a reference plant. The state model does not claim to reproduce the plant in all details correctly and completely. But it demonstrates the treatment of a complex plant with a practicable number of states and degrees of freedom.

(3.7) According to the contract the state of development of the project was reported to the BMFT-Adhoc-Gruppe RISIKO UND ZUVERLÄSSIGKEIT and the BMI-Adhoc-Arbeitskreis RISIKO on 6 July 1978. On request of the BMFT a written version of this interim report was compiled.

6 Results

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7 Next Steps

- Development of the computer code MISIM
- Treatment of system history in the computer code ZUSTA
- Further elaboration and quantification of the state model of a specific plant.

8 Relation with other Projects

9 References

ITB 78.88 Zwischenbericht über das Forschungsvorhaben RS-189.

10 Degree of Availability of the Reports

The interim report ITB 78.88 was distributed to the members of the BMFT-Adhoc-Gruppe RISIKO UND ZUVERLÄSSIGkeit.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	14	RS-201

Vorhaben/Project Title	Land/Country
Zuverlässigkeitsbeurteilung für den Sicherheitseinschluß (SE) am Beispiel des Druckwasserreaktors Reliability Assessment of the Secondary Containment of a PWR	FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Techn. Univ. München Inst.Bauingw.III,GRS, Abt.f.Werkstoffphysik

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.6.1976	30.9.1980	Prof.H. Kupfer,Dr.G.I. Schuëller

Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	December 1978	

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 the 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 to the probabilities of failure of those structures, is investigated. Load effects are determined by static and dynamic structural analysis. The concept to be developed during the 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 Evalutation of the material properties under load effects which are generated by loads as described above. Consideration of fracture mechanical methods for impact loads.

3.5 Evaluation of the load effects using stochastic theory of plasticity.

3.6 Calculation of the structural failure probabilities under single and combined loading.

3.7 Global reliability assessment of the containment.

4. Experimental Facilities, Computer Codes
(With reference to item 3.: Research Program)

3.1 Given a LOCA the ZOCO-Program is used to evaluate the load conditions due to the resulting pressure- and temperature increase.

3.2 The mass- and energy-flow of water and steam out of the primary circuit into the containment during a loss of coolant accident is calculated with the code BRUCH at blow-down phase and with the code WAK at refill phase. The built-up of pressure and temperature in the steel hull is calculated with the code ZOCO.

3.4 An experimental set-up to simulate impact load conditions has been developed. Circular mortar discs can be tested. The force spectrum as well as the resulting strains can be recorded simultaneously. Parallel runs under quasi-static loading conditions can be carried out in a controlled way.

3.5 For the calculation of the load effects under static and dynamic load action in the linear and non-linear range, the SAP and NONSAP-Program systems are used respectively.

5. 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 analyzed as the first case. For this purpose it was assumed that a leak with a cross-sectional area of >1000 cm^2 occuring between the main pump of the coolant and the pressure vessel is the worst case. According to WASH 1400 this failure event has an estimated probability of occurrence of 10^{-4}/a.

Following this major failure event at the end of blow-down and during refill and low pressure recirculation phase maximums of pressure and temperature are possible. The different accident sequencies during refill phase result from function or failure of the systems reactor protection, high pressure injection and low pressure injection. Four accident sequencies of refill phase have been computed to get a first glance at the different possibilities of the entire spectrum of accident sequencies. During blow-down and low-pressure recirculation phase only one accident sequence had to be analyzed.

For calculating the pressure values with the Program ZOCO the following input data have been evaluated:

-the simplified one-room model of the reactor building with internal walls

-the mass and energy flow from the primary system into the containment during blow-down phase resulting from the code BRUCH

-the mass and energy flow from the primary system into the containment during refill phase resulting from the code WAK

-material properties

The influence of 18 parameters on the maximum overpressure at the end of blow-down was checked by a sensitivity study. By means of this study the first terms of the taylor series could be calculated.

After evaluating the statistical values of the parameters, the standard deviation of the resulting overpressure at the end of blow-down was determined. Assuming normal distribution for the parameters and taking into account only the first terms of the taylor series, the maximum overpressure at blow-down is also normally distributed.

3.4 An experimental set-up to test dynamic loading of circular mortar discs has been designed. In this way we can measure simultaneously the force function generated in the missile and the strains at different places on the disc. The time dependence of the implied force and the resulting strains can be observed with a sampling rate to up to 10^6 points/sec. The transient-recorder is connected with a table computer for further processing of the data. With the help of the computer a Fourier analysis of the strain spectrum is calculated. The damping coefficient is also obtained. Parameters of dynamic fracture mechanics are calculated. All calculated data as well as the recorded force and strain functions are finally plotted or printed.
One part of this investigation deals with the dynamic excitation of a mortar disc under impact load. In this way we simulated the conditions in a concrete containment. The aim of the second part is primarily to evaluate material properties.

3.5 The containment structure under the internal load case large LOCA and earthquake loading, loads due to aircraft impact and loads caused by external pressure waves were analyzed. The structure and the soil were modeled by shell and continuum elements. The random characteristics of the material properties, i.e. the statistical scatter of the steel and concrete strengths was also recognized in the analysis. Various failure modes,such as the yield and fracture condition,were utilized.

3.6 Structural failure probabilities of the steel hull have been calculated for the following load conditions: earthquake, internal pressure and temperature build-up due to LOCA as well as the combination of these two loading types. The failure probabilities of the reinforced concrete dome under aircraft impact and external pressure wave have been determined as well. The intensity distributions of the load and the structural resistance have been modeled by Extreme Value Distribution.

Finally, the sensitivity of the reliability due to the statistical uncertainties of some significant parameters was analyzed. For this purpose the influence of a change of mean value and coefficient of variation on the probability of failure was examined.

6. Results

3.1 The numerical analysis of the large LOCA was finished. The mean value of the maximum overpressure in the steel hull has been determined to 3.6 bar and the standard deviation to 0.3 bar (Fig.1). As outlined before some significant parameters for the analysis of the internal load case have been varied in the sensitivity analysis. For this purpose the influence of different mean values and standard deviations was analyzed and is shown e.g. in Fig.2. The investigation of changing the different distribution types for sensible parameters has been started by coupling a generating program for random variables (STREUSL,/7/) and the deterministic program for pressure calculation (COFLOW,/8/).

3.4 The Fourier analysis of the dynamically excited strains indicates that several different vibrational modes are present. Flexural vibrations with the natural and higher modes are generally excited. In addition pulsation of the mortar disc is recorded. Under high impact rate a considerable amount of the energy is transferred to higher mode vibrations.

The dynamic excitation of a mortar disc has been studied in detail. Different transfer processes such as travelling hinge and flexural strains can be clearly separated.

It has been shown that the force function of a given missile is severely influenced by the vibrating structure. It could be shown experimentally that the duration of the implied impact is approximately doubled (in comparison to an impact on a rigid wall) if the duration of the vibration of an essential mode is comparable to the duration of the impact. The peak load, however, is reduced by a factor of two.

An extensive literature survey of material properties under high rate of loading has been carried out. Results have been presented at the SMIRT'77 /2/. From our experimental results we can conclude that failure strain can be increased up to 6 times the quasi-static value at strain rates of $\dot{\varepsilon}=50s^{-1}$. The ultimate load, however, is

only doubled under the same loading conditions. With increasing rate of loading the failure mode changes from normal flexural to local shear failure. The maximum crack propagation in the mortar disc has been estimated to be 250 m/s.
All the final results of the investigations are outlined in more detail in /3/.

3.6 Assuming linear elastic structural behaviour the combined stresses (v.Mises) resulting from the large LOCA have been determined for the boundary regions of the penetrations and the restraint of the steel hull. Failure probabilities p_{f_i} for the fracture mode have been compared in Table 1 to the yield failure condition. Utilizing the Griffith equation the total failure probabilities of the hull have been obtained for an undetected crack of 2 cm length.

For El Centro and Golden Gate earthquake loading as well as for the aircraft impact and external pressure wave, failure probabilities p_{f_i} for the steel hull and the reinforced concrete dome are compared in Table 2. Since the internal and external loads may be considered as rare events their probability of occurrence is statistically modeled by a Poisson process with the mean occurrence rate ν.

The reliability of the steel hull for combined loading, i.e. large LOCA following the El Centro earthquake is outlined in /3/.

To show the sensitivity of the reliability due to statistical uncertainties the probability of failure is plotted in Fig. 2 as a function of the coefficient of variation of the yielding stress. Four different mean values have been used as parameters. If the coefficient of variation V_{σ_y} is small, e.g. below 6 %, a small increase of the mean value σ_y has a relatively large influence on the failure probability p_f. In Fig. 3 the failure probability p_f is plotted as a function of the coefficient of variation of the material resistance. From this figure a large decrease of p_f for small coefficients of variation of load $V_{p_{LOCA}}$ and resistance V_{σ_y} can be concluded. For a large $V_{p_{LOCA}}$, e.g. about 10 %, the increase of p_f for V_{σ_y} between 3 % and 15 % is about two orders of magnitude. On the other hand, for a large V_{σ_y} of about 10 %

the increase of the failure probability for V_{σ_y} between 3 % and 15 % is less than one order of magnitude.

7. Next Steps
(with reference to item 3.: Research Program)
The future work of this research program is subjected to the problem areas as outlined in the RS 201 C project. The main field of investigation in this project is a more detailed reliability assessment for the internal load case small LOCA as well as for the external hazards such as earthquake loading, aircraft impact and special types of load combinations.

8. Relation with Other Projects

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9. References

[1] Schueller, G.I.: A Concept for the Reliability Assessment of the Containment of a PWR, Proc. Int. Conf. Nucl. Syst. Rel. Engr. Risk Assess., Univ. Tennessee, Gatlinburg, June 20-24, 1977

[2] Augustin et al.: A Complex Study on the Reliability Assessment of the Containment of a PWR, Proc. 4th Int. Conf. Struct. Mech. in Reactor Techn., San Francisco, Aug. 15-19, 1977

[3] Augustin et al.: Zuverlässigkeitsbeurteilung für den Sicherheitseinschluß (SE) am Beispiel des DWR, Forschungsbericht zum Vorhaben RS-201, Sept. 1978

[4] Kafka,P., Schueller,G.I.: Probabilistische Zuverlässigkeitsbeurteilung von Sicherheitsbehältern am Beispiel des Druckwasserreaktors, Fachvortrag, GRS-Fachgespräch 1978, Köln, Im Gürzenich, 19.-20. Oktober, 1978

[5] Bauer,J., Schueller, G.I.: Probabilistic Aspects of Containment Design against External Hazards, Proc. Conf.on Struct. Anal. Design and Constr. in Nucl. Power Plants, Porto Alegre, Brasil, April 18-20,1978

[6] Bauer,J.: Reliability-Based Design of the Containment of a PWR - Parameter Study, ICASP 3, Sydney, Australia, January 29-February 2, 1979

[7] Schlösser,L.: STREUSL, ein Rechenprogramm zur Ermittlung der Streuung in Zuverlässigkeitskenngrößen auf Grund der Streuungen der Eingabedaten , Programmbeschreibung, GRS-A 183, (Mai 1978)

[8] Mansfeld,G.: Zum instationären Druckaufbau in Volldrucksicherheitsbehältern wassergekühlter Kernreaktoren nach einem KMV-Störfall, Dissertation, TUM, 4.Juli 1977

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

non-classified

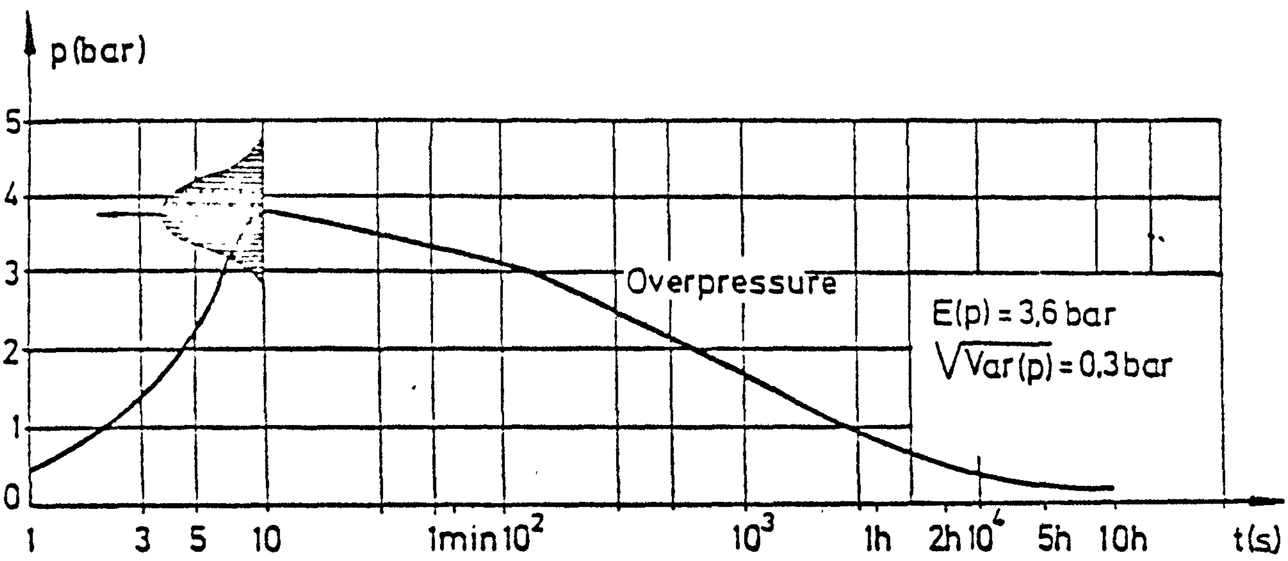

Fig. 1: Pressure Distribution of the Containment
Resulting from a large LOCA

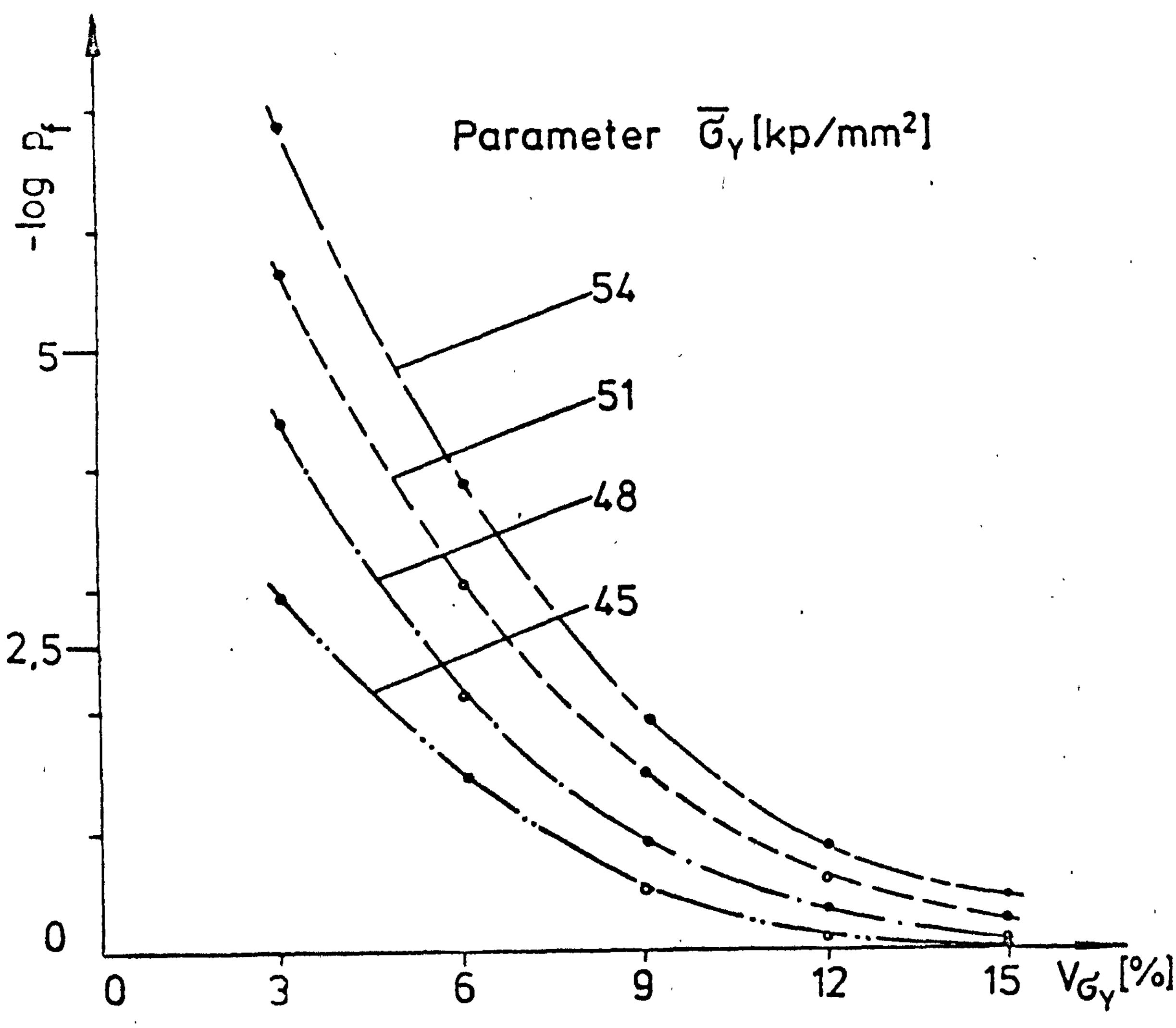

Fig. 2: Failure Probability as a Function of the
Coefficient of Variation of the Yielding Stress σ_Y

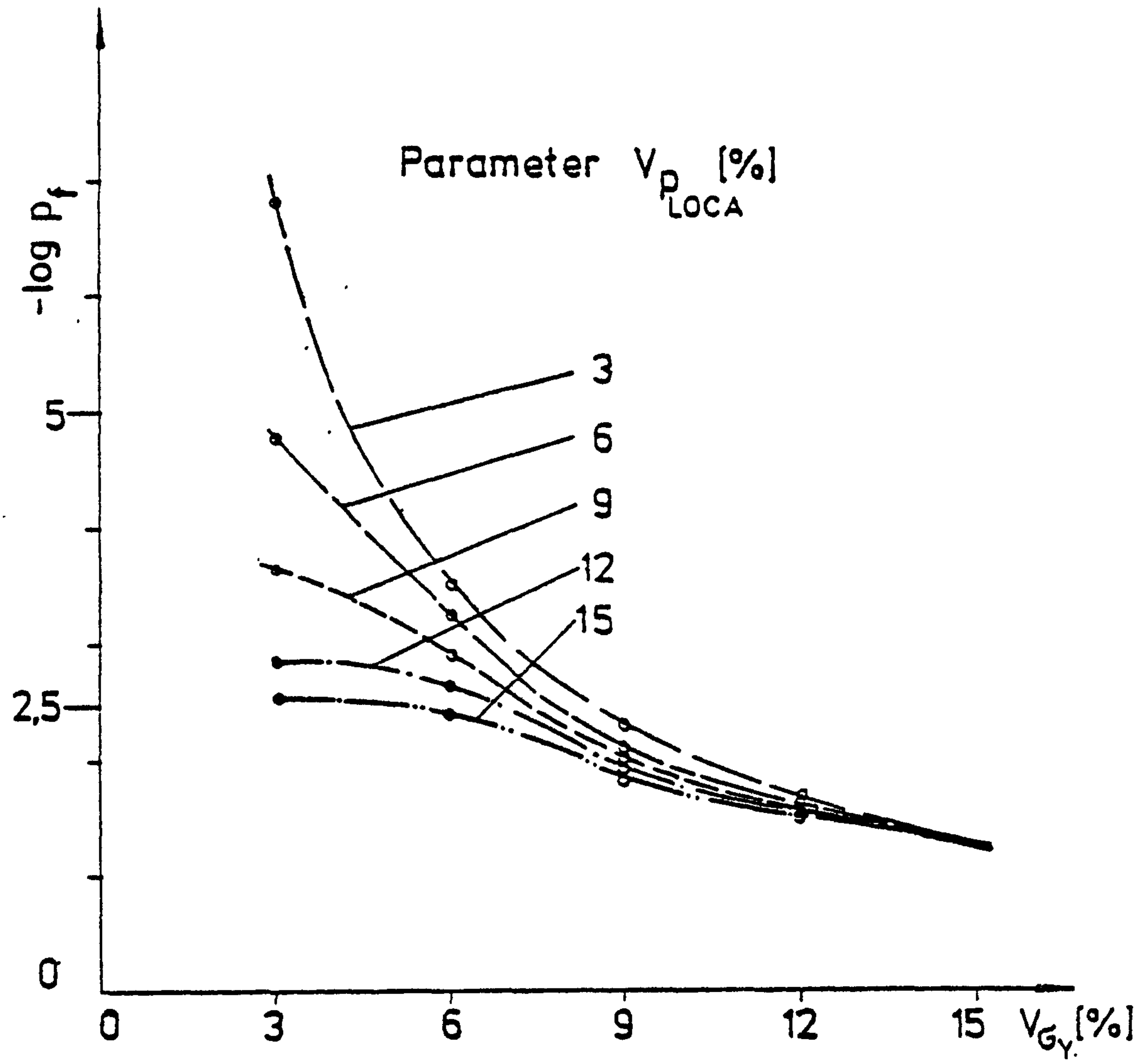

Fig. 3: Failure Probability as a Function of V_{σ_Y} and Parameter $V_{p_{LOCA}}$

Failure Condition	Shell	Load		Resistance		P_{f_i}	ν	$F_L(t)$ **
		$u\,[kp/mm^2]$	$\alpha\,[kp/mm^2]^{-1}$	$u\,[kp/mm^2]$	$\alpha\,[kp/mm^2]^{-1}$			
Yield	no holes	0,0430	890,3	0,091	41,7	$<10^{-8}$	10^{-4}	$<10^{-8}$
Yield	holes	0,0430	890,3	0,053	43,0	$8,7\cdot 10^{-4}$	10^{-4}	$2,6\cdot 10^{-6}$
Fracture*	no holes	0,0430	890,3	0,103	20,0	$4,1\cdot 10^{-8}$	10^{-4}	$<10^{-8}$
Fracture*	holes	0,0430	890,3	0,061	20,0	$1,6\cdot 10^{-3}$	10^{-4}	$4,8\cdot 10^{-6}$

*) crack length 2c = 2cm **) design life L = 30 years

Table 1: Failure Probabilities of the Steel Hull Following a Large LOCA

LOAD TYPE	STRUCTURE		FAILURE CONDITION	P_{f_i}	ν	$F_L(t)$
AIRCRAFT IMPACT	REINFORCED CONCRETE DOME		GLOBAL YIELD	$8.2 \cdot 10^{-3}$	10^{-6}	$2.5 \cdot 10^{-7}$
			LOCAL YIELD	$2 \cdot 10^{-2}$	10^{-6}	$6 \cdot 10^{-7}$
EXTERNAL PRESSURE WAVE	REINFORCED CONCRETE DOME		GLOBAL YIELD	$< 10^{-8}$	10^{-6}	$< 10^{-8}$
EARTH-QUAKE LOAD	REINFORCED CONCRETE DOME	EC*/ROCK	YIELD	$3 \cdot 10^{-6}$	$1 \cdot 10^{-5}$	$< 10^{-8}$
		GG**/ROCK	YIELD	$< 10^{-8}$	$5 \cdot 10^{-4}$	$< 10^{-8}$
		EC/SOIL	YIELD	$< 10^{-8}$	$1 \cdot 10^{-5}$	$< 10^{-8}$
		GG/SOIL	YIELD	$< 10^{-8}$	$5 \cdot 10^{-4}$	$< 10^{-8}$
	STEEL HULL	EC/ROCK	YIELD	$< 10^{-8}$	$1 \cdot 10^{-5}$	$< 10^{-8}$
		GG/ROCK	YIELD	$< 10^{-8}$	$5 \cdot 10^{-4}$	$< 10^{-8}$
		EC/SOIL	YIELD	$< 10^{-8}$	$1 \cdot 10^{-5}$	$< 10^{-8}$
		GG/SOIL	YIELD	$< 10^{-8}$	$5 \cdot 10^{-4}$	$< 10^{-8}$

Table 2: Failure Probabilities of the Containment
Structure due to External Hazards
*"El Centro" Record; **"Golden Gate" Record

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78 - 31.12.78	14	RS 270
Vorhaben/Project Title Zuverlässigkeitserhöhende Untersuchungen an Meß- und Regelgeräten der Kernkraftwerkstechnik Investigations to improve the reliability of measurement and control devices for nuclear power station technology		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor INTERATOM GmbH
Arbeitsbeginn/Initiated 01.07.1977	Arbeitsende/Completed 31.12.1981	Leiter des Vorhabens/Project Leader H. Banasch (Koord.)
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

The aim is to improve the reliability of measuring and control devices by applying aging procedures still to be defined.

The suitability of applying effective aging methods shall be proved by means of a comparative analysis between preaged and nonaged devices.

2. Particular Objectives

Elaborating optima aging methods by testing different aging procedures.

Investigating the failures of the devices and verifying the change of their technical data.

Compiling the reliability data of the electronic construction elements, of the construction groups and of complete measuring and control devices used for the investigations in order to calculate the reliability and comparing them with the data determined experimentally.

3. Research Program

3.1 Elaboration of aging specifications taking into account the parameters: time, temperature, acceleration as well as overvoltage and undervoltage.

3.2 Application of selected aging methods, optimization of these applied procedures and definition of aging specifications. Testing the aging procedures in a life test running for approximately 3 years under normal conditions.

3.3 Calculating the reliability of the devices used for the investigations.

Compiling the reliability data of each individual construction element provided in the devices being investigated, taking into account the manufacturer's data. Calculating the failure probability of the devices included in the test, using the preceding data. Comparison of the failure data determined experimentally with the failure probabilities determined theoretically.

4. Test Facilities, Computer Codes

Applying a process-computer-controlled test system and various problem-oriented test programmes as well as temperature chambers and a servo-hydraulic vibration table.

5. Progress to Date

To 3.1 The aging methods have been defined.

To 3.2 The manufactoring of the necessary additional specimens has been finished.

Planning, installing and putting into operation of the setup for testing and aging could be finished. The software for the process computer could nearly be completed.

The process of aging has been started.

6. Results

To 3.2 The first measurement showed 45 per cent. of the burst disks' monitoring electronics being defective. As a reason for the failure wrong component parts were located.

7. Next Steps

To 3.2 Continuation of the aging procedures.

Optimization of the aging methods applied.

Attending of the long-time test.

8. Relation with other Projects

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9. References

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	14	RS 264
Vorhaben/Project Title Collection and Evaluation of Reliability Data at the Nuclear Power Plant Biblis B Zuverlässigkeitskenngrößenermittlung am Kernkraftwerk Biblis B		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Gesellschaft für Reaktorsicherheit mbH Glockengasse 2 5000 Köln 1
Arbeitsbeginn/Initiated 1.7.1977	Arbeitsende/Completed 31.12.1980	Leiter des Vorhabens/Project Leader Hömke
Stand der Arbeiten/Status coutinuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

The failure behavior of components is influenced by various parameters, as construction data, operational data and operational stress. By analysing the data collected in the above mentioned data collection it is intended to estimate reliability data which are specified with respect to the most important determining parameters. Where possible it is intended to calculate distributions of reliability data and confidence intervalls.

2. Particular Objectives

The project has four different objectives.

2.1 Development of the Basic Structure and Methods for the Data Collection

A pilot data collection was carried out in a lignite power plant with this experience a new data base structure and new data sheets have to be developed.

2.2 Collection of Raw Data

Before starting any evaluation of reliability data it is necessary to collect all engineering, operation and event data of interest.

2.3 Setting up of a Data Bank

For the handling, the updating of data and the evaluation of results is was found necessary to set up a data bank. It is intended to use a data bank managementsystem.

2.4 Evaluation of Reliability Data

It is intended to make three subsequent evaluations of the data base to get the first results of the data collection as soon as possible.

3. Research Program

3.1 Preparation of the Data Collection

The collection is based on a data collection hand book, which describes the sheets and the method of collection. All the coding and the structure of the data base has to be developed.

3.2 Collection of Raw Data

The engineering data, as system and component data, will be collected for up to 15ooo components. The failure data are based on the maintenance sheets, which are collected and completed. The operation time is provided by the process computer of the station.

3.3 Installation of a Data Bank System

The data base is managed and updated with a data bank management system. A data bank has to be structured and set up.

3.4 Storage of Data into the Data Banks

The collected data will be punched and formal checks are applied. After storing in the data banks there is a logical check of the data.

3.5 Evaluation of Reliability Data and Documentation of the Results

Evaluation of reliability data will be performed after the collection of the basic data when enough failure reports are available.

4. Experimental Facilities Computer Codes

The data bank management system "System 2000" from MRI is used for storage of the data.

5. Progress to Date

to 3.1 Preparation of the Data Collection

The data collection handbook is nearly completed, the nescessary sheets are available. Momentary an update is carried on.

to 3.2 Collection of Raw Data

The collection of engineering data is still running. Some difficulties in the updating and availability of technical data habe lead to a delay in the collection. The event data and the operation data have been collection since April 78. By the end of August a computer program has been provided to punch the operation data.

to 3.3 Installation of a Data Bank System

The data bank management system "System 2000" has been fully implemented. Neccessary instruction courses are finished. Untill now the folowing two data banks have been developed

-"DB Counter" for the storage of Operation times and cycles and

-"DB Anlagen" for the list of systems and components, the system data, and component key data.

The computer codes for the preload formal check and the load modules for DB Counter and DB Anlagen are developed and nearly completed. Due to a change of our computer, this part of the program has got a delay.

to 3.4 Storage of Data into the Data Banks

The available operation data are stored in "DB Counter". For test purposes the "DB Anlagen" was loaded with data from one system.

6. Results

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7. Next steps

to 3.1 Preparation of the Data Collection

It is planned to update and to publish the handbook.

to 3.2 Collection of Raw Data

The work of data collection is continued

to 3.3 Installation of a Data System

The computer codes for the formal check and the load modules for the component typ data and event data will be provided next, afterwords the data bank for both will be structured.

to 3.4 Storage of Data into the Data Banks

The data will be fed in the data bank as they are available from the collection people.

8. Relation with other Projects

The project is a successor of the project "Modellfall IRS - RWE - zur Ermittlung von Zuverlässigkeitskenngrößen.

9. References

P. Hömke, H. Krause
Der Modellfall IRS - RWE - Zur Ermittlung von Zuverlässigkeitskenngrößen im praktischen Betrieb -
IRS-W-16 (November 1975)

B. Wohak, G. Meinlschmidt
Ein Informationssystem zur Gewinnung von Zuverlässigkeitsdaten für Kraftwerkskomponenten
MRR 159 (Juni 1976)

1o. Degree of Availability of the Reports

The reports are distributed by Gesellschaft für Reaktorsicherheit mbH, Glockengasse 2, 5ooo Köln 1

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
Jan.1-Dec.31 1978	14	RS 252
Vorhaben/Project Title: Schadensumfangsanalyse für HTR Accident Consequence Analysis for HTGR		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: HRB GmbH 6800 Mannheim P 5
Arbeitsbeginn/Initiated: Jan. 1, 1977	Arbeitsende/Completed: Dec. 31, 1978	Leiter des Vorhabens/Project Leader: Gabriel/Redondo
Stand der Arbeiten/Status: Completed	Berichtsdatum/Last Updating: Dec. 15, 1978	Bewilligte Mittel/Funds

1. <u>General Aim</u>

The objective of the project is to analyse ultimate accident sequences of the HTGR 1160 system /1/. Given the start accident conditions of the simultanious failure of the active systems (no trip, no core-cooling, worst position of valves), the time-dependent consequences for passive components of the plant (fuel claddings, primary boundary, containment) as well as for the environment should be determined. The goal is the quantification of the shortest time scale available

- to stop the accident sequence in the plant
- for the reduction of public doses by means of administrative measures.

Based on the results of the analysis, a new HTGR-design should be optimized against ultimate accident consequences.

All accident sequences in this project are defined without probabilistic assumption - only the inherent physical character of the reactor system should be analysed.

2. Particular Objectives

2.1 Models

The first part of the analysis elaborates physical-technical descriptions (conditions, effects and failures) in order to deepen the knowledge of the main "event steeps" and to render possible a correct definition of the Ultimate Accident Sequences (UAS) with the largest consequences (highest environmental doses) considering the passive-systems response under UAS-start conditions and their time-dependent effectiveness as barriers, and taking into account their physical and structural properties. These descriptions are called "models".
The models for the UAS are the following:

Model 1	Primary system temperatures
Model 2	Steam Generator / Heat Exchanger Failure
Model 3	Reactivity effects
Model 4	Pressurized PCRV response
Model 5	Depressurized PCRV response
Model 6/7	Fuel failure and particle release, diffusion in graphite

2.2 Actions to stop the accident sequence

The shortest time available to take action against the accident as well as the actions itself have been analysed for the postulated hypothetical accident sequences.

2.3 Ultimate accident sequences

The second part of the analysis deals with three UAS branches, each of them with a certain of Release Models (RM):

UAS 1.1 Steam/Water Ingress (16 RM)
UAS 1.2 Air Ingress (10 RM)
UAS 1.2 Reactivity Excursion (3 RM)

The RM have to quantify path of emission, environmental doses and contamination as a function of time.

3. Research Program

Tasks and time-schedule of the program specified in the appendix of HRB letter from Dec. 8, 1976 have been up-dated in May 1978 /1/. The institutions GAC (San Diego, USA), Fichtner (Stuttgart), IKE (Stuttgart) are involved in the program. The ISF of KFA/Jülich is involved as an adviser. Program and results are discussed in the framework of the LARi (Lenkungsausschuß Risikostrategie), a group made up of representatives from all these institutions.

The program has been specified (see Ref.2,3,4,5,6,7 of /2/) mainly in 1977, the main calculations to the UAS 1.1, UAS 1.2 and UAS 1.3 have been performed /3; 4; 8; 9; 10; 11; 12; 13; 14/ in 1977 and finally reported /17/ in 1978.

4. Experimental Facilities

The specified scope of the program includes only analytical tasks (s. point 8).

5. Progress to Date

HRB, GAC, Fichtner and IKE completed the task described in /1/ after some increase in the costs of the project.

6. Results

The summary of results has been stated and referenced in /17/.

6.1 Models

The mechanistic analysis of the HTGR 1160 on the time schedule for failures of the passive radioactivity barriers under UAS 1-Conditions shows - as expected - that the coated particles and the fuel elements are a very resistant barrier: after aproximately 4 h, through 8 h there is the beginning of significant acitivity release.

The corresponding schedule for the metalic components shows failure times of > 1,5 h for Steam Generators, > 2 h for the closures of the PCRV and the failure time for the containment is >4,5h after initiation of the core heat-up.
A recriticality event as a potential course to shorten the failure time for passive components does not need to be considered before 18 h.

6.2 Actions to stop the accident sequences

On the basis of the developed Models (system behaviour) the possiblity to take action to stop the event and/or to avoid irreparable damage to components important to safety using operational means (i.E. using the hardware of the power plant in the foreseen of in a original manner) has been analysed. Dominant is the reactivation of a cooling chain and this must be achieved in a time between 1,5 and 3 h after initiation of the core heat-up.

6.3 Radiological consequences

Considering that these actions against the event do not happen, 29 release models have been investigated and the environmental dose has been calculated as a function of exposition time and distance from the source. Additionaly to the mechanistic of possible release paths, other investigations have been performed varying the mechanistic parameters (water ingress-mass and -rate; failure time of the passive barriers and the plate-out factors), to clarify the sensibility of the results w.r.t changes in the mentioned parameters.
The highest consequences for the environmental are given by the water ingress event, but even in this case, a presence time up to 5 h at a distance of 5 km is possible without surpassing the remissible limits for the whole body dose of 5 rem.

For the further development of the HTGR, some ideas on possible Design-Optimization to get an additional safety improvement, especially with regard to consequence reduction have been elaborated out of the analysis.

7. Next steps

On the basis of the described results, a program on Design Optimazation and Ultimate Consequence Analysis for the new HTGR systems (with pebble bed core) has to be written in detail. This is out of the scope of this project-phase.

The research task RS 252 has been completed.

8. Relations to other projects

Most related projects with the Ulitmate Consequence Analysis for HTGR in the field of HTGR-safety research are

- AIPA-Study of GAC sponsored by DOE.
- Probability Safety Study HTGR (PSH) by ISF-KFA/GRS supported by BMI,Germany.

Both studies use the probabilistic approach to analyse event sequences. It was possible to observe a complementary character of PSH-Phase Ia with the task of the Ultimate Consequence Analysis.

9. References

/1/ HRB-Loseblattsammlung zum Sicherheitsprojekt "Schadensumfangsanalyse für Hochtemperaturreaktoren"
19.8.76/5.5.77/last up-dating 22.5.78

/2/ "Schadensumfangsanalyse für HTR"
"Ultimate Consequence Analysis for HTGR"
RS 252 , Jahresbericht 1977

/3/ HRB-Risk-Study, Phase II Final Report (Draft)
Nov. 1977, GAC-San Diego

/4/ Action Items 1 though 20, Feb. 21, 1978,
Responses resulting from Nov./Dec. 1977
GAC/HRB meeting in San Diego

/5/ Kritische Durchsicht des HRB/GAC-Berichtes
zur Phase II der Schadensumfangsanalyse
Fichtner Bericht, Stuttgart, Juni 1978

/6/ HRB BB 2419 vom 22.5.78, Besprechungsbericht
Zusammenfassung der Diskussion über HTR-Schadensumfangsanalyse Phase 2 -(GAC-Beitrag).

/7/ P.-G. Ceyrowsky
Kritische Durchsicht des HRB-GAC-Berichtes
zur Phase II der Schadensumfangsanalyse,
Fichtner-Bericht, Juni 1978

/8/ Arbeitsbericht der Fa. Fichtner (Stuttgart)
vom 17. Juli 78
"Konsequenzenanalyse der Störfallsequenzen
der HSK 1.1 und HSK 1.2"

/9/ Arbeitsbericht der Fa. Fichtner (Stuttgart) vom 24. Juli 78 "Fortran Programm CONT mit Beispielen"

/10/ Arbeitsbericht der Fa. Fichtner (Stuttgart) vom 19. Juli 78 (848 S06, Pflieger) "Konsequenzenanalyse der Störfallsequenzen der HSK 1.1 und HSK 1.2 - Programmdokumentation"

/11/ Arbeitsbericht der Fa. Fichtner (Stuttgart) vom August 78 "Beschreibung technisch möglicher Betriebsmaßnahmen zur Unterbindung der Störfallabläufe der HSK 1.1 und 1.2" P.-G. Ceyrowsky

/12/ HRB-Arbeitsbericht BA 2590 vom 18.8.78 "Spaltproduktfreisetzung aus dem Kern des HTR 1160 für die HSK 1.3"

/13/ Schadensumfangsanalyse Phase II Reaktivitätstransient in HSK 1.3 HRB-Bericht BF 0272 vom 17.11.78

/14/ Untersuchungen zum Reaktivitätsverhalten von HTR bei der Störfallsequenz HSK 1.4, IKE RS 252

/15/ Untersuchungen zu den Auswirkungen hypothetischer Störfälle bei Hochtemperaturen

Teil 1: Schadensumfang beim Ausfall der gesamten Wärmeabfuhr mit zusätzlicher Versagen aktiver Sicherheitseinrichtungen (HSK 1) HRB-BA 1854/77, Sept. 77

/16/ Untersuchungen zu den Auswirkungen hypothetischer Störfälle bei Hochtemperaturreaktoren

Teil 2: Schadensumfang beim Bruch des Primärkreislaufes mit zusätzlichem Versagen aktiver Sicherheitseinrichtungen KFA/ISF-Jül-1466, Nov. 77

/17/ HRB-Arbeitsbericht BA 2702 RS 252, Dez. 1978 Schadensumfangsanalyse für Hochtemperaturreaktoren

10. Degree of Availability of the Reports

All references from chapter 9 are available from BMFT, GRS and LARi. A distribution of these References to third persons, groups or organizations needs the previous permission of HRB.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1978-31.12.1978	14	RS 338

Vorhaben/Project Title	
Wechselwirkung zwischen technischen und rechtlichen Aspekten bei Risikobeurteilungen Legal problems of risk valuation considering particulary the nuclear energy law	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Universität Münster

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.4.1978	31.12.1979	Lukes
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

Study and report of the legal problems in connection with risk assessment and the use of risk analysis in the legal systems of Germany, the United Kingdom, France and the United States of America.

2. Particular Objectives

Study of how the 4 legal systems mentioned before consider risk situations - in particular in the field of nuclear energy law - in acts, administrative decisions and court decisions.

3. Research Program

Survey of acts, administrative decisions, court decisions and literature as to Germany, the USA, France and the UK.

4. Experimental Facilities, Computer Codes

–

5. Progress to Date

France: Draft report is written and is being revised.
England: Draft report is written and is being revised.
USA: Draft report is written and is being revised.
Germany: Survey is started on, but not yet finished.

6. Results

It is not possible to state a short result in only a few sentences, for the draft reports as to France, England and the USA comprise each some 160 typewritten pages. Summaries of the results (about 1 typewritten page) can only be given in the annual report.

7. Next Steps

France: Completion of the report after revision. Realisation of the proposed stay in France to complete the report.

England: Completion of the report after revision. Realisation of the proposed stay in England to complete the report.

USA: Completion of the report after revision. Realisation of the proposed stay in USA to complete the report.

Germany: Continuing collection of material and making the report.

8. Relation with other Projects

-

9. References

No final report made yet; references are too voluminous to be used in this quarterly report.

10. Degree of Availability of the Reports

-

Summary (Great Britain)

The law relating to industrial installations consists of a number of acts and regulations within the framework of the Health and Safety at Work Act 1974.

In the conventional technological HSW field risk assessments mostly apply but a qualitative general and flexible approach. The regulations as to fire precaution and major hazardous installations contain a combined qualitative-quantitative approach based upon engineering experience. The operator of a major hazardous installation may use hazard analysis in the "hazard survey" he is to make.

The administrative implementations of the lapidary legal requirements for nuclear installations and aircraft are mainly based upon deterministic assessments. Probabilistic approach has achieved increasing importance.

In the field of administrative planning decisions the Canvey Island investigation provides a first example for use of risk analysis to determine the risk potential of an industrial complex of conventional chemical and petrochemical installations.

Summary (France)

The French law of nuclear power plants is very little elaborated. Particularly questions of nuclear safety are understood as an exclusive "domaine de l'ingénieur". The same impression is given by the reserve of the "Conseil d'Etat" controlling licensing desicions.

So it's not surprising that one won't find any article regarding our special theme written by a lawyer. On the contrary engineers are contributing to the discussion at a high extend. Interesting to be noted by a lawyer was a kind of "touching anxiety" technicians showed treating with probabilistic methods, meeting the limits of the "countable" and feeling the necessity to enter the area of normativity, of values, i.e. "le domaine du juriste". An example is the discussion about the "niveau de l'acceptabilité". A view cast an other fields of legal science gave the impression that in French law the term of "risk" is not completely unknown (for ex. the "théorie des risques anormaux de voisinage").

Summary (USA)

In the licensing process of nuclear power plants adequate protection of public health and safety is determined by a finding that the facility will operate in conformity with the provisions of the AEA and the rules and regulations of the US NRC. Generally there is no risk assessment on a quantitativ basis except for potential accidents involving hazardous materials or activities in the vicinity of the plant.

Under the regulatory scheme of the NEPA there is a broad cost-benefit analysis in order to determine if the construction and operation of a nuclear power plant can be favoured. One factor considered in the cost-benefit analysis is the environmental risk of the facility. Concerning the licensing of floating nuclear power plants the environmental risk was quantified for the first time in monetary terms using a quantitative numerical probability analyis of postulated accidents and an analysis of their consequences.

The application of a cost-benefit analysis and a risk-benefit approach is found in many areas of the american law, especially in the environmental law.

	CLASSIFICATION: 14
TITLE (ORIGINAL LANGUAGE): Vurdering af systemers pålidelighed	COUNTRY: Denmark SPONSOR: Risø National Laboratory
TITLE (ENGLISH LANGUAGE): Evaluation of the Reliability of Systems	ORGANISATION: Risø National Laboratory 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

For the purpose of analysing the reliability of systems with a high degree of complexity in design or operation various Monte Carlo programs were developed. The most recent of these programs is MOCARE (ref.2). The modelling in this program is particularly flexible since all conditions for the occurrence of failures can be specified by means of subsystems. Extensive control of system and model performance can be obtained by graphical displays on a plotter, lineprinter or screen.

4.2 Essential results

The MOCARE program has been tested extensively for instance on a power supply system of a nuclear power station, with very complex operational requirements.

5. Next steps

The MOCARE program will be further developed to include options for application of variance reduction techniques as part of a Ph.D. thesis project (see below).

6. Relation with other projects

The project has close relations to the Risø projects concerning the FAUNET program, the Ph.d. thesis on Optimization of Reliability Techniques and on Structural Reliability.

7. Reference documents

1. REDIS, A Computer Program for System Reliability Analysis by Direct Simulation. H.E. Kongsø. IAEA-SM-195/17.

2. MOCARE, A Computer Program for System Reliability Analysis by Monte Carlo Simulation. Program Description and Manual. Hans Erik Kongsø, Hans Larsen, Kjell Nilsson and Kurt Erling Petersen. October 1978. RISØ-M-2109.

8. Availability

The project information is freely available apart from cases where commercial interests may be violated.

Classification: 14

Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: Faunet - a program package for fault tree and network calculations.	Organization: Risø National Laboratory
Initiated date: 1976 Status: Completed date: 1. January 1979 Last updated	Scientists: O. Platz J.V. Olsen

1. General aim

To develop a versatile program package for calculation of reliability and availability for systems represented by fault trees or networks.

2. Particular objective

To develop a modularization technique in order to make it possible to perform fault tree analysis on a minicomputer.

3. Experimental facilities and programmes

4. Project status

1. Progress to date Two versions of the package working, one on a 16 K PDP8 and another on a Burroughs B 6700.

2. Essential Results The package has been extensively tested on fault trees and networks taken from the literature.

5. Next Steps

6. Relation with other projects

7. Reference Documents

1. O. Platz and J.V. Olsen, FAUNET: A Program Package for Evaluation of Fault Trees and Networks. Risø Report no. 348, September 1976.

2. O. Platz and J.V. Olsen, FAUNET: A Program Package for Fault Tree and Network Calculations. In Proceedings of ANS Topical Meeting on Probabilistic Analysis of Nuclear Reactor Safety, May 8-10 1978, Los Angeles, California.

8. Degree of availability

Classification: 14

Title: Optimering af pålidelighedstekniske metoder	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: Optimization of Reliability Techniques	Organization: Risø National Laboratory
Initiated date: 1977 Completed date: Status: In progress	Scientists: Kurt Erling Petersen

1. General Aim

To optimize Monte Carlo methods as well as numerical methods in analysis of reliability of structures and systems.

2. Particular objectives

Development of computer codes based on numerical integration in several variables and Monte Carlo methods with variance-reduction techniques.

3. Experimental facilities

Not applicable.

4. Project status

The project is a part of a Ph.D. thesis and was started in October 1977.

Until now work has concentrated on generators of random numbers and numbers from specified distribution functions.

5. Next steps

Work concerning Monte Carlo methods will concentrate on different variance-reduction techniques. The aim is to optimize the methods, i.e. to minimize the number of simulation trials when a specified accuracy is given.

Work concerning numerical methods will concentrate on integration of functions in several variables. The aim is to minimize the number of evaluations of the function when a specified accuracy is given.

6. Relation with other projects

The project has close relations to the Risø-project concerning evaluation of the reliability of systems (classification 14) and the Risø-project concerning reliability of structures (classification 14).

7. Reference documents

8. Availability

The project information is freely available.

Classification: 14

Title:	Country: DENMARK
	Sponsor: Risø National Laboratory
Title: RIKKE - a program system for automatic fault tree construction	Organization: Risø National Laboratory
Initiated date: 1977 Completed date: Status: continuing Last updated May 1978	Scientists: J.R. Taylor

1. General aim

Automatic or semi automatic construction of fault trees, cause consequence diagrams, and process plant simulations.

2. Particular objective

To be able to perform routine failure analyses of process plants, and construct simulation models interactively, directly from a flow sheet of the plant, the transformation to mathematical models being carried out by computer.

3. Experimental facilities and programmes

4. Project status

1. Progress to date First version of program now working on DEC PDP 11 and Burroughs B 6700 computers.

2. Future work Practical use of the program .

5. Next Steps

6. Relation with other projects

7. Reference Documents

1. Experience with Algorithms for Automatic Failure Analysis. J.R. Taylor, E. Hollo.
2. Nuclear Systems Reliability Engineering and Risk Assessment. SIAM 1977.

8. Degree of availability

	CLASSIFICATION: 14
TITLE (ORIGINAL LANGUAGE):	COUNTRY: Denmark
	SPONSOR: Risø National Laboratory
TITLE (ENGLISH LANGUAGE): Stafan - a program package for statistical analysis of reliability and life data	ORGANISATION: Risø National Laboratory
	PROJECT LEADER: O. Platz

INITIATED: 1978	COMPLETED:	SCIENTISTS:
STATUS: Continuing	LAST UPDATING: April 1979	

1. General aim

To develop a versatile program package for interactive statistical analysis of reliability and life data on a minicomputer.

2. Particular aim

To provide facilities for graphical displays of empirical distribution functions - and to provide programs for parameter estimation and testing of censored failure data.

3. Experimental facilities and programmes

4. Project status

1. Progress to date. Programs for probability plotting of complete, grouped or censored data from the exponential, normal, log-normal, Weibull and extreme value distributions are developed as well as a program for total time on test plotting.

2. Essential results.

5. Next steps

6. Relation with other projects

144-3-01		14
TITRE PROBABILITES D'ACCIDENTS SUR LES PWR - RECHERCHE DES INITIATEURS D'ACCIDENT ET DETERMINATION DES SEQUENCES ACCIDENTELLES		Pays FRANCE Organisme Directeur CEA/DSN/EdF/N/FRA
TITLE (Anglais) PWR ACCIDENT PROBABILITIES - INITIATING EVENTS AND ACCIDENT A SEQUENCE STUDIES		Organisme exécuteur CEA/DSN/SETS/BEPS Responsable
Date de démarrage 9/76	Etat actuel en cours	Scientifiques
Date d'achèvement 12/81	Dernière mise à jour	

1. OBJECTIF GENERAL

Evaluation du risque présenté par une installation nucléaire PWR par l'utilisation de méthodes probabilistes.

2. OBJECTIFS PARTICULIERS

- Mise au point d'une méthode pour effectuer la recherche des évènements initiateurs d'accident.
- Mise au point d'une méthode de représentation des séquences accidentelles.
- Mise au point d'une méthode de quantification probabiliste.
- Application aux centrales PWR

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES : Néant

4. ETAT DE L'ETUDE

- La méthode de recensement des initiateurs est pratiquement au point.
- Pour les autres objectifs, on s'efforce à l'heure actuelle, de mettre en évidence les problèmes présentés par les méthodes habituelles (par exemple celles du WASH 1400) de manière à voir dans quel sens il est nécessaire

Prochaines étapes

- Les prochaines étapes sont pratiquement décrites en (2) par les objectifs particuliers.

- Comme il semble bien que seuls les processus-stochastiques puissent permettre une quantification convenable des séquences, une étape importante parait-être celle consistant à essayer d'introduire ces processus dans cette étude.

5. RELATION AVEC D'AUTRES ETUDES

Au niveau de la quantification, cette étude est en relation avec celle portant sur les processus stochastiques (fiche n°144-1-03)

6. DOCUMENTS DE REFERENCE

Rapport DSN n° 62 : Séquences accidentelles importantes pour la sûreté d'une centrale à eau - A. CARNINO, B. GACHOT, J.P.SIGNORET Avril 1975

International Conference on Nuclear System - TENNESSEE Juin 1977 "Determination of initiating events and sequences of reactor accident by a barrier analysis" - A. CARNINO, J. DUBAU (DSN n° 163)

7. DEGRE DE DISPONIBILITE

Libre

144-3-03 /4105-20 14

TITRE Accident de perte des pompes primaires PWR : APPLICATION DE LA METHODE DES SURFACES DE REPONSE	**Pays** FRANCE **Organisme Directeur** CEA/DSN
TITLE (Anglais) Accident of loss of primary pump on a PWR : ASSESSRMENT BY RESPONSE SURFACE METHODOLOGY	**Organisme exécuteur** CEA/DRE-SERMA **Responsable**

Date de démarrage 1.10.1978	**Etat actuel** en cours	**Scientifiques**
Date d'achèvement 30.3.1979	**Dernière mise à jour** 1-2- 79	

1. OBJECTIF GENERAL

Réaliser une étude donnant la distribution statistique des effets de perte des pompes primaires à partir des distributions des paramètres et données d'entrée en appliquant la methode de la surface réponse .

2. OBJECTIFS PARTICULIERS

Mettre en oeuvre la méthode des surfaces de réponse couplée à une simulation MONTE-CARLO pour calculer l'accident de perte des pompes primaires.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

néant

4. ETAT DE L'ETUDE

Avancement à ce jour :
La méthodologie des surfaces de réponses et MONTE-CARLO a été établie et un programme permet de réaliser cette étape. La détermination de la surface de réponse en est l'élément essentiel.

5. PROCHAINES ETAPES

- Détermination des paramètres influents sur le calcul dynamique de l'arrêt des pompes primaires et de leur distribution statistique.
- Détermination de la surface de réponse adaptée et simulation MONTE-CARLO.

144-1-01 / 154-1-01 /4105-20		14
TITRE DISPONIBILITE DE SYSTEMES EN ATTENTE PERIODIQUEMENT TESTES APPROCHE ANALYTIQUE		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) PERIODICALLY TESTED STANDBY SYSTEMS AVAILABILITY ANALYTICAL APPROACH		Organisme exécuteur CEA/DSN - SETS-BEPS Responsable
Date de démarrage 1.12.75	Etat actuel en suspens	Scientifiques
Date d'achèvement 1978	Dernière mise à jour 12.77	

1. OBJECTIF GENERAL

- Calcul de la disponibilité instantanée et moyenne des systèmes en attente testés à intervalles réguliers.
- Optimisation de la disponibilité moyenne.

2. OBJECTIFS PARTICULIERS

1) Elaboration d'une formule d'indisponibilité utilisable par le code PATREC.

2) Calcul de la disponibilité d'un système à 1 composant

3) Calcul de la disponibilité d'un système complexe

4) Prise en compte de paramètres divers

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

Néant

4- Etat de l'étude :

1) Avancement à ce jour :

Les objectifs N° 1 et 2 sont réalisés, une méthode a été proposée pour aborder l'objectif N° 3, quand à l'objectif N° 4 il vise à l'amélioration du modèle mis au point dans les 3 premiers objectifs. C'est à dire qu'il n'a pas de limite définie a priori mais des résultats partiels ont été obtenus.

2) Résultats essentiels :

- Mise au point d'un modèle conduisant par un formalisme rigoureux à l'élaboration de formules analytiques pour :
  - le calcul de la disponibilité instantanée prévisionnelle d'un système considéré comme faisant un seul bloc.
  - le calcul de la disponibilité moyenne d'un tel système.
  - l'optimisation de l'intervalle entre test d'un tel système.
- Mise au point d'une méthodologie permettant grâce à un raisonnement "physique" basé sur des hypothèses d'approximation généralement vérifiées en pratique de faire les mêmes calculs que ci-dessus.
- Regroupement des 2 méthodes : les formules approchées trouvées directement par le raisonnement "physique" se retrouvent bien comme développements limités des formules rigoureuses.
- Mise au point de la théorie permettant de calculer par PATREC la disponibilité de système complexe en utilisant les formules ci-dessus comme des lois à entrer dans les feuilles des arbres de défaillance.
- Résolution analytique rigoureuse d'un système à 2 composants identiques en redondance parallèle et d'un système dont la durée du test ne peut plus être considérée comme négligeable.
- Résolution analytique rigoureuse d'un système dont l'efficacité du test n'est pas de 100 %.
- Mise au point de version provisoire des codes INDI-I et INDIGO basés sur le modèle ci-dessus. Ces 2 codes ne sont que des études de faisabilité destinés à la vérification des formules mais non destinés à un usage extérieur dans leur état actuel.

- Prochaines étapes :

Cette étude est à l'heure actuelle en suspens. Néanmoins les prochaines étapes devraient être :

- Mise au point de versions définitives des codes INDI-I et INDIGO.
- Affinement du modèle pour la prise en compte du maximum de paramètres permettant de caractériser plus finement le système en attente de fonctionnement périodiquement testés.

Il faut considérer qu'une étude comme celle-ci est ouverte c'est à dire n'a pas de point d'achèvement bien déterminé.

5. RELATION AVEC D'AUTRES ETUDES

- Relation avec l'étude du code PATREC
- Relation avec l'étude menée en collaboration avec l'IRIA et ARMINES fiche n° 144-1-05 (154-1-05)

144-1-04/4112-20 154-1-04/4112-20		14
TITRE UTILISATION DES TECHNIQUES DE MONTE CARLO POUR LES CALCULS DE FIABILITE		Pays CEA/DSN
		Organisme Directeur
TITLE (Anglais) THE USE OF MONTE CARLO TECHNIQUES FOR RELIABILITY CALCULATIONS		Organisme exécuteur CEA/DRE - SERMA
		Responsable
Date de démarrage 1.1.77	Etat actuel en cours	Scientifiques
Date d'achèvement 31.12.79	Dernière mise à jour 12.77	

1. OBJECTIF GENERAL

Amélioration du code PATREC en particulier par les techniques de MONTE CARLO.

2. 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. Diverses applications seront réalisées.

3. INSTALLATION EXPERIMENTALE ET PROGRAMMES : Néant

4. ETAT DE L'ETUDE

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. Par ailleurs on a inclus une possibilité de traitement des modes communs dans un arbre de défaillance.

2) Résultats essentiels

Code de calcul PATREC-MC actuellement opérationnel.

D'autre part, certaines parties de PATREC ont été réagencées et reprogrammées dans le but :

- de simplifier automatiquement l'arbre de défaillances entré en données.
- d'inclure dans le code une nouvelle méthode numérique pour le traitement des modes communs.

Prochaines étapes :

- Version unique de PATREC regroupant la version "déterministe" et la version "Monte-Carlo"

5. RELATION AVEC D'AUTRES ETUDES

- Construction automatique d'arbres de défaillances à partir de diagrammes logiques (Fiche n° 144-1-09 et 154-1-09).

6. DOCUMENTS DE REFERENCE

- 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.

- FEUIDEP - Module du programme PATREC pour la simplification des arbres de défaillances - H. KALLI - rapport SERMA n° 352 - année 1978.

- Nouvelle méthode numérique pour le traitement des modes communs - B. DUCHEMIN, H. KALLI, JM. LANORE, MJ. de VILLENEUVE - Rapport SERMA n° 353 - Année 1978.

- PATREC a computer code for fault tree calculation - A. BLIN; B. DUCHEMIN, etc... rapport DSN n° 235 année 1978.

144-1-05 / 4105-20 154-1-05		14
TITRE DISPONIBILITE DES SYSTEMES EN ATTENTE PERIODIQUEMENT TESTES RECHERCHE D'UN MODELE GENERAL		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) AVAIBILITY OF PERIODICALLY TESTED STAND-BY SYSTEMS GENERAL APPROACH		Organisme exécuteur IRIA ARMINES Responsable
Date de démarrage 1.1.78	Etat actuel en cours	Scientifiques
Date d'achèvement 31.12.79	Dernière mise à jour	

1. OBJECTIF GENERAL

- Evaluation de la disponibilité des systèmes en attente périodiquement testés .
- Optimisation de la politique de test afin d'obtenir la meilleure disponibilité moyenne .

2. OBJECTIFS PARTICULIERS

1) Mise au point d'un modèle permettant :

   - l'évaluation de la disponibilité instantannée prévisionnelle
   - l'évaluation de la disponibilité moyenne
   - l'optimisation de la politique de test .

2) Généralisation du modèle en prenant compte des hypothèses les plus proches possibles de la réalité (durée aléatoire des tests, lois des temps de bon fonctionnement et de réparation non exponentielles ...)

3) Mise au point d'un code de calcul souple d'emploi .

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

Néant

4. ETAT DE L'ETUDE

- Le cas d'un système en attente simple a été traité . Des recouvrements avec d'autres méthodes ont été effectués afin de vérifier ces résultats .

- Le cas d'un système redondant d'ordre 2 est en cours .

- Un code de calcul préliminaire fonctionne .

- Ces premiers résultats sont très encourageants et mettent bien en évidence la puissance et la souplesse de la méthode utilisée .

5. PROCHAINES ETAPES

Les prochaines étapes concernent essentiellement le développement du modèle en vue de traiter des systèmes de plus en plus complexes et d'introduire de plus en plus de paramètres afin de se rapprocher le plus possible des problèmes réellement rencontrés .

6. RELATIONS AVEC D'AUTRES ETUDES

- Cette étude se fait en relation avec la fiche 144-1-01 (154-1-01) qui traite du même sujet par une méthode analytique .

- Cette étude est destinée aux évaluations de la sûreté des systèmes .

144-1 - 06/4105-20		14
TITRE PROBLEME DES EVENEMENTS RARES DANS L'ANALYSE DE FIABILITE DES CENTRALES NUCLEAIRES DE PUISSANCE PWR		Pays FRANCE Organisme Directeur CEA/DSN EdF, FRAMATOME
TITLE (Anglais) Problem of rare events in the reliability analysis of nuclear power plants (PWR)		Organisme exécuteur CEA/DSN - SETS EdF, FRAMATOME Responsable
Date de démarrage 1/77	Etat actuel en cours (prolongée)	Scientifiques
Date d'achèvement 11/78 Prolongé en 1979	Dernière mise à jour 1.2.79	

1. OBJECTIF GENERAL

Etude générale OCDE/CSIN sur le problème des évènements rares dans l'analyse de fiabilité des centrales nucléaires de puissance. Elle a pour but de faire progresser la connaissance des problèmes soulevés par les évènements rares dans l'analyse de fiabilité des Centrales Nucléaires.

Dans ce but, six groupes de travail ont été constitués :

- Analyse et quantification des erreurs humaines,
- Analyse des défaillances de mode commun,
- Collecte et analyse des données,
- Théorie de la décision et études statistiques,
- Techniques de communication interdisciplinaires,
- Evaluation de la fiabilité d'un système réel.

La France (CEA, EdF, Université, FRAMATOME) qui participe aux travaux des différents groupes internationaux, a la responsabilité d'animer le dernier de ces groupes "Evaluation de la fiabilité d'un système réel", groupe constitué uniquement d'experts français.

2. OBJECTIFS PARTICULIERS

Le système choisi est le système d'arrêt d'urgence de FESSENHEIM ; ce système étant sollicité à la suite d'un retrait incontrôlé des grappes de contrôle.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

Néant

4. ETAT DE L'ETUDE

4.1 - Avancement à ce jour

L'étude initiale est terminée. Une prolongation a été décidée sur certains points particuliers (voir "Prochaines étapes"). Un rapport OCDE a été rédigé : SIN DOC (78) 82.

4.2 - Résultats essentiels

La probabilité de trouver le système de protection indisponible à l'occasion d'un test sachant qu'il était en bon état de fonctionnement lors du test précédent (1 mois) est de :

1.10^{-8} (si l'on ne considère que les défaillances aléatoires et indépendantes)

Par contre, si l'on prend en compte les défaillances de mode commun par la méthode dite "du β" et pour β = 0,1 on trouve les résultats suivants :

Valeur ponctuelle	: 2.10^{-5}
Valeur médiane ajustée	: 3.10^{-5}
Limite inférieure à 5 %	: 7.10^{-6}
Limite supérieure à 95 %	: 1.10^{-4}
Facteur d'incertitude	: 4

4.3 - Prochaines étapes

Par décision du CSIN, l'étude se poursuivra en 1979 dans les domaines suivants :

- Fiabilité des mécanismes et barres de sécurité
- Défaillance humaine

5 - RELATIONS AVEC D'AUTRES ETUDES : Néant

6 - DOCUMENT DE REFERENCE :

"Groupe d'experts CSIN sur l'estimation de la fiabilité du système de protection du réacteur FESSENHEIM I - Rapport final"

7 - DEGRE DE DISPONIBILITE : Libre

144-1-07		14
TITRE MODELISATION DU TAUX DE DEFAILLANCE DE VANNES EN FONCTION DE DIFFERENTS PARAMETRES		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) MODELISATION OF THE FAILURE RATE OF VALVES ACCORDING TO DIFFERENT PARAMETERS		Organisme exécuteur CEA/DSN - SETS Responsable
Date de démarrage 1.12.77	Etat actuel Terminé	Scientifiques
Date d'achèvement 31.12.78	Dernière mise à jour 4/78	

1. OBJECTIF GENERAL.

Il s'agit de trouver un modèle de fiabilité pour les vannes permettant de prendre en compte l'influence de différents paramètres .

2. OBJECTIFS PARTICULIERS

1 - A partir des données tirées du fichier de défaillances d'EdF à St-LAURENT-des-EAUX, on s'efforcera d'établir des corrélations entre les caractéristiques des vannes (type, taille, pression, fluide, etc...) et les taux de défaillance observés . Les méthodes de l'analyse statistique (analyse de la variance, etc...) seront utilisées .

2 - Une analyse supplémentaire et approfondie d'une partie du fichier de St-LAURENT-des-EAUX sera nécessaire ?

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES :

Néant

4. ETAT DE L'ETUDE

A partir d'une analyse très fine du fichier et de l'aide de différentes méthodes d'analyses mathématiques et statistiques, nous avons pu mettre à jour des paramètres explicatifs des défaillances, de trouver des catégories les plus influentes pour les défaillances . A titre d'exemple pour la tranche 1 de St-LAURENT-des-EAUX sur 2.057 vannes, du mode pneumatique, température < 100°C, toutes pressions : 69 ont été défaillantes et on peut affirmer que le taux de défaillance décroît exponentiellement dans le temps pour cette catégorie .

5. RELATION AVEC D'AUTRES ETUDES

Néant

6. DOCUMENTS DE REFERENCE

- Valves faile rate modelling from the main parameters

  F. BOUSCATIE, P. FOURCADE, J-P. GEORGIN, C. ROY
  présenté au Congrès de l'International Topical Meeting on
  Nuclear Reactor BRUXELLES Octobre 1978

- Modélisation des taux de défaillance des vannes de St-LAURENT-des-EAUX
  selon les paramètres influents

  Rapport DSN N° 246(f) - F. BOUSCATIE, J-P. GEORGIN, C. ROY, P. FOURCADE .

144-1-09 154-1-09 /4105-20		14
TITRE CONSTRUCTION AUTOMATIQUE D'ARBRES DE DEFAILLANCES A PARTIR DE DIAGRAMMES LOGIQUES		Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) AUTOMATIC CONSTRUCTION OF FAULT TREES DROM LOGIC DIAGRAMS .		Organisme exécuteur CISI Responsable
Date de démarrage 1.6.77	Etat actuel en cours	Scientifiques
Date d'achèvement 31.12.79	Dernière mise à jour 4/78	

1. OBJECTIF GENERAL

L'objectif global de ce projet est la construction automatique des arbres de défaillances pour les analyses de fiabilité de systèmes à partir :

- d'arbres élémentaires stockés en bibliothèques
- de diagrammes logiques
- si possible de diagrammes de fonctionnement .

Trois points justifient ce projet :

1) La construction d'un arbre de fautes est complexe, fastidieuse et sujette à erreurs . La construction automatique libère l'ingénieur de cette tâche ingrate .

2) Le stockage en bibliothèque d'arbres élémentaires diminue considérablement les temps de calculs globaux var les calculs partiels ont été effectués une fois pour toutes .

3) Le problème du passage du diagramme de fonctionnement à l'arbre de fautes admet généralement plusieurs solutions . Le temps de calçul

varie énormément avec la solution choisie; l'étude entreprise permettra d'optimiser le choix de la solution .

2. OBJECTIFS PARTICULIERS

1) - Construction d'une bibliothèque

- Les arbres élémentaires représentés en notation polonaise (endorder traverse) .
- Les listes des structures
- Les lois de probabilités

Cette bibliothèque devra son efficacité (identification, temps d'accès) à l'utilisation du "traitement de liste" ("list processing") .

2) Développement de petits programmes auxiliaires pour augmenter, diminuer, modifier ...etc, le contenu de la bibliothèque .

3) Mise au point du stockage de la bibliothèque sur supports externes (bande, disque ... etc) .

4) Compatibilité du code PATREC avec la partie de la bibliothèque relative aux arbres élémentaires .

5) Compatibilité du code PATREC avec l'entrée "diagrammes logiques" de la bibliothèques .

6) Si possible, compatibilité du code PATREC avec l'entrée "diagramme de fonctionnement" de la bibliothèque .

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMMES

Néant ..

4. ETAT DE L'ETUDE

Avancement et résultats obtenus

Ce projet se déroule en deux étapes pour chaque objectif partiel :

Développement d'une théorie informatique d'une part, écriture du programme d'autre part .

La première de ces deux étapes "développement d'une théorie informatique" est franchie par les objectifs partiels 1 à 5; la seconde "écriture d'un programme" est franchie pour 3 et en voie de l'être pour 1 et 2 (rapports à paraître) .

5. PROCHAINES ETAPES

Pour la partie théorique la prochaine étape est l'objectif 6 .
Pour la partie programmation, on travaillera principalement sur les objectifs 2 à 5 .

6. RELATIONS AVEC D'AUTRES ETUDES

Les résultats de cette étude pourront être utilisés ultérieurement dans le programme PATREC (fiche 144-1-04 et 154-1-04) .

7. DOCUMENTS DE REFERENCE

- Traitement de liste en PL/1: possibilités de traitement et de conservation sur support permanent des arbres de défaillances .
  M. LORIGEON - CISI - 1977 - R 19 .

- Création de bibliothèques de composants en vue de l'évaluation de la fiabilité .
  M. LORIGEON - B.V. KOEN rapport SETSSR N°69

- Programme de gestion de bibliothèques de composants en vue de l'évaluation de la fiabilité d'un système complexe - Notice d'utilisation -
  M. LORIGEON (à paraître) .

144-1-11
154-1-10 14

TITRE Etudes de fiabilité dans le domaine de l'instrumentation nucléaire : essais de fiabilité de composants et de circuits ; développement d'un ensemble de simulation et de calcul de la fiabilité des systèmes complexes.	Pays FRANCE Organisme Directeur CEA/DSN
TITLE (Anglais) Reliability studies in the field of nuclear instrumentation : reliability testing of components and circuits ; development of a device for reliability simulation and calculation.	Organisme exécuteur CEA/DSN/SEESNC/VALDUC Responsable
Date de démarrage: 01/01/73 — Etat actuel: EN COURS	Scientifiques
Date d'achèvement: 31/12/79 — Dernière mise à jour: 12/78	

1. OBJECTIF GENERAL

Améliorer les procédés et les moyens disponibles pour les études de fiabilité des centrales nucléaires, et fournir expérimentalement des données utilisables dans ce type d'études

2. OBJECTIFS PARTICULIERS

1) Etude, réalisation et exploitation d'un ensemble de simulation et de calcul de la fiabilité (ESCAF) dont le but est d'apporter des possibilités nouvelles pour l'aide à la conception et l'étude de la fiabilité des systèmes ; ce programme est susceptible de déboucher sur une commercialisation par cession de licence.

2) Etude de composants :

a) fin du programme d'essais d'endurance de relais de prise d'information du type de ceux qui sont utilisés dans le traitement des températures des coeurs de réacteurs surrégénérateurs ;

b) essais de composants microprocesseurs en environnement nucléaire afin d'étudier les conséquences d'irradiations accidentelles sur leurs performances.

3. INSTALLATIONS EXPERIMENTALES ET PROGRAMME

Un premier prototype d'ensemble de simulation et de calcul de la fiabilité (ESCAF) a été réalisé. Le programme consistera à exploiter les performances du système et à améliorer ses possibilités, tout en mettant ces résultats à profit pour développer et réaliser un prototype industrialisable.

4. ETAT DE L'ETUDE

1) Ensemble ESCAF
Le premier prototype de ce système a été achevé et présenté en démonstration à l'exposition NUCLEX 78 où il a rencontré un vif intérêt. Un certain nombre de circuits ont d'ores et déjà été simulés ce qui a permis d'obtenir des résultats significatifs en ce qui concerne les études de fonctionnement

2) Etude de composants
La campagne d'essais sur des relais bas niveau à contacts mouillés au mercure a été complétée par l'essai d'une plaque de multiplexage à plusieurs relais qui n'a pas révélé de défaillance.

5. PROCHAINES ETAPES

1) Amélioration des possibilités de simulation par le développement d'un ensemble répondant aux critères suivants :

- mise en oeuvre plus simple ;
- puissance de simulation plus importante ;
- possibilité de simuler des automatismes séquentiels ;
- possibilité de couplage à de gros moyens de calculs par l'intermédiaire d'une bande magnétique.

2) Poursuite des essais de fiabilité sur les composants microprocesseurs hors et en fonctionnement.

6. RELATIONS AVEC D'AUTRES ETUDES

NEANT

7. DOCUMENTS DE REFERENCE

- Compte rendu d'essais sur des relais ITT type F 65-2
  R. AGAISSE - C. BERARD - A. CARNINO - R. LELAIT - A. PAVRET - R. QUENEE
  Rapport SETS n° 25/SEESNC N° 120 - Novembre 1973

- Contrôle de l'état de relais bas niveau utilisés dans le traitement des températures du coeur Phénix
  C. BERARD - A. CARNINO - G. JUIF - A. LAVIRON - A. PAVRET - R. QUENEE
  Rapport DSN n° 95 - FEVRIER 1976

- A la demande du C.E.A. - Fiche Technique FT/CPM-FMI/74-78 · CNET
  J.Y. BOULAIRE, IG/ACO   Analyse de relais ILS

- Rapport DSN 181 - MARS 1978
  Essais d'endurance de relais d'acquisition bas niveau (contact Mercure) susceptibles d'équiper le T.R.T.C. des réacteurs rapides
  A. CARNINO - G. JUIF - A. LAVIRON
  C. PAVRET - R. QUENEE - M. SAZERAT

- Rapport SEESNC n° 146 - SETS n° 79
  Suite des essais d'endurance de relais d'acquisition bas niveau (Contact Mercure) susceptibles d'équiper le T.R.T.C. des réacteurs rapides - Essais à 50 Hz
  M. BURGGRAEVE - A. CARNINO - A. LAVIRON - R. QUENEE - M. SAZERAT

- Ensemble de simulation et de Calcul de Fiabilité ESCAF
  Plaquette de présentation en français ou en anglais - Octobre 1978

- Note SEESNC. 78. 528/IF - OCTOBRE 1978
  Compte rendu de mission effectuée à l'exposition NUCLEX 78
  par MM. BERARD - LAVIRON - SIGNORET

8. DEGRE DE DISPONIBILITE

Diffusion interne seulement sauf plaquette publicitaire ESCAF

Une communication sur ESCAF est prévue au Congrès de Fiabilité (BIRMINGHAM - MARS 1979)

	CLASSIFICATION: I4
TITLE (ORIGINAL LANGUAGE):	COUNTRY: ITALY
	SPONSOR:
TITLE (ENGLISH LANGUAGE): MULTIVALUED LOGIC IN RELIABILITY-ORIENTED REPRESENTATION OF NUCLEAR SAFETY SYSTEMS	ORGANISATION: CESNEF
	PROJECT LEADER: S. GARRIBBA
INITIATED: 1976 / COMPLETED: 1980	SCIENTISTS: P. MUSSIO, F. NALDI
STATUS: IN PROGRESS / LAST UPDATING: JUNE 1979	

1. General Aim. Purpose of the research is to improve the representation of (nuclear safety) systems. In general, assessment of nuclear risk is based upon the construction and analysis of a representation of the system made of binary logical trees (fault trees). It is realized that a crucial question is how good a representation of the system is the fault tree used. The adoption of multivalued logic seems to offer means to consider into the representation more than the traditional "good-bad" states since components and system show one of several "behaviors" each of which may be of importance. Behaviors are defined in terms of couples consisting of internal state and state transition function. In this respect the notion of behavior appears to be more general than the notion of state which results to be a particularization.

2. Particular objectives. Two types of problems must be afforded in the analysis of MultiValued Logical Trees (MVLTs). First, is the problem of finding the path sets or combinations of primary events (behaviors and terminal variables) by which a specified behavior of the entire system may occur. Second, is the problem of estimating the probability associated with the behavior of the entire system.

4. Project status. Given a system, a formal method has been developed which allows to construct its representation in terms of MVLT. Different values are made to correspond with different behaviors of the system and its components. The subsequent analysis allows to transform the MVLT into an equivalent forest of binary trees.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
MULTIVALUED LOGIC IN RELIABILITY-ORIENTED REPRESENTATION OF NUCLEAR SAFETY SYSTEMS	I4

5. Next steps. Developments are envisaged for the theory. The possibility will be explored of introducing representations based upon a logic endowed by a continuity of values. On the other hand, applications will be pursued by the consideration of a full scale practical case.

7. Reference documents. S. Garribba, E. Guagnini, P. Mussio, F. Naldi, G. Volta, "Use of Multiple-Valued Logical Tree to Extend Reliability Analysis of Nuclear Plants", pp. 564-578 in Proceedings of ENS/ANS Int. Topical Meeting on Nuclear Power Reactor Safety, Brussels, October 16-19, 1978.

	CLASSIFICATION: 14
TITLE (ORIGINAL LANGUAGE): *DIAGNOSTICA E SICUREZZA TRAMITE ANALISI DI RUMORE*	COUNTRY: *Italy*
	SPONSOR: *CNEN*
TITLE (ENGLISH LANGUAGE): *DIAGNOSTICS AND SAFETY VIA NOISE ANALYSIS*	ORGANISATION: *CNEN*
	PROJECT LEADER: *CPS*
INITIATED: *1976* / COMPLETED: *1983*	SCIENTISTS: *A. Colombino, N. Pacilio, G. Sena*
STATUS: *in progress* / LAST UPDATING: *July 1979*	

1 GENERAL AIM : Development of physical models, mathematical and statistical methods, experimental techniques for monitoring signals from sensors placed in reactors and power plants in order to identify procedures for diagnostics and safety

2 PARTICULAR OBJECTIVES: (1) Identification of models for interpreting measurements of multizone reactor neutron noise
(2) Set-up of a stochastic heat-transfer model with one- and two-phase coolant flow
(3) Production of computational codes for analyzing data coming from various types of sensors

3 EXPERIMENTAL FACILITIES: research reactor RANA at CSN Casaccia, CNEN, Rome
research reactor RITMO at CSN Casaccia, CNEN, Rome

4 PROJECT STATUS: Progress to date

A theory for the interpretation of experimental data from neutron noise analysis techniques

Essential results

Analytical solutions in terms of factorial cumulants, auto- and cross-correlation functions of the number of neutron counts from several detectors

5 NEXT STEPS: - Advanced research in stochastic heat transfer for multizone nuclear systems
- Investigation over the existence of eigenvalues in stochastic models of power reactors

39.507

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
DIAGNOSTICS AND SAFETY VIA NOISE ANALYSIS	*14*

6 RELATION TO OTHER PROJECTS: 3.8 LMFBR

7 REFERENCE DOCUMENTS:

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

(2) A. COLOMBINO, N.PACILIO, G.SENA - Further developments toward a unified theory of reactor neutron noise analysis techniques - Annals of Nuclear Energy (1978) in print

(3) N.PACILIO, A.COLOMBINO, V.M.JORIO, R.MOSIELLO, F.NORELLI - The analysis of reactor noise: measuring statistical fluctations in nuclear systems - Advances in Nuclear Science and Technology, vol. 12, 67-134 (1978) Plenum Publishing Co.

(4) A. COLOMBINO, N.PACILIO, G.SENA - The role of factorial cumulants in reactor neutron noise theory - Annals of Nuclear Energy (1979) in print

(5) A. COLOMBINO, D.FIORE, N.PACILIO - An introduction to one-phase heat transfer : deterministic and probabilistic formulae - CNEN Report RT/FI (79)

(6) N. PACILIO, A.COLOMBINO, D.FIORE - Temperature noise in one-phase heat transfer: stochastic models - submitted to Annals of Nuclear Energy, Pergamon Press

8. DEGREE OF AVAILABILITY: free

please contact

A. Colombino, N. Pacilio or G. Sena
RIT-FIS CNEN
CSN Casaccia C.P. 2400
00100 ROME (ITALY)

	CLASSIFICATION: 14
TITLE (ORIGINAL LANGUAGE): Sviluppo di una procedura avanzata per l'analisi di sicurezza di un FWR seguendo l'approccio probabilistico	COUNTRY: ITALY SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Development of an advanced procedure for the PWR safety analysis following the probabilist approach	ORGANISATION: SOPREN PROJECT LEADER: G.P. Pozzi
INITIATED: 1976 — COMPLETED: 1981	SCIENTISTS: P. Sala, E. Mascellani, U. Monasterolo
STATUS: in progress — LAST UPDATING: September 1979	

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 perform 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

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Development of an advanced procedure for the PWR safety analysis following the probabilist approach	14

4. Project status

For each accident to be analyzed, a detailed "event tree" has been developed in order to establish: a) the different paths 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.
A reliability analysis of the Safety Injection System of an advanced Westinghouse PWR plant was performed.

5. Next steps

a) The digital programs necessary for the analysis of each accident path will be set up (deterministic programs)

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. (see NSRI 1977)

7) Reference documents

1) Preliminary Safety Analysis Report of CLEOPATRA Loop to be inserted in the ESSOR Reactor, Report prepared by FIAT-TTG, October 1977

2) Reliability Analysis of 3000-XL Reactor Safety Injection System and comparison wiyh the Reliability of Westinghouse 3 loop standard Safety Injection System, SOPREN Report AEFP 012, March 1979

8) Degree of availability

To a limited extent

9) Budget, personnel involved

8 engineers for 5 years

	CLASSIFICATION: 14
TITLE (ORIGINAL LANGUAGE): Studi di affidabilità	COUNTRY: Italy SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Reliability studies	ORGANISATION: CNEN PROJECT LEADER: G. Tomassetti
INITIATED: 1974 | COMPLETED:	SCIENTISTS: S. Pia
STATUS: In progress | LAST UPDATING: July 1979	

General aim: Experimental reliability studies with particular attention to problems of transferability of data (organisational and geographical context effect).

Particular objectives:

1) Analysis of structure and workability of data banks in relation to the results obtained from direct data collection work.
2) Reliability of emergency diesel generator systems in Italy; in particular analysis of the influence of organisational context on reliability.

Project status; next steps:

Data from a sodium loop at Casaccia Nuclear Center were elaborated.
Data from the operating experience of the four emergency diesel generators system of Casaccia Nuclear Center were elaborated.
A study for the collection and analysis of operating experience data from 100 emergency diesel generator systems in the industrial and miscellaneous field in Italy is now under progress.
Data collected refer to five main arguments: technical parameters of main plant and its auxiliaries; installation and building arrangement; maintenance organisation; specifications and procedure of periodic surveillance testing; operating experience (with particular emphasis to cause and mode of failure).
The study is carried out by visits to the plants and direct enquiries (final compilation of a questionnaire) with people charged with different managerial and executive functions.
Particular attention is given to problems connected with common mode phenomena and their causes.

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Reliability studies	14

Reference documents:

1) CNEN RT/ING(79)5 - Alcuni problemi relativi al sistema di alimentazione elettrica del CSN-Casaccia.
   Parte I - Storia e dati di esercizio dei diesel elettrogeni d'emergenza (S. Pia, A. Taglioni, G. Tomassetti)
   Parte II - Organizzazione ed esperienza di manutenzione (F. Dolci, S. Pia, A. Taglioni)

2) CNEN RT/ING(78)19 - Ricerca sull'esperienza operativa dei diesel elettrogeni d'emergenza in Italia. Prima fase: problemi di composizione del campione e metodo (S. Pia, G. Tomassetti)

Degree of availability:

Open.
Contact persons: G. Tomassetti, S. Pia, CNEN, CSN Casaccia, C.P. 2400, I-00100 Roma.

	CLASSIFICATION: 10-14
TITLE (ORIGINAL LANGUAGE): ANALISI STATISTICA DI SEGNALI	COUNTRY: ITALY
	SPONSOR: C.N.E.N.
TITLE (ENGLISH LANGUAGE): STATISTICAL ANALYSIS OF SIGNALS	ORGANISATION: C.N.E.N.
	PROJECT LEADER: A.FEDERICO
INITIATED: 1966 / COMPLETED:	SCIENTISTS:
STATUS: In progress / LAST UPDATING: June 1979	

	CLASSIFICATION: LWR 14
TITLE (ORIGINAL LANGUAGE): European Reliability Data System (ERDS)	COUNTRY: C.E.C.
	SPONSOR: Commission of the European Communities
TITLE (ENGLISH LANGUAGE):	ORGANISATION: J.R.C. ISPRA
	PROJECT LEADER: G. MANCINI

INITIATED: 1976	COMPLETED: December 1979	SCIENTISTS: Al, Balestreri, Capobianchi, A.G. Colombo, Luisi
STATUS: In progress	LAST UPDATING: March 1979	

1. General aim: Feasibility Study for an European Reliability Data Bank: data pooling on LWR at an European level.

2. Particular Objectives
   - Feasibility study for centralized event data system aimed at the collection, storing and processing of raw data on LWR component operation coming from national data systems.
   - Updating of the Reliability Parameter Data Bank, with data from litterature and from compaigns on operating reactors.
   - Feasibility study for a centralized Abnormal Occurrences Reporting System where information on safety related occurrences of operating reactors in Europe could be stored.

4. Project Status

4.1. Progress to date

The pilot experiment for the component event data bank has been consolidated in its informatic structure; reference coding system are being developed.

The Reliability Parameter Data Bank has been fed with litterature data.

An analysis of the information contained in existing abnormal occurrences reports shemes has been achieved; the LER tape has been obtained through agreement with NRC.

59.507

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4.2. Essential Results

The compatibility between event data on components derived from various national systems (EdF, SRS, GRS-RWE, ENEL, NPRDS) has been in principle ascertained by the help of the above mentioned pilot experiment and by the development of reference classifications for systems, component characteristics, environment, duty, and failures. A detailed study of the information contained in the LER system is under way in order to extract information on failure dependency, on human error, on the incident initiating events and on the incident sequences.

5. Next steps

- Final implementation of the pilot experiment with insertion of transcoding from national data to the reference european data.
- Insertion in the pilot experiment of large sample of data from national systems.
- Implementation of various possible enquiries for family of components.
- Issue of a final report on the feasibility study for a component Event Data Bank.
- Implementation of a pilot informatic system for retrieval of information contained in the LER system and in other Abnormal Occurrences Reporting schemes.
- Detailed comparative analysis of reliability parameters data from various sources for some major components (valves, pumps etc..)
- Collection of productivity data of the European Nuclear Power Plant.

7. Reference Documents

- J.R.C. Ispra Safety Programme Semiannual Progress Reports (II semester 1978, I semester 1979)
- G. MANCINI, G. VOLTA " The European Reliability Data System for LWR. Aspects concerning structural and mechanical components" contribution to "In service data reporting and analysis" Symposium, ASME Winter annual meeting, December 78, S. Francisco

Netherlands Energy Research Foundation (ECN)		CLASSIFICATION : 14
TITLE : Faalweg en faalkansvoorspellingen van reaktorsystemen.		**COUNTRY: THE 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** : June 1979	**SCIENTISTS :** -
STATUS : terminated	**COMPLETED** : Summer 1978	

General aim

To predict the probability of failure of systems on basis of possible failure modes and component reliability, taking into account accidental as well as systematic failures due to external causes or inherent faults of components.

Particular objectives

The first stage of the program consists of:
- making an inventory of the general available methods of reliability analysis and "data banks" on failure rates
- establishing a standard procedure to collect data on failures in nuclear power stations in The Netherlands
- developing procedures to evaluate the reliability of reactor systems for specific cases applicable to Dutch power reactors.

Experimental facilities: None

Project status

a. An assembly of computercodes called "RECAL" has been made operational. Main calculations (including repair) are:
   * Determination of MCS
   * System, MCS and component characteristics
   * Ranking of MCS and components according to their importance (several criteria available)
   * Confidence intervals for the system unavailability
b. A draft report, titled "System Reliability Analysis", has been written, containing:
   * The theoretical background of "RECAL"
   * How to use "RECAL"
   * An introduction into the state of the art of Data Bank Systems for LWR

Next steps

The commission of the Ministry of Social Affairs will be terminated. A new commission is foreseen which will deal with the addition in the RECAL codes of:
* Phased missions
* Dependency between basic events

Relations with other projects: -

Reference documents

Reliability analysis of systems;
Part I : Theory , ECN-78-142
Part II : Annexes , ECN-78-143
Terpstra K , April 1978

Netherlands Energy Research Foundation (ECN)	CLASSIFICATION : 14

Degree of availability

The reports are written in the Dutch language. They are submitted to the Ministry of Social Affairs, Postbus 69, Voorburg.

Budget: -

Personnel: 0.2 manyear

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

N.V. KEMA	CLASSIFICATION : 14
TITLE : Faalkans-analyse met behulp van gebeurtenissen- en foutenbomen	COUNTRY: THE NETHERLANDS
	SPONSOR : KEMA
	ORGANIZATION : KEMA
TITLE (ENGLISH LANGUAGE): Failure analysis by application of event- and fault trees	PROJECTLEADER : R.W. van Otterloo
INITIATED : - / LAST UPDATING : 1978	SCIENTISTS : R.W. van Otterloo
STATUS : - / COMPLETED : 1977	

General aim

To analyse new systems or changes in existing systems. And by application of this technique to make decisions concerning these new systems or changes in existing systems in the right way.

Particular objectives

- Analysis of the unavailability of reactor safety systems and of the unreliability of reactor systems.
- Analysis of the probability of different groups of radioactive releases of a nuclear power reactor.

Experimental facilities

Not applicable.

Project status

Methods and computer codes have been compared.

Next steps

Not applicable

Relation to other projects

This project was started to do the "Risk analysis of the fuel cycle in the Netherlands" (RASIN-study) which was finished in June 1975.

Reference documents

See 6.
Several applications of this method are written in the Dutch and English language.

Degree of availability

Through the organization KEMA

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

TITLE COMPONENT RELIABILITY THEORY				CLASSIFICATION 14
COUNTRY UK	SPONSOR SRD		ORGANISATION SRD	PROJECT LEADER G W PARRY D H WORLEDGE
INITIATED MARCH 1977		COMPLETED DECEMBER 1977		SCIENTISTS G W PARRY
STATUS COMPLETED		LAST UPDATING		

General Aim

To calculate availability, the distribution of number of failures, the distribution of downtime, etc for a component subject to failure and repair with an aim to understanding the fundamentals of reliability theory.

Particular Objectives

To investigate the effect of time structure of failure and repair processes on the calculations.

Experimental Facilities and Programme

QWA.

Progress to Date

Have derived a unified treatment of different time structures.

Next Steps

To extend to simple systems.

Relation with Other Projects

Back-up to SRD fault tree work.

References

Nuc Eng & Des 45 (1978) p 261, and ibid p 271
SRD R90 and 95; and SRD R113 (Nuc Eng & Des to be published).

Degree of Availability

Freely

	CLASSIFICATION: 14
TITLE (ORIGINAL LANGUAGE): RELIABILITY THEORY	COUNTRY: UK
	SPONSOR: SRD
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD UKAEA
	PROJECT LEADER: G W PARRY D H WORLEDGE
INITIATED: MARCH 1977 / COMPLETED: JULY 1978	SCIENTISTS:
STATUS: COMPLETED / LAST UPDATING:	

GENERAL AIM

To explore the fundamental aspects of the models for components and systems used in system reliability calculations.

PARTICULAR OBJECTIVES

To investigate the effect of time structure of failure and repair processes.

PROGRESS TO DATE

Unified treatment of the different time structures commonly used and a classification of the Vesely-Maschland debate.

REFERENCES

Nucl Eng & Design 45 (1978) p 261 and ibid p 271
49 (1978) p 295
and SRD R143 (to be published in the NATO AS1 on Synthesis and Analysis Methods for Safety and Reliability studies Sogesta, Urbino, Italy 2-14 July 1978).

	CLASSIFICATION: 14
TITLE (ORIGINAL LANGUAGE): CONFIDENCE LIMITS IN COMPLEX SAFETY ARGUMENTS	COUNTRY: UK
	SPONSOR: EEC
TITLE (ENGLISH LANGUAGE):	ORGANISATION: UKAEA
	PROJECT LEADER: G W PARRY
INITIATED: OCTOBER 1978 / COMPLETED: -	SCIENTISTS: F BRISCOE, P WINTER
STATUS: ON GOING / LAST UPDATING: MAY 1979	

GENERAL AIM

To discuss the confidence that can be attached to the results of a risk assessment exercise using the fault/event tree methodology.

PARTICULAR OBJECTIVES

To identify the sources of uncertainty that arise and to investigate and develop statistical techniques for quantifying these uncertainties.

	CLASSIFICATION: 14
TITLE (ORIGINAL LANGUAGE): CONFIDENCE LIMITS IN PROBABILISTIC SAFETY ARGUMENTS	COUNTRY: UK
	SPONSOR: SRD
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD UKAEA
	PROJECT LEADER: D H WORLEDGE

INITIATED: JULY 1975	COMPLETED: JANUARY 1978	SCIENTISTS: G W PARRY, P SHAW
STATUS:	LAST UPDATING: MAY 1979	

GENERAL AIM

To improve the status of probabilistic safety arguments by deciding how an overall figure of confidence can be placed on the final answer.

PARTICULAR OBJECTIVES

The investigation of the meaning of combining statements on tolerance which are derived from limited data.

PROGRESS TO DATE

For normal populations we have phenomenological proofs of a method for deriving lower bounds on the confidence on the tolerance figure for a sum of variables.

NEXT STEPS

To extend the work to other distributions and to construct a mathematical proof.

RELATION WITH OTHER PROJECTS

Relevance to event tree work.

REFERENCES

SRD R99 "The Use and Interpretation of Confidence and Tolerance Intervals in Safety Analysis"
SRD R129 "The Tolerance-Confidence Relationship and Safety Analysis".

DEGREE OF AVAILABILITY

Freely

	CLASSIFICATION: 14
TITLE (ORIGINAL LANGUAGE): INDIVIDUAL RISK - A COMPILATION OF RECENT BRITISH DATA	COUNTRY: UK
	SPONSOR: SRD/HSE
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD, UKAEA
	PROJECT LEADER: J R BEATTIE
INITIATED: SEPT 1977 / COMPLETED: 1 JULY 1978	SCIENTISTS: D R GRIST
STATUS: / LAST UPDATING: MAY 1979	

GENERAL AIM

To compile data on risk from recent British population and mortality statistics, and derive risk data eg risk of death, as function of age, natural causes, accidents, violence, location etc.

PARTICULAR OBJECTIVES

As above.

EXPERIMENTAL FACILITIES AND PROGRAMME

None

PROGRESS TO DATE

Essentially complete but a limited amount of further work has been undertaken on certain aspects which will be published later.

RELATION WITH OTHER PROJECTS

Related to R F Griffiths and L S Fryer's work on multiple fatality accidents - See SRD R110.

REFERENCES

Published by D R Grist, under above title, as SRD R125, August 1978.

DEGREE OF AVAILABILITY

Freely

	CLASSIFICATION: 14
TITLE (ORIGINAL LANGUAGE): DEVELOPMENT AND APPLICATION OF DEVELOPMENT/ STATISTICAL TECHNIQUES TO PWR SAFETY DEVELOPMENTS (GSD 4.1)	COUNTRY: UK SPONSOR: UKAEA
TITLE (ENGLISH LANGUAGE):	ORGANISATION: SRD/UKAEA PROJECT LEADER: RNH McMILLAN

INITIATED: DECEMBER 1978	COMPLETED:	SCIENTISTS/ENGINEER: RF WHITE, AG CANNON
STATUS:	LAST UPDATING: MAY 1979	

BACKGROUND

For many years SRD has advocated the use of probabilistic methods for quantifying reactor risks. These methods have been developed and applied in a comprehensive manner to PWRs by Rasmussen (Ref 1). A recent evaluation by Lewis of the Rasmussen report, although critical of some aspects including some of the statistical techniques and data, nevertheless concluded that the fault-tree/event tree methodology is sound and should be developed and used more widely (Ref 2).

It is thus anticipated that a significant proportion of PWR safety justification/assessment will be based upon such methods. However, there are a number of important aspects which require further research and development. Significant amongst these is the identification and treatment of Common Mode Failure (CMF) mechanisms since such mechanisms can invalidate predictions of reliability which are based upon the assumption of random unrelated events. Whilst it is clear that final quantitative analyses must await the acquisition of a defined design, there is at present sufficient knowledge of Generic PWR aspects to allow meaningful research and development of potentially useful techniques. A closely related aspect is ATWS which has received considerable attention. Here event sequence methodology coupled with examination of operating statistics is valuable in ensuring a comprehensive identification of anticipated transients.

A common source of criticism of the method is the use of inadequate or questionable fault data in the generation of failure frequency estimates - or alternatively improper allowance for the uncertainty of such data. Improvements in both the data base and in the evaluation of sensitivity to data uncertainty is required. Finally where - as is frequently the case - data is lacking the validity of using subjective (judgemental) estimates and the necessary safeguards in its use warrants further research.

SRD are already working on methods development in some of these areas. However, it is important that the work be extended to cover PWR applications.

OBJECTIVES

1. Review of probabilistic methodology used in existing PWR documentation eg SNUPPS, RESAR, WASH-1400, NUREG/CR-0400, and provide summary report including review of shortcomings, treatment of CMF, etc.
2. To apply and if necessary develop CMF identification techniques to PWRs. Report potential common mode failures. Report techniques for defence against common modes.
3. Review initiators of anticipated transients without scram (ATWS)
   (a) review of mechanisms leading to ATWS
   (b) evaluate relevant operating data.
4. To extend the existing data bank for PWR conditions. To validate and recommend adoption of existing data (eg Rasmussen) for Generic PWR safety justification/assessment.
5. To review and, if appropriate, develop subjective probability evaluation methods in areas relevant to PWRs.

		CLASSIFICATION:
TITLE (ORIGINAL LANGUAGE):		COUNTRY:
		SPONSOR:
TITLE (ENGLISH LANGUAGE):		ORGANISATION:
		PROJECT LEADER:
INITIATED:	COMPLETED:	SCIENTISTS:
STATUS:	LAST UPDATING:	

FACILITIES

(a) Analytical Codes — PREP KITT
ALMONA

(b) Other Facilities — National Centre for Systems Reliability Data Bank.

OVERSEAS COLLABORATION

RELATED DOCUMENTS

1. WASH-1400 (NUREG-75/014) US NRC Reactor Safety Study

2. NUREG/CR-0400 Risk Assessment Review Group Report to the USNRC

TITLE Error propagation by sampling techniques in the use of large computer codes.				CLASSIFICATION 14
COUNTRY UK	SPONSOR SRD		ORGANISATION SRD	PROJECT LEADER H L BROWN
INITIATED October 1977		COMPLETED July 1978		SCIENTISTS K J MARSHALL
STATUS In progress		LAST UPDATING 3 May 1978		

1. General Aim

To develop methods whereby large, deterministic model, computer codes can be economically used to generate a probability distribution for an output given the probability distributions of the inputs.

2. Particular objectives

To compare the efficiencies, under various conditions, of 3 numerical sampling schemes: Monte Carlo; factional stratified; and latin hypercube. To write a computer code enabling any of these schemes to be used with a large computer code (eg RELAP, CALROS, FRAX), for arbitrary distributions of input variables.

3. Experimental facilities

None available.

4. Project Status

1. Progress to date

Initial comparisons of sampling efficiencies for a nonlinear combination of inputs has been made. Comparison for a linear combination of normally distributed inputs is nearing completion.

2. Essential results

Factorial stratified and latin hypercube sampling are far more efficient than Monte Carlo techniques; latin hypercube sampling may have a significant advantage over both other methods.

5. Next Steps

A working and validated sampling code is to be written up and issued. It will be called SARCASM (Safety and Reliability Code Analysis Sampling Methods).

It is intended that the work should be extended to cover the fitting of response surfaces by sampling techniques, and the derivation of sensitivity coefficients. Complementary computer codes for doing this are to be developed.

6. Relation to other projects and codes

None at present but the work should be useable in any project using large codes to study stochastic-variable problems.

7. <u>Source of availability</u>

Will soon be available. Contact Dr M L Brown, UKAEA, SRD, Wigshaw Lane, Culcheth, Cheshire.

Classification 5/14/16/18

Title 1	Prediction of Reactor Releases	Country	UK
		Sponsor	CEGB
Title 2	Building Entrainment Effects	Organisation	Research Divn. Berkeley Nuclear Laboratori
Initiated	1974	Completed	
Status	Continuing	Last updating	

Project Leader:
Dr. H. F. Macdonald
Dr. B. M. Wheatley

15. INTERRELATION BETWEEN REACTOR PLANT AND OPERATING PERSONNEL

Classification 15

Title 1	COUNTRY Denmark
Operatør studier	SPONSOR Risø National Laboratory
	ORGANIZATION Risø National Laboratory
Title 2	**Project leader:**
Study of the Process Operator	L.P. Goodstein
Initiated: Approx 1966 - in its present form 1973 — Completed:	Scientists:
Status: progressing — Last updating: 1979	L.P. Goodstein J. Rasmussen M. Lind

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 on control room design is currently underway. For information on available reports, contact L.P.Goodstein.

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.

J. Rasmussen & J.R. Taylor
Notes on Human Factors Problems in Process Plant Reliability & Safety Prediction - Risø-M-1894 1976.

Rasmussen, Jens: Notes on Diagnostic Strategies in Process Plant Environment. Riso-M-1983, 1978, 36 pp.

Rasmussen, Jens: Man as a System Component - to be published in "Man-Computer Research" H.Smith and T.Green (eds) Academic Press

Rasmussen, Jens: Notes on Human Error Analysis and Prediction. Risø-M-2139, 1978, 53 pp.

<u>8. Available</u> **from the library at Risø.**

144-2-01/4600-50 154-2-01/4600-50	15-1

TITRE PARAMETRES HUMAINS DE LA SURETE NUCLEAIRE		Pays FRANCE
		Organisme Directeur CEA/DgCS
TITLE (Anglais) HUMAN FACTORS IN NUCLEAR SAFETY		Organisme exécuteur CEA/DSN - SETS
		Responsable
Date de démarrage 1.1.78	Etat actuel en cours	Scientifiques
Date d'achèvement 31.12.81	Dernière mise à jour 1.79	

1. OBJECTIF GENERAL

Lorsqu'on étudie la causalité élémentaire des accidents ou incidents nucléaires, on constate que les composantes d'origine humaine sont majoritaires. Il est donc capital de s'efforcer d'étudier scientifiquement l'organisation humaine, 3ème composante d'une installation nucléaire.

2. OBJECTIFS PARTICULIERS

a) A partir du fichier d'incidents des réacteurs français (72 à 76) recherche approfondie des causes d'erreurs humaines avec évolution des facteurs de risques des erreurs humaines en fonction de l'âge de la Centrale.
cf. rapport de Melle. BOUTIN - DSN/SETSSR/CONF/CEA/78-02 "Essais de recherche des causes d'erreurs humaines à partir des comptes rendus des incidents survenus sur PHENIX" (CONFIDENTIEL CEA)

b) Analyse dans le cadre de l'O.C.D.E. groupe Evènements Rares, de l'exécution d'un test du système d'arrêt d'urgence de la Centrale de BUGEY 2 par les méthodes SWAIN et LEPLAT.
cf. Rapport SETS n° 85 (Août 78) "Utilisation de la méthode THERP pour l'étude d'une chaîne de contrôle de Bugey" T. ROBERT.

cf. Task force on Problems of Rare Events in the Reliability analysis of Nuclear Power Plants - June 78 } CSNI report n° 51.

c) Analyse de poste de maintenance d'un réacteur graphite-gaz de SAINT-LAURENT-DES EAUX
cf. CNAM - Laboratoire de Physiologie du Travail et d'Ergonomie rapport n° 58 "Le contrôle du système d'arrêt d'urgence dans une Centrale Nucléaire - Nov. 78"

d) Contrat CEA-DSN n° SA 7293

- "Recherche bibliographique" qui traite des facteurs humains de la sûreté (exploitation d'incidents avec identification des facteurs humains) et des études ergonomiques pouvant intéresser la sûreté.

- "Rapport préliminaire" esquisse d'une méthodologie pour l'identification des causes des erreurs humaines

- "Sélection des incidents avec erreurs humaines" analyse critique de la constitution du fichier DSN (SAINT-LAURENT 1 et 2).

- Méthode d'analyse des incidents avec dysfonctionnements humains opératoires

e) Contrat C.E.A. n° BC-1481
Etude des laboratoires du Département de Génie Radioactif à FONTENAY-AUX-ROSES (Bât.18)
"Etude sur l'installation de la Centrale des Monts d'Arrée (EL.4)

Il s'agit de dégager, à partir de ces 2 études, les relations existant entre certaines caractéristiques d'une installation nucléaire et la manière dont la sûreté/sécurité est gérée dans l'installation.

Les trois axes principaux sont :

- Y a t-il adaptation équipement/matière ?
- Comment la sûreté dans l'installation nucléaire est-elle analysée et ressentie par le personnel ?
- Quels sont les rôles des règles de sécurité ?

cf. Rapport à paraître.

Tous ces documents sont disponibles exceptés ceux qui sont notés "à paraître" et "confidentiels".

3. <u>INSTALLATIONS EXPERIMENTALES ET PROGRAMMES</u> : Néant

4. <u>ETAT DE L'ETUDE</u>

a) A partir de la "méthode d'analyse des incidents avec erreur humaine" (à paraître) pour rechercher les causes des dysfonctionnements humains, va être entreprise l'étude de 2 incidents survenus à MARCOULE ces dernières années.

b) Démarrage d'une étude sur le simulateur du BUGEY pour effectuer la saisie d'informations du comportement des opérateurs en situations dites normale et accidentelle.

c) Etude des incidents américains d'origine humaine et établir un parallèle avec ce qui pourrait se passer chez nous et analyser les conséquences au niveau système.

d) Etude de postes et d'incidents sur l'usine de séparation isotopique de PIERRELATTE.

5. RELATIONS AVEC D'AUTRES ETUDES

- Fiche n° 144-1-06/4105-20 : Problème des évènements rares dans l'analyse de fiabilité des Centrales Nucléaires de puissance PWR.

Prochaines étapes :

- simulateur
- incidents américain
- étude de 2 incidents PHENIX
- étude de postes et d'incidents sur l'usine de séparation isotopique de PIERRELATTE.

16. ENVIRONMENTAL PROTECTION

		Classification	5/14/16/18
Title 1	Prediction of Reactor Releases	Country	UK
		Sponsor	CEGB
Title 2	Building Entrainment Effects	Organisation	Research Divn. Berkeley Nuclear Laboratori
Initiated	1974	Completed	
Status	Continuing	Last updating	
Project leaders	Dr. H. F. Macdonald Dr. B. M. Wheatley		

17. NUCLEAR ACCIDENT RECOVERY AND DECOMMISSIONING

		Classification	17.1
Title 1	Development of Improved Reagents for Chemical Decontamination	Country	U.K.
		Sponsor	C.E.G.B.
Title 2		Organisation	Berkeley Nuclear Laboratories
Initiated 1976	Completed	Project Leaders	
Status Continuing	Last updating	Dr. C.J. Wood Dr. T. Swan	

1. General Aim

To develop improved methods and reagents for in service and decommissioning decontamination of reactor primary circuits so that Man Rem requirements may be kept to a minimum.

2. Particular Objective

To find new reagents to clean contaminated metal surfaces prior to man access for maintenance or repair.

3. Experimental Facilities and Programme

The majority of the work is conducted on a laboratory scale, but successful developments will be continued in a small recirculation loop in a Cobalt-60 irradiation facility. Experiments are being conducted to determine the chemical parameters which influence the operation of cleaning solutions, e.g. pH redox potential, chelating strength, etc..

4. Project Status

Recent experiments indicate that single-electron reducing agents are effective for the dissolution of Heamatite and Nickel Ferrite. This work is a collaborative project with EPRI.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1978 - 31.12.1978	17.2	RS 300

Vorhaben/Project Title	Land/Country
Specification of Conditions of a Nuclear Power Plant with a PWR Following a LOCA for Purposes of Studying the Ensuing Decontamination and Transport Problems Spezifizierung des Anlagezustandes eines DWR nach einem Kühlmittelverluststörfall für die Untersuchung der daraus folgenden Dekontaminations- und Transportprobleme	FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: NIS Frankfurt Abt. Reaktoranlagen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.10.1977	30.6.1978	A. Gasch
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	31.12.1978	

1. General Aim

Based on the NIS-study RS 155 "Result analysis of LOCA's for the dismantling of nuclear power plants" the post LOCA system condition of a 1300 MW PWR should be researched at which new boundary conditions are defined. The research results should supply input data for the RS 156 project.

2. Particular Objectives

In this study the conditions of nuclear components and equipments in some neuralgic regions of the reactor building are specified, which are important for the post LOCA decontamination and shipping problems.

3. Research Programme

3.1 Description of the course of LOCA
3.2 Investigation of radioactivity
3.3 Local dispersion of the radioactive nuclides
3.4 Local dose rates at component surfaces
3.5 Chemical conditions in the plant
3.6 Mechanical conditions of components and equipments

4. Experimental Facilities, Computer Codes

The radioactivity of the core was calculated by the computer code ORIGEN. The calculations of readioactivity produced by activation in the core components was done by NIS code AKAT 2. Further calculations of the local dispersion of radioactive nuclides were done by the well known computer code CORRAL.

5. Progress to Date

The study was finished in June 1978.

6. Results

Assuming a double-ended rupture of the hot line in the piping chamber and a fuel assembly cladding tube damage of 10% corresponding to the licensing guidelines currently valid for the release of iodine, the nuclide-specific distribution of the radioactivity in reference chambers in the containment is determined with the CORRAL computer programme. The dose rates resulting from the nuclide-specific distribution of the radioactivity are calculated for 1 year, 10 years, and 30 years after the accident. The dominant radionuclides here are Cs 137/Ba 137 m and Cs 134.

In all, about $7 \cdot 10^8$ Ci of fission and corrosion products are released into the containment during the accident.

As a result of the radioactivity released, a maximum dose rate of about 3 150 rem/h must be expected in the chamber where the rupture occured one year after the accident.
After 30 years, the maximum dose rate is still 350 rem/h. In the chambers located further from the chamber where the rupture occured, dose rates in the order of magnitude of 24 rem/h are expected 1 year after the accident. A possible reduction in the radiation as a result of decontamination measures was not considered here.

In addition, the corrosion of the fuel assemblies still located in the reactor pressure vessel, of building structures, and on the surface of system components was studied, and the possible pentration of radioactivity into damaged concrete surfaces was determined.

The adhesion of the radionuclides to the contaminated surfaces due to their element-specific behaviour is also shown here so that suitable decontamination processes and materials can be developed and applied.

7. Next Steps

-

8. Relations with Other Projects

-

9. References

-

10. Degree of Availability

-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.1978 - 31.12.1978	17.3	RS 236

Vorhaben/Project Title	
Sprengtechnische Zerstörung von radioaktiven Großkomponenten bei der Stillegung von Kernkraftwerken Controlled-Blasting Demolition of Radioactive Primary-Loop Components of Decommissioned Nuclear Power Plants	Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Messerschmitt-Bölkow-Blohm GmbH, D-8898 Schrobenhausen

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.10.1977	31.08.1978	P. Größler
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Completed	December, 1978	DM 363,222.--

1. General Aim

Removal of nuclear power plants after an estimated useful service time of 40 years.

2. Particular Objective

Applicability of blasting means and explosive devices in dismantling large radioactive components.

3. Research Program

3.1 Analysis into the radioactive reference components
3.2 Examination of explosive dismantling means
3.3 Placement of the explosive devices
3.4 Auxiliary tools and equipment
3.5 Component decontamination
3.6 Comparison of different dismantling techniques

4. Experimental Facilities

All explosive charge manufacturing facilities and a well-equipped blasting range for charges of up to 25 lb. of TNT were available.

5. Progress to Date

5.1 Analysis Into the Radioactive Reference Components

The size, weight, material properties and specific location features of the reference components, i. e., primary loop and biological shield of a Biblis-B-type pressurized-water reactor, have been compiled in tabular form. Expected component contact dose levels and the resulting permissible occupational exposure times - based on the present German Radiation Safety

Regulations - have been estimated.

5.2 Examination of Explosive Dismantling Means

A large number of experiments have been conducted to determine the cutting performance of specific linear, circular, and conical shaped charges in steel and concrete targets. Moreover, tests were made to blast air- and water-filled steel pipes (representing 1:50 and 1:10 scale reactor vessel models) by detonating explosive charges inside. These pipes had been previously notched on their inside to ensure regular fragmentation.

Tests with explosive sheets detonated in contact with both sides of concrete blocks were made to strip the concrete off the reinforcement, either locally for severing into smaller blocks, or entirely with the aim of a separate disposal of steel and concrete.

5.3 Placement of the Explosive Devices

A removal sequence of the primary loops of a Biblis-B-type PWR has been worked out. The proposed procedures of dismantling the individual components and the type, size, and placement of the required explosive devices have been described. Moreover, the expected blast effects inside the reactor building as well as the occupational exposures to radiation have been estimated.

5.4 Auxiliary Tools and Equipment

The machines, equipment, and staff required for dismantling the primary loop components have been listed in context with the description of the proposed dismantling procedures.

5.5 Component Decontamination

Tests have been made to remove an inner surface layer of steel pipes by externally applied explosive sheets that were hoped to produce large scabs.

5.6 Comparison of Different Dismantling Techniques

The explosive techniques have been juxtaposed to the more commonly used cutting techniques by various criteria.

6. Results

6.1 Analysis Into the Radioactive Reference Components

The tables with the component data were used in devising dismantling procedures and auxiliary tools needed for applying the explosive devices. Estimates of the radiation levels and of the permitted exposure times indicate that 1 year after reactor shutdown, the decontaminated primary loop components, except for the reactor vessel, permit reasonable working times; the reactor vessel would require remote handling.

6.2 Examination of the Explosive Dismantling Means

Linear and circular shaped charges appear suitable for severing primary piping and other steel components. The largest pipe tested was a 20-inch outer diameter steel #52 (approx. SAE 1020) pipe, 1.18 in. thick, which can be neatly severed in one shot by a 4.4-lb. circular shaped charge applied from the inside. Real primary piping (35 in. outer diameter, 2 in. wall thickness) is expected to require some 30 lb. of explosive for one complete circumferential cut.

The tests have shown that linear shaped charges will also prove useful in severing primary piping and other steel structures from the outside. Even in cutting the reactor vessel (from inside), the use of linear shaped charges has not been ruled out, although it is suggested that in this case the vessel be cut only into two or three pieces that can be handled by the polar crane. Cutting into more sections on-site may be un-economical if the vessel head were to be closed for every shot; a perforating cut 1 ft. long would require 50 to 60 lb. of high explosive, and not more than 3 ft. should be done in one shot. However, one possibility that would have to be tested is to place a large number of charges at one time and to detonate them sequentially and at spacings such that one detonation does not destroy other charges; naturally, exhaust and filter operations would have to be performed between ever so many shots.

Dismantling the reactor vessel from inside by detonating blast charges appears not feasible since a very large amount of explosive would be needed and fragmentation is neither regular nor safe.

The tests with explosive sheets applied to reinforced concrete blocks showed that concrete and rebar can be separated quite well in this way, giving small-size debris. The amount of explosive, however, is much larger than in ordinary drill-pattern delay blasting. The explosive-sheet technique may be useful in removing the biological shield since occupational radiation exposures can be kept low.

6.3 Placement of the Explosive Devices

Several mechanical fixtures for putting linear and circular shaped charges into the required position, mainly at the steam generators, primary piping, and (remotely) inside the reactor vessel have been designed.

6.4 Auxiliary Tools and Equipment

The staff and tools lists can be used is estimating the dismantling costs as well as the number of teams that have to be exchanged owing to radiation exposure.

6.5 Component Decontamination

Tests with explosive sheets wrapped around ordinary steel pipes showed that detonation does not produce any significant scabbing; the result was merely a reduction in pipe diameter. Separation of the plating from the backing in plated piping is perhaps possible; however, no tests have been made with plated pipes.

6.6 Comparison of Different Dismantling Techniques

In view of the short radiation exposure times, cutting by linear and circular shaped charges appears to be the most advantageous technique for dismantling primary and other piping as well as for cutting up certain other heavy structures (coolant pumps etc.) into shipable sizes. However, the spread of contamination by the blasting operations is unknown, hence, either this must be examined beforehand or an amount of costs must be allowed for to consider increased safety and decontamination activities after blasting, which may or may not become necessary.

Cutting the reactor vessel of a PWR by means of linear shaped charges may turn out to be prospective if other techniques fail because of large material thickness.

Dismantling the biological shield by means of explosive sheets has one essential advantage over conventional blasting: no drilling is required, which means less radiation exposure to the personnell.

7. Next Steps

None (work has been completed).

8. Relation to Other Projects

None.

9. References

/1/ J. Herzog et al., Dekontaminations- und Transportprobleme bei der altersbedingten Stillegung und bei der Beseitigung von Störfall-Folgen in Kernkraftwerken; Rept. #312 of NUKEM Corp., BMFT code # RS156, Hanau (1977).

/2/ W. Buschmann, Die Beseitigung von stillgelegten Kernkraftwerken mit Druckwasserreaktoren am Beispiel des Kernkraftwerkes Obrigheim; Thesis, Ruhr-Universität, Bochum (1976).

/3/ A. Gasch and G. Lörcher, Quantitative Mengenstromanalyse für radioaktive Abfälle, die bei der Stillegung von Kernkraftwerken entstehen; Rept. #277 of NIS Consulting Engnrs., BMFT code # RS220, Frankfurt/M. (1977).

/4/ W. J. Manion and T. S. LaGuardia, An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternatives, NESP Rept. AIF/NESP-009-009SR, Atomic Industrial Forum Inc., Washington (1976).

/5/ Structures to Resist the Effects of Accidental Explosions, Dept. of the Army TM 5-1300, June 1969; Amendments as of March, 1971.

/6/ Biblis-B Nuclear Power Facility design drafts, made available thru Kraftwerk Union Inc., Offenbach/M.

/7/ Private communications from Gesellschaft für Strahlen- und Umweltforschung, Neuherberg/Munich; Research Reactor Garching/Munich; Obrigheim Nuclear Power Station; Unterweser Nuclear Power Station.

10. Availability of Reports

Messerschmitt-Bölkow-Blohm GmbH., D-8898 Schrobenhausen, Federal Republic of Germany.

Berichtszeitraum/Period 01.01.1978-31.12.1978	Klassifikation/Classification 17.3	Kennzeichen/Project Number RS 274
Vorhaben/Project Title Untersuchungen über die Anwendbarkeit und Leistungsfähigkeit des Pulverbrenn- und Plasmaschneidens für das Zerteilen von Kernkraftwerkskomponenten. Investigation on the Applicability and Efficiency of Powder Cutting and Plasma Cutting for Dismantling Components of Nuclear Power Stations.		Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Salzgitter AG
Arbeitsbeginn/Initiated 01.09.1977	Arbeitsende/Completed 31.12.1979	Leiter des Vorhabens/Project Leader Deipenau
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Under the Federal Government research program for reactor safety, it is necessary to investigate dismantling methods for activated and contaminated components in connection with the decommissioning and disassembly of nuclear power stations.

2. Particular Objectives

This project aims to check the applicability of flame cutting by powder and plasma for cutting up nuclear reactor components. If a specific dismantling case arises, then a decision can be taken as to which method may be used for taking a component to pieces and what conditions must exist for this.
The great number of components in a nuclear power station makes it necessary to collect and catalogue the available technical information on the individual components. This collection serves as a basis for a typification in order to reduce the number of different dismantling tasks. Accompanying dust and aerosol measurements are used to obtain reference values for the design of efficient removal by suction.

3. Research Program

3.1 Drawing up a catalogue of the activated and contaminated components within the containment shell of a nuclear power station.

3.2 Experiments for demonstrating the efficiency of the powder and plasma cutting processes.

3.2.1 Powder cutting of reinforced concrete.

3.2.2 Powder cutting of steel.

3.1.3 Plasma cutting in the atmosphere.

3.2.4 Plasma cutting under overpressure in air and water.

3.2.5 Plasma cutting in water.

3.3 Assessment of the powder and plasma cutting processes in relation to the dismantling of reactor components.

4. Experimental Facilities

re 3.2.1 & 3.2.2 The experimental station for powder cutting is equipped with two cutting outfits of different capacities. Propane is used as fuel gas. The cutting process is controlled from a control cabin and observed via television cameras installed in this cabin. An existing ventilation system can remove up to 60,000 m^3/h of combustion air from the cutting cubicle. The dust particles are conveyed via a bypass to a dust collector.

re 3.2.3 The experimental station for plasma cutting in atmosphere consists of a chamber in which the burner manipulator is installed; the speed of this manipulator can be adjusted and reproduced. The manipulation of the burner can be observed via a television camera. By means of an induced draught fan the cutting waste gases (about 2,000 m^3/h) can be extracted and analyzed in the bypass. With the sources of electric power available a loading of the burner up to 1,000 A. can be attained by parallel connection (modification as against the 1977 Annual Report).

re 3.2.4 For plasma cutting under overpressure conditions in air and water, a pressure chamber has been provided permitting cutting tests up to pressures of 2.5 Bars. Burner manipulation will be external. Sources of electricity: see 3.2.3.

re 3.2.5 The plasma cutting tests in water depths up to 1 m will be carried out in a water basin which permits burner and television camera to be manipulated.
Sources of electricity: see 3.2.3.

5. Progress to Date

re 3.1 Work on drawing up the catalogue of activated and contaminated components has not yet been completed.

re 3.2.1 & 3.2.2 Construction of the experimental test floor has been completed. Cutting experiments have been carried out.

re 3.2.3 Construction of the test stand has been completed. Cutting tests have been initiated.

re 3.2.4 & 3.2.5 Construction of the pressure chamber is almost complete. Testing of the parallel connection of the sources of electricity up to a burner loading of about 650 A. has been concluded.

6. Results

Test results are available but are not yet representative.

7. Next Steps

re 3.1 Continuation of work.

re 3.2.1 & 3.2.2 Systematic cutting tests.

re 3.2.3 Systematic cutting tests.

re 3.2.4 Testing of pressure chamber for overpressure tests in air and water.

re 3.2.5 Continuation of underwater cutting tests using one or two burners.

8. Relation with other Projects

–

9. References

9.1 R.J. Smith, G.J. Konzek, W.E. Kennedy, Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station. U.S. Nuclear Regulatory Commission (1978) NUREGICR 0130, Vol. 1

9.2 International Symposium on the Decommissioning of Nuclear Facilities. International Atomic Energy Agency Wien 1978, JAEA - SM - 234.

10. Degree of Availability of the Reports

re 9.1 ZAED Leopoldshafen

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.06.78 - 31.12.78	17.3	3S 341

Vorhaben/Project Title	
Sealing of Radioactive Contaminated Components of Nuclear Power Reactors Versiegelung readioaktiv kontaminierter Komponenten aus Kernkraftwerken	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: NUKEM GmbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.06.78	31.12.80	Dr. G. Wagner
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

Improvement of safety, minimizing of secondary waste volume, and saving of time and labour obtained by application of a sealing technique which is an alternative to decontamination.

2. Particular Objectives

Experimental examination of a sealing concept for radioactive contaminated components in nuclear power stations in order to reduce contamination and incorporation during repairs, maintenance, replacement, transport, and storage.

3. Research Program

3.1 Selection of components

- compilation of the relevant components in nuclear power stations
- description according to materials, dimensions, and activities
- estimation of the frequency for repairs and maintenance
- definition of the representative components.

3.2 Coating of inactive samples

- definition of the materials and dimensions of the samples
- selection of resins corresponding to the properties desirable,
- coating of the samples by variation of coating material, thickness of the layer, and coating technique.

3.3 Test methods
- selection of the test methods,
- classification of the radiation resistance according to the literature,
- experimental testing of the samples and classification of sealing quality.

3.4 Sealing of inactive components
- Construction of a sealing facility for small components,
- sealing of small components,
- concept for large components,
- model tests with the sealing of large components.

3.5 Sealing of active components.

4. Experimental Facilities/Computer Codes

ref. 3.4: construction of a sealing facility for small components will be projected in 1980

5. Progress to Date

ref. 3.1:
- compilation of the relevant components of a PWR and BWR has been performed,
- description according to materials and dimensions has been completed according to the literature available,
- informations about activities and frequencies of repairing and maintenance interventions are insufficient and will be completed as far as possible in 1979,
- definition of the representative components thus could not yet be done.

ref. 3.2:
- preliminary definition of basic materials for the inactive coating is done,
- comparison of the resins corresponding to their properties has been started.

ref. 3.3: Literature research with reference to test methods has been started. Modifying the original program this point has been started earlier because the dimensions of the samples (ref. 3.2) depend on test methods.

ref. 3.4) Refering to the programm no activities were started in 1978.
ref. 3.5)

6. Results

The data selected for a PWR (1300 MW-type) and for a BWR (860 MW-type) were scheduled. The materials 1.4308, 1.4408, 1.4439 and 1.4550 were elected for the inactive tests.

Neither any evaluation or adaption of computer codes nor any application for an invention took place. We did not receive knowledge of results obtained by other people that could have an effect on the general aim of the programm.

7. Next Steps

ref. 3.1: This part of the programm shall be completed during the first quarter of 1979 having further discussions with power reactor operators.

ref. 3.2: Work will be continued defining the dimensions of the samples and defining the coating resins.

ref. 3.3: Test methods will be defined and applied.

ref. 3.4) Referring to the program these parts shall be started in
ref. 3.5) 1980.

8. Relation with Other Projects

- RS 156: Decontamination and Transportation Problems related to the Decommissioning of Nuclear Power Plants and the Elimination of Failure Consequences - NUKEM - 312
- Untersuchung der radioaktiven Festabfälle bei Reparaturen und Stillegung von kerntechnischen Anlagen - NIS - 176.

9. References

-

10. Degree of Availability of the Reports.

-

18. FUEL CYCLE

	CLASSIFICATION: 18
TITLE (ORIGINAL LANGUAGE): Inglobamento in cemento di: a) rifiuti radioattivi a media e bassa attività; b) zeoliti argentate.	COUNTRY: Italy
	SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Evaluation of Concrete as a Matrix for Incorporation of: a) Low and Medium Level Radioactive Wastes; b) Spent Silver-coated Zeolites from Nuclear Plants	ORGANISATION: CNEN
	PROJECT LEADER: a) G. De Angelis b) G. Beone

INITIATED:	COMPLETED:	SCIENTISTS:
a) 1979 (present phase) b) 1978		
STATUS: In progress	LAST UPDATING: July 1979	

General Aim

The 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. Concentrates from evaporation processes and spent ion-exchange resins are incorporated into cement. The following characteristics are taken into account: leachability, thermal stability, radiation damage, compressive strenght and weathering resistance.

Particular Objectives

Achievement of concrete specimens which can be carried and stored safely. Evaluation of the effects of spent silver-coated zeolites on the quality of cement incorporation products.

Project Status

a) Thermal stability and compressive strenght have been evaluated.
b) Water absorption, compressive strenght, flame and heat resistance have been evaluated.

Next Steps

Investigations concerning the other characteristics previously indicated.

Reference Documents

G. Beone, I. Di Stefano, "Inglobamento in cemento di zeoliti argentate. Parte I - Studio delle variabili e primi risultati", CNEN Technical report (in press).

Contact Person

G. Beone, CNEN, CSN Casaccia, CP 2400, I-00100 Roma.

	CLASSIFICATION: 18
TITLE (ORIGINAL LANGUAGE): Ricerca e sviluppo su contenitori di trasporto per materiale fissile	COUNTRY: Italy SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Research and development on shipping casks for fissile material	ORGANISATION: (°) PROJECT LEADER: G. Tomassetti
INITIATED: 1974 (present phase) | COMPLETED: STATUS: In progress | LAST UPDATING: July 1979	SCIENTISTS:

(°) CNEN, AGIP Nucleare, Nuovo Pignone, Pisa University, in cooperation with ENEL and FIAT Nucleare.

General Aim:

Research and development on radioactive material shipping casks.

Particular Objectives:

The project is supporting actions to allow design, testing and licensing in Italy of a spent fuel shipping cask.

Project Status:

A first set of research contracts (CNEN-AGIP Nucleare, CNEN-FIAT Nucleare, CNEN-Pisa University) is completed. A testing facility for drop tests and fire tests is in operation (Pisa University). Some components have been developed and tested.

Next Steps:

Tests for evaluating particular constructive solutions of a cask designed for Italian needs and large model tests for licensing and approval of such a cask.

Relation to Other Projects:

18 (Pisa University).

Availability:

Information available to a certain extent. Contact person: G. Tomassetti, CNEN, CSN Casaccia, CP 2400, I-00100 Roma.

	CLASSIFICATION: 18
TITLE (ORIGINAL LANGUAGE): Tecniche non distruttive per l'analisi quantitativa di miscele di isotopi fissili	COUNTRY: Italy SPONSOR: ENEL
TITLE (ENGLISH LANGUAGE): Nondestructive methods for quantitative analysis of fissile isotopes mixtures	ORGANISATION: Politecnico di Milano (*) PROJECT LEADER: G. Sandrelli - ENEL; V. Sangiust - Politecnico

INITIATED:	COMPLETED:	SCIENTISTS:
1975	December 1979	A. Cesana - Politecnico G. Sandrelli - ENEL
STATUS: In progress	LAST UPDATING: July 1979	V. Sangiust - Politecnico M. Terrani - Politecnico

(*) Istituto di Ingegneria Nucleare - CESNEF.

1. General aim

The nondestructive methods for the assay of nuclear materials have up today provided worse results than those expected on the ground of the quoted errors on the related nuclear data.

So, our research has been concentrated on the careful analysis of the parameters conditioning those techniques, with the aim of selecting the most promising methods for an accurate quantitative evaluation of some fissile isotopes.

2. Particular objectives

The methods investigated in this research are the gamma-passive analysis, the analysis by delayed neutron emission following fission by fast neutrons and in some cases the activation analysis by thermal neutrons.

These methods are widely employed as relative ones, i.e. making use of standards. We thought it worthwhile studying their applicability as absolute ones, with the additional purpose of performing a selection and an evaluation of some important nuclear data relating to the fissile isotopes, such as gamma-ray branchings, neutron cross sections and delayed neutron yields.

3. Experimental facilities

All the tests have been carried out at the Nuclear Engineering Institute of the Polytechnique of Milan.

The analyzed samples consisted of few milligrams of ^{232}Th, $^{233/235/238}$U, ^{237}Np, $^{238/239/240/241}$ Pu and ^{241}Am, encapsulated as oxide powder in small vials of stainless steel or zircaloy.

Gamma-ray countings were performed by a 60 cm^3 coaxial Ge-Li detector and by a 0.5 cm^3 Ge(I) detector, both coupled to a 4096 channels analyzer.

Thermal neutron irradiations were performed in a well thermalized flux ($r\sqrt{\frac{T}{T_o}} \leq 0.001$), in the thermal column of the L54 water boiler (50 kW) reactor operating at CESNEF ($nv_o = 1.35 \cdot 10^9$ n/cm^{2}sec).

Fast irradiations were performed at the surface of the reactor core in a flux

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TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Nondestructive methods for quantitative analysis of fissile isotopes mixtures	18

filtered by a 5 cm thick B_4C shield, the spectrum of which had been carefully determined by activation technique using fifteen suitable selected reactions. The total fast flux intensity was $1.2 \cdot 10^{11}$ n/cm^{2}sec.

The neutron counter was a BF_3 detector embedded in paraffin and a pneumatic system was used to transfer the samples from the irradiation to the counting position.

4. Project status

All the experimental tests have been completed. The measurements have pointed out the need of employing two different techniques at least to perform accurate fissile material assays. The results can be resumed as it follows:

- the γ -ray analysis can give very good results, particularly in the assay of small fissile masses. This tecnique can be at present considered as the best one for assaying fissile isotopes mixtures, when the activities of the different isotopes are comparable. However the considerable complexity of the γ -ray spectra makes sophisticated unfolding procedures indispensable;
- the analysis by delayed neutron emission following fission is easier and quicker than the γ -ray analysis. However in general a high degree of accuracy in absolute measurements is not achievable because of incertainties in the fast spectrum evaluation of the flux used for irradiating, and in some cases because of the low reliability of the values of the delayed neutron yields reported in literature. (e.g. for ^{233}U, ^{237}Np, $^{240/241}$Pu, ^{241}Am).
  Therefore for these isotopes the absolute yields of the first five groups of delayed neutrons were reevaluated. Moreover the absolute yields of the same five groups were estimated for ^{238}Pu, for which as far as we know no value is reported in literature.

5. Next steps

The final part of the data unfolding is running and it is planned it will be completed during 1979.

6. Relation with other projects

These nuclear assay methods have been investigated in the framework of a program sponsored by ENEL, whose main object is the setting-up of a quick experimental technique for neutron spectra evaluation by neutron emission following fission measurements in fissile monitors.

7. Reference documents

7.1 A. Foglio Para, G. Sandrelli, M. Terrani - "Una procedura semplificata per il best fit di picchi gaussiani di rivelatori nucleari" - En. Nucleare, vol. 25 (1978) pp. 24-30

7.2 A. Cesana, G. Sandrelli, V. Sangiust, M. Terrani - "Nondestructive quantitative analysis of actinide samples" - 1st Annual Symposium on Safeguards and Nuclear Material Management, Bruxelles 25-27 April 1979 (sponsored by ESARDA).

8. Degree of availability

References are full available.

(G. Sandrelli, ENEL-DSR, Centro di Ricerca Termica e Nucleare,
Bastioni di Porta Volta, 10 - 20121 Milano (Italy).

	CLASSIFICATION: 18
TITLE (ORIGINAL LANGUAGE): Sicurezza del trasporto del materiale radioattivo	COUNTRY: ITALY
	SPONSOR: CNEN
TITLE (ENGLISH LANGUAGE): Safety problems in design of packaging for transportation of radioactive materials	ORGANISATION: University of Pisa
	PROJECT LEADER: Giuseppe FORASASSI
INITIATED: (present phase) 1977 / COMPLETED: 1980	SCIENTISTS: Donato AQUARO
STATUS: in progress / LAST UPDATING: June 1979	

1. General aim
   Study of safety problems in design, construction and use of transport cask for radioactive materials.

2. Particular objectives
   Present research activity covers the following items:
   a) determination of energy adsorption characteristics of several types of steel structures;
   b) testing of the capabilities of available computer codes pertaining to transport casks thermal behaviour in normal as well as accident conditions.
   c) testing of several transport packagings models and components in order to support the design.

3. Experimental facilities and programme
   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
   Tests were performed on:
   - steel honeycomb
   - straight and circular fins
   - tube clusters.

5. Next steps
   Up to date study and test programs include:

TITLE (ENGLISH LANGUAGE):	CLASSIFICATION:
Safety problems in design of packaging for transportation of radioactive materials	18

- Study of containment problems as well as performances of gaskets;
- Studies and dynamic tests on bolt and tie-downs;
- Comparison of computer calculations with results of thermal tests on cask models and components.

6. Relation to other projects

A large part of the above mentioned research activity is related to the development of the preliminary design of a spent fuel shipping cask. This activity is connected to a set of R & D programs (partially already completed) carried out by Agip Nucleare, Fiat Nucleare and Pisa University under the supervision, coordination and financial support of CNEN.

7. Reference documents

1- G. ELETTI, G. FORASASSI
"Italian Testing Station Following IAEA Regulation; CF6, a Fissile Class II Type B Packaging"
4th International Symposium on Packaging and Transportation of Radioactive Materials, 22-27 Sept. 1974.

2- P. CITTI, G. FORASASSI, S. REALE
"Problemi di similitudine connessi con due prove standard per contenitore per trasporto di materiale radioattivo"
Istituto di Impianti Nucleari, Università di Pisa, RL 171(74).

3- G. FORASASSI, B. GUERRINI
"Comportamento di una struttura metallica a celle esagonali in prove di simulazione di 'fuel cask drop'"
Istituto di Impianti Nucleari, Università di Pisa, RL 204(75).

4- G. FORASASSI
"Assorbimento di energia meccanica mediante superfici alettate"
Istituto di Impianti Nucleari, Università di Pisa, RL 209(75).

5- G. FORASASSI
"Prove d'urto su superfici alettate in acciaio inossidabile"
IV Convegno AIAS, Ott. 1976, Roma.

6- C. FALOCI et alii
"The Transport of Irradiated Fuel Elements within the Italian Nuclear Power Program"
5th Int. Symposium on Packaging and Transportation of Radioactive Materials, May 7-12-1978, Las Vegas.

7- G. FORASASSI
"Evaluation and Design Oriented Experiences on Steel Shock Absorbing Structures"
ENS/ANS, Topical Meeting on Nuclear Power Reactor Safety, October 16-19/1978, Brussels, Belgium.

8. Degree of availability: All the reports are available but the numbers 2, 3, 4 which are subjected to authorization of the sponsoring organization (Ist.Impianti Nucleari, Università, V.Pietrasalvi 2, Pisa)

N.V. KEMA	CLASSIFICATION : 18
TITLE : Ontwikkeling van een computercode voor het schrijven van een éénduidige en optimale herladingsprocedure	COUNTRY: THE NETHERLANDS
	SPONSOR : KEMA
	ORGANIZATION : KEMA
TITLE (ENGLISH LANGUAGE): Development of a computercode which writes an unambiguous and optimum reload procedure	PROJECTLEADER : K.P. Termaat
INITIATED : Jan. 1976 — LAST UPDATING : June 1978	SCIENTISTS : K.P. Termaat
STATUS : see below — COMPLETED : 1978	

General aim

The programme "RELOAD" will provide the operator of a nuclear power plant with a stepwise written optimum reload procedure. The programme is based on octant symmetric reload patterns, however non symmetric fuel assembly movements can be included.

Particular objectives

The programme is developed to be applied in the Dodewaard nuclear power plant. The objective is to minimize the number of refuelling steps and the quantity of time to reload the core. The programme will prevent errors which can possibly be made by handwriting the elaborous procedure, especially when a large number of reload elements, shuffle elements, dummy elements and inspection elements are involved in one reload scheme.

Experimental facilities

Not applicable.

Project status

The programme "RELOAD" has succesfully been applied in the Dodewaard nuclear power plant during the 8th refuelling outage. Minor modifications will be introduced. "RELOAD" will also be used for the next refuelling outage in 1979.

Next steps

Complete development and test with previous handwritten reload procedures.

Relation to other projects

Physics reload scheme, fuel inspection programme, fuel test programme.

Reference documents

Three reports (in Dutch) are available through the organization N.V. KEMA.

Degree of availability

Through the organization KEMA.

			Classification	5/14/16/18
Title 1	Prediction of Reactor Releases		Country	UK
			Sponsor	CEGB
Title 2	Building Entrainment Effects		Organisation	Research Divn. Berkeley Nuclear Laboratori
Initiated	1974	Completed	Project Leaders	Dr. H. F. Macdonald Dr. B. M. Wheatley
Status	Continuing	Last updating		

19. ECONOMICS OF SAFETY

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1978 - 31.12.1978	20	RS 0123 A
Vorhaben/Project Title HDR-Sicherheitsprogramm HDR Safety Program		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum (KFK) Karlsruhe
Arbeitsbeginn/Initiated 1.4.1974	Arbeitsende/Completed 1981	Leiter des Vorhabens/Project Leader Müller-Dietsche
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

On the project RS 0123 A it is reported within the framework of RS 0123 B, see there.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1978 - 31.12.1978	20	RS 0123 B

Vorhaben/Project Title	
Durchführung des HDR-Sicherheitsprogramms Investigations on safety technology at the HDR-plant to improve the state of knowledge concerning the properties and structurel behavior of LWR systems and components.	Land/Country FRG Fordernde Institution/Sponsor BMFT Auftragnehmer/Contractor Kernforschungszentrum Karlsruhe (KFK)

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
January 1, 1976	1979	Müller-Dietsche
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	December 1978	

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. Particulars Objectives

The HDR safety program is divided into five Specific Projects (SP):

HDR Safety Program

Spec.Proj. 1000	Spec. Proj. 2000	Spec. Proj. 3000	Spec.Proj. 4000	Spec.Proj. 5000
Non-destructive Tests	Pressure vessel and Piping Studies	Blowdown Studies	Earthquake Studies	Leak rate Studies

These five Specific Projects were determined due to the main objectives presently pursued under the Reactor Safety Research Program of the Ministry after harmonization with the involved authorities.

3. Experimental Facilities and Program

3.1 Experimental Facilities

The HDR with 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 full pressure containment and the reactor pressure vessel. No nuclear operation is envisaged for the experiments. The experimental conditions will be obtained with an electrically heated boiler.

3.2 Program

3.2.1 Specific Project 1000: Non-destructive Testing

Non-destructive testing serve for controlling the reactor pressure vessel and the piping under pressure tests and blowdown-tests and for testing in-service inspection systems under development. The following objectives will be investigated:

- Assessment of record of the initial conditions
- Detection of failure initiation and failure propagation
- Testing of inspection methods; proof of the suitability of new test methods
- Comparative evaluation of non-destructive testing systems, such as ultrasonic impulse testing, magnetic particle testing, penetration testing, eddy current method, ultrasonic scattering, acoustic emission, acoustic holography, radiography.

3.2 Specific Project 2000: Pressure and Piping Studies

Studies of pressure vessel and piping schould be a further step of intensifying and securing the basic knowledge of their safe design. The most important aspect is a study of the effective safety margin above the design ratings of components. The following objectives are pursued:

- Material properties at the beginning of the experiments.
- Component behavior under special loading conditions.
  e.g., operating conditions, blowdown, thermalshock, earthquake.

- Component behavior under specific weakening - fabricational and operational defects and crack formations - to determine boundary strengths or critical defects.
- reliability of the employed methods of calculation.

3.2.3 Specific Project 3000: Blowdown Studies

In the blowdown studies it is possible to measure radial stresses of reactor pressure vessel internals, containment structures and full scale valves by means of reactor typical impulse and mass-flow excitations in their realistic dimensions. The studies comprise the following objectives:

- Stress behavior of the reactor pressure vessel internals and containment structures
- Behavior of full scale valves under mass-flow excitation.
- Testing and further development of the various fluid and structural dynamic computer codes.

3.2.4 Specific Project 4000 : Earthquake Studies

Vibration-studies serve to improve and correct the basic knowledge about the safe design of building structures and pipe-systems with respect to the effects of earthquake. This implies the following objectives:

- Verfication 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.

4. Project Status

4.1 Progress to Date

Central Preparation of the HDR-Experiments

The experimental test loop incorporating electric heating which is used, to simulate approximate operating conditions prior to loss of coolant experiments was sucessfully tested and then modified to incorporate a two loop heating-cooling circuit.
The central data acquisition system for the approximately 450 quick and 150 slow response measuring stations and the computer supported documentation system following the experience gained in the first tests, was extended to include fast on and off line data processing. Important components within the containment-crane, electronic installations - were made blowdown proof.

Specific Project 1000: Non-destructive Tests

The full volume ultra sonic baseline testing of the reactor pressure vessel was supplemented by manual testing of the nozzles and thereby completed. A program was started for theoretical and experimental analyses of crack growth in conjuction with the thermal fatigue experiments (SP 2000) on RPV-nozzles.

Specific Project 2000 : RPV and Piping Studies

The loading of the material in the pipe-system was experimentally analysed under the blowdown tests with feed water check valve (SRV 200) and steam isolation valve (DIV) (S.SP 3000). Laboratory experiments on an HDR-Piping system section were prepared and the first static loading test with strain mesurements performed. In preparation of the main thermal fatigue tests, pretests on plates under large thermal strains were completed and pretesting on a nozzle in a model vessel was initiated.

Specific Project 3000 : Blowdown Studies

So that the exact initial conditions for later blowdown tests with the corebarrel installed could be determined, temperature stratification experiments were conducted in the HDR plant. Moreover, the prototype development of exact differential and absolute pressure transducers for such testing was completed. Two blowdown test series were prepared and completed:

5 experiments on a feed water che⁻˙ valve:

Nominal data : 70 bar, 220/280^{o} C PWR conditions
rupture nozzle NW 450

Variable data: valve damping
weakly damped to undamped

4 additional experiments on a steam isolation valve:

Nominal data : 70 bar, 280^{o} C rupture nozzle NW 450
valve closure delay of 4 seconds, 2 phase flow

Variable data: water level in RPV 4,6,8 m
through put 200 and 400 %.

Accompanying the experiments were both fluid and valve dynamics calculations. The final evaluation and analysis of the results is not yet completed. The installation of equipment for the next two sets of blowdown experiments

- containment steam blowdown
- feed water check valve test NW 350

has been initiated.

Specific Project 4000 : Earthquake Studies

The comparison of the calculation describing the vibrational behavior of the building, piping systems and pressure vessel with the experimental results from low level excitation tests was completed and published. In addition two experimental groups:

- shaker and snapback testing to high excitation levels, and
- explosive testing to mid excitation levels

were outlined and preparatory work initiated. The tests should be conducted in 1979.

Specific Project 5000 : Leak Rate Studies

The investigation on the plant in a cold state and its evaluation were completed.

4.2 Essential results

The results of the baseline NDT investigation have shown that no noticible flaws exist in any of the RPV basemetal or its welds (SP 1000). The loading in the piping system during blowdown testing produced maximum stresses which only slightly exceeded the materials' yield strength (SP 2000).

The temperature stratification pretesting on the HDR with corebarrel (SP 3000) indicated that sufficiently stabile and reproducible conditions can be achieved for later blowdown testing.
It was shown quite clearly in the blowdown testing on a feed water check valve that closure of the valve can produce pressure peaks (water hammer) of up to 330 bar for a nominal system pressure of 70 bar. However, both valve damping and temperature of the fluid affect the pressure peak formation strongly; with optimal valve damping the water hammer can be greatly reduced.
The second blowdown test series on a steam isolation vlave indicated that increasing water level within the RPV does indeed lead to increasing water fraction in the blowdown flow, but that still no noticible pressure shock is observed at the valve.
Within the final evaluation process for the low level earthquake testing (SP 4000), criteria were formulated to rank the calculational methods, particularly model type, as to their areas of optimum applicability. For example, a simple beam model performed equally well as a time consuming and costly FE model for building calculations.

5. Next Steps

Several points which will be included in further investigations are:
- Blowdown experiments with BWR conditions to investigate containment processes (Cont Dampf)
- blowdown experiments with a feed water check valve of NW 350 (SRV 350)
- snapback testing to determine the natural frequencies and vibrational modes of the corebarrel
- mid and high level earthquake testing using shaker, snapback and explosive excitation.

Moreover, thermal fatigue testing on a RPV nozzle corner as well as jet load studies are planned to supplement the upcoming blowdown testing.
For all experiments both calculations and comparisons of these calculations with experimental measurements will be performed.

6. Relation with other Projects

1. Investigation of the Phenomena Involved in the Depressurization of Water Cooled Reactors — RS 16/2
2. Investigation of the Phenomena Occurring within a Multicompartment Containment after Rupture of the Primary Cooling Circuit in Water Cooled Reactors. — RS 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 Spectroscopy in Ultrasonic NDT. — RS 54
6. Investigations on a Continuously Operating System for Crack Growth Surveillance in Pressure Vessels, Part IV: Further Development of Acoustic Emission with Regard to the Application at the Reactor. — RS 31/3

   Nondestructive Inservice Inspection for Reactor Pressure Vessel with Eddy Current Methods. — RS 89
7. Development of Non-Destructive Testing Methods for In-Service Inspections of Reactor Pressure Vessels. — RS 27/2

7. Reference Documents

- Paper for IEA - SLT - ERD - Expert Group on Reactor Safety "Safety Investigations in the Decommissioned HDR-Plant." April 1975 (in English)
- Paper: HDR Safety Program - General Program - Status as of December 1975 Gesellschaft für Kernforschung (in English)
- Quarterly reports: in the IRS-Research Report series (in German) Reports IRS - F - 21, 23, 25 - 27
- Annual report: IRS - F - 24, 25, 26 (in English)
- Technical reports: PHDR 1 - 78
  Investigation of a Steam-Isolation-Valve following Rupture of the Primary-Cooling-Piping, Quicklook-Report DIV I
- Technical reports: PHDR 2 - 78
  Non-Destructive Testing and Materials' Investigations to Establish the Initial Condition of the HDR Pressure Vessel.
- Technical reports: PHDR 4 - 78
  Comparison of mesured and calculated results of earthquake investigations during the low level excitation phase - A overview

8. Degree of Availability

Unrestricted destribution, except for "technical reports" which have temporarily restricted distribution.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	20	RS 263/9

Vorhaben/Project Title	
Analytische Tätigkeiten der GRS im Rahmen des Reaktorsicherheitsforschungsprogramms des BMFT Störfallanalyse bei Schnellen Brutreaktoren Analytical Activities of the GRS in the Frame of the BMFT Research Program on Reactor Safety Accident Analysis of Fast Breeder Reactors	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Gesellschaft für Reaktorsicherheit (GRS) mbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.3.1977	31.12.1980	Dr. A. Scharfe
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	31.12.1978	

1. General Aim

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

2. Particular Objectives

The design of liquid metal fast breeder reactors is strongly influenced by the consequence of both, the hypothetical core disruptive accidents and the operating failures. To allow a deterministic calculation of such transients the computer codes SAS3D /1/, VENUS-II /2/ and SSC /3/ shall be procured and implemented.
SAS3D and VENUS-II compute the phases of the accident initiation and of the mechanical disassembly during hypothetical accidents respectively. The code SSC describes the plant. These codes shall be used to compute selected accidents of the SNR-300 and of the future commercial fast breeder reactor (about 1000 MW).

3. Research Program

3.1 Procurement of the codes and implemantation.

3.2 Analysis of the loss-of-flow and the transient overpower accident without scram for the SNR-300.

3.3 Analysis of the loss-of-flow and the transient overpower accident withour scram for the SNR-2.

3.4 Analysis of main circuit accidents.

3.5 Further development of computing methods.

4. Progress to Date

Ad 3.1 The SAS3D Release Version 1.0 has been implemented.
For the evaluation of SAS3D results an auxiliary code has been developed and implemented. For the graphical display of SAS3D calculated results some mudules of the SAS3D Plotting Package (PLO3D) have been implemented.
The computer code VENUS-II has been implemented.

Ad 3.2 For initial calculations with SAS3D a reference dataset for the loss-of-flow accident at the end of the SNR-300 was compiled. On the basis of the reference dataset different parameters have been varied to analyse the sensitivity of single assumptions during the initial phase of the accident.
For initial calculations with VENUS-II a dataset for a transient overpower accident was taken from a KFK-report /4/. The results have been compared with calculations of the code KADIS /4/ and show marked differences, probably caused by considering a fuel-coolant interaction in KADIS. The present VENUS-II version does not contain a fuel-coolant interaction model.

Ad 3.4 The computer code SSC (CDC-version) has been taken over from BNL. The conversion to the IBM-compatible computer AMDAHL 470 V/5 has been started but not concluded up to now.

6. Results

Ad 3.2 The first results of the parameter variation of the SAS3D reference dataset are as follows: The temperature criterion (fuel melting temperature instead of the failure melt fraction) for the fuel pin failure leads to a more mitigant excursion during the initiation phase.
By reducing the fuel particle heat transfer coefficient for the fuel coolant interaction in the order of some decades the maximum power is smaller by the factor 2.
Separate clad motion is only possible under consideration of axial expansion reactivity feedback. For an effective fraction of 0.5 for the feedback separate clad motion occurs in two channels. The reactivity feedback of clad motion is small compared with the feedback caused by fuel slumping.

During the evaluation of some other cases a code failure was detected. One special parameter for plotting purposes influences the instationary part of the code, too. The reason is not known up to now. Nevertheless it is possible to avoid this by not using this option.
Certain parameter variations influence the accident only in a later phase. Therefore the RESTART-option has been tested to save computer time and costs.
For the disassembly phase of a transient overpower accident with an initial ramp of 15 ¢/sec in the SNR 300 Mark 1 A core, VENUS-II shows considerably later neutronic shutdown, higher maximum power, higher energy release and higher final pressures and temperatures than the KADIS calculations. These differences are attributed to the fact that KADIS applies an option to describe the heat transfer from fuel to sodium, which is not available in VENUS-II. Due to this heat transfer the liquid sodium expands and produces, together with the hot fuel, high single phase pressure peaks. These pressure peaks cause an early fuel movement which leads to early neutronic shutdown. In VENUS-II, pressure is only produced by fuel expansion and, hence, high pressure and fuel movement occur later. Some other differences between the two codes, e.g. different mesh sizes, are believed to have only minor effects on the different results.

7. Next Steps

The analysis of the loss-of-flow accident and the implementation and testing of remaining plot modules will be continued according to the research program.

9. References

/1/ W.R. Bohl et al.: An Analysis of the Unprotected Loss-of-Flow Accident in the Clinch River Breeder Reactor with an End-of-Equlibrium-Cycle Core. ANL/RAS 77-15, 1977

/2/ J.F. Jackson, R.B. Nicholsen: VENUS-II - An LMFBR Disassembly Program; Argonne National Laboratory, ANL-7951, 1972

/3/ A.K. Agrawal: SSC - An Advanced Thermohydraulic Transient Code for LMFBRs, Brookhaven National Laboratory. BNL 50467, 1975

/4/ P. Schmuck et al.: KADIS - Ein Computerprogramm zur Analyse der Kernzerlegungsphase bei hypothetischen Störfällen in schnellen, natriumgekühlten Brutreaktoren. KFK 2497, November 1977

10. Degree of Availability of Reports

Reports are available through the specified companies.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	20	RS 139

Vorhaben/Project Title	
Programmentwicklung zur mehrdimensionalen Kontinuumsmechanik Code Development in Multidimensional Continuummechanics	Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor INTERATOM GmbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.9.74	31.12.79	H. Banasch
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	DM 2.251.877

1. General Aim

Developments of methods and computer codes for calculations in multi-dimensional hydrodynamics coupled with multi-dimensional structural dynamics.

2. Particular objektive

Development of a computer code treating "KOntinuumsmechanik mit Euler-Langrange-Koordinaten und SCHalentheorie (KOELSCH)". Validation of developed methods and codes via available experiments. Participation in the COde VAlidation-Programme at Euratom, Ispra (COVA).

3. Research Programme

3.1 ELK

Computational methods for multi-dimensional hydrodynamics with "Euler-Lagrange-Koordinaten" (for the three-dimensional case only Eulerian coordinates).

3.2 BLASE

Methods for the description of expansion of bubbles (vapour bubbles or bubbles of explosive charges) in two dimensions.

3.3 SCHAL/PLAT

Methods for the two-dimensional structural dynamics of shells and plates.

3.4 EOS (Equation of State)

Treatment of material properties.

3.5 KOELSCH

Main programme for coupling the various routines.

3.6 INKOEL/OUTKOEL

Routines for input, output and plotting.

3.7 Validation, COVA-Programme at Euratom/Ispra

Calculation of appropriate experiments and participation in the COVA-Programme at Euratom/Ispra.

4. Experimental Facilities, Computer Codes

In section 3 there are given the names of the various routines which will be developed and used.
For participation in the COVA-Programme at Euratom/Ispra the code ARES is used.

5. Progress to Date

ad 3.1

The coupling between hydrodynamics and structural dynamics has been completed so far as it is necessary to treat single Lagrangian lines. In order to dispose of sure debugging facilities the largely geometrical coupling work is performed in two different algorithms implemented as KOSH/G respectively KOSH/S.
Based on the uniqueness of the geometrical situation the two ways must lead to equivalent results or indicate mistakes, if they do not do so.
In scope of code validation (KOEVA) some analytical test examples were calculated in view of the following check points

- energy conservation
- mass conservation
- influence of numerical damping on the deviation from analytical solution

ad 3.3

The shell theory was extended to any boundary conditions (free ends, fixed ends, any branching points). An analytical example tested the programmed theory.

ad 3.5

The coupling of the various routines was optimized by a modular setup.

ad 3.6

A standardization of data input was developed and realized in the modul INKOEL.

ad 3.7

A conception for validation of the KOELSCH code (KOEVA) was under discussion.
Reports of the Von Karman Institute/Brussels dealing with hydrodynamic flow through perforated structures have been analysed during the participation in the COVA-Programme at Ispra. The results of the Von Karman Institute have been evaluated with respect to the problem of treatment of perforated structures in containment codes.

ad 3.8

The second part of Zwischenbericht des Forschungs- und Entwicklungsauftrages RS 139 was edited in May.

6. Results

ad 3.1

An analytical test example for coupling between hydrodynamics and structural dynamics demonstrated the coupling technique.
A comparison of the numerical and analytical results showed excellent agreement.
With regard to code validation some first steps were made relative to demonstration of energy and mass conservation.

ad 3.3

The extended shell theory was validated by agreement within 5 % with an analytical solution.

ad 3.5

The modular setup showed its practicability and effectivity as well in some representative test examples as in all other following applications.

ad 3.7

With respect to the work for perforated structures in the frame of COVA/Ispra a report has been completed and is available.

ad 3.8

The second part of Zwischenbericht des Forschungs- und Entwicklungsauftrags RS 139 has been completed and is available since May. The documentation of representative applications of the KOELSCH-Code is envisaged for the third part of this report, which is being prepared at present.

The progresses and results obtained in the evaluation of modul ELK are documented in an INTERATOM-note.

7. Next Steps

ad 3.1

The coupling modules KOSH/G and KOSH/S are to be generalized to treat more than one Lagrangian line and to connect various lines.

The observations in energy and mass conservation will be continued.

ad 3.3

Initial work for the evaluation of the plate theory will be taken up.

ad 3.6

The use and availability of plotter software is to be discussed to get a scheme for development of the planned modul PLOKOEL.

ad 3.7

The KOEVA-conception will be elaborated.

8. Relation with other projects

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9. References

L. Lange et al.
Zwischenbericht des Forschungs- und Entwicklungsauftrages RS 139 - Programmentwicklung zur mehrdimensionalen Kontinuumsmechanik. KOELSCH (KOntinuumsmechanik in Euler-Lagrange-Darstellung und SCHalentheorie)
Teil 2: Strukturmechanik und ihre Koppelung mit der Fluiddynamik.

B. Persson, G. Liebecq
Appraisal of the possibilities to include the treatment of instationary fluid flow through perforated plates in containment codes.

K. Gerling
KOELSCH-Validation, Teil 1
INTERATOM-Notiz Nr. 7o.758.9

10. Degree of Availability of the Reports

On request by INTERATOM GmbH, 5o6o Bergisch Gladbach 1

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	20	RS 272
Vorhaben/Project Title APRICOT, Phase B APRICOT-program, phase B		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor INTERATOM GmbH
Arbeitsbeginn/Initiated 1.7.77	Arbeitsende/Completed 31.12.78	Leiter des Vorhabens/Project Leader H. Banasch
Stand der Arbeiten/Status Completed	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds 116.983.--

1. General Aim

Within the APRICOT-Program (Analysis of PRImary Containment Transients) a series of benchmark problems are to be calculated under international participation with continuum mechanics codes designed for the description of the dynamic behaviour of the primary containment of sodium cooled fast reactors. Their results will be compared with each other and with experiments specially chosen for this purpose.
The knowledge gained here can be used to solve engineering problems more economically and to increase confidence in usage of internationally validated codes for safety cases.

The APRICOT-Program was initiated and organized by USERDA.

2. Particular Objectives

The participation in the APRICOT-Program takes place by treating the proposed benchmark problems with the INTERATOM code ARES. Results and comparisons with experiment will be evaluated by a group of independent experts from which an increased confidence in the codes will result.
Moreover the participation in the APRICOT-Program will give access to foreign experimental results and will permit to become acquainted with the capabilities of the other codes.

3. Research Program

3.1 Ten organisations from five countries participate in the APRICOT-Program with eleven computer programs. For comparison with experiments results from explosion tests of the Code-Validation-Program in England and Ispra (EURATOM) and from explosion tests of the Stanford Research Institute/USA (SRI) are available.

3.2 The APRICOT-Program consists of two phases. Phase A contained three tasks.This phase, where INTERATOM participated in cooperation with GRS (Köln) is terminated. For phase B (considered here) tasks four through seven were defined. With ARES the tasks six and seven will be treated and the results will be compared with the corresponding SRI-experiments.

Task six consists in the calculation of an explosion test in a water filled thin walled cylinder. It contained a thick walled inner tank made of steel. Pressure and strain gauges which were fixed to the cylinder walls produced experimental curves which can be compared to the calculations.

Task seven treats a similar SRI-experiment which differs only in the inner container, which consists of lead with a thin surrounding shell of aluminium.

3.3 The geometric layout of these experiments is modelled with ARES which calculates the loading and deformation history of the tanks.

3.4 The calculated results are compared with the experimental ones and with those of the other codes.

4. Experimental Facilities, Computer Codes

The INTERATOM-code ARES is a two-dimensional (cylinder symmetrical) compressible Lagrange program which works by the finite difference methode. If pertinent initial and boundary conditions are given, the code calculates pressure wave propagation, hydrodynamic flow and the structural deformation by means of the programmed differential equations, equations-of-state and stress-strain behaviour of the materials. As results coordinates, velocities, pressures, stresses, strains, energies and other variables can be plotted as functions of space and time.

5. Progress to Date

The APRICOT-Problems No. 5,6 and 7 were modelled and calculated with ARES. The computed pressure, impulse and strain curves were compared with the corresponding experimental ones and were documented in /‾9 d_7. This documentation was handed over to the coordinators of the APRICOT-Program at the Brussel's Meeting in October 1978.

6. Results

For the APRICOT-Problems 5 - 7 the desired local pressure, impulse and strain curves were computed by the ARES-Code. These calculational results compared reasonably well with the corresponding experimental curves produced by the SRI-experiments.

7. Next Steps

None

(Note: The comparison of ARES-results with results of other codes will be done by the reviewers of the APRICOT-Programm.)

8. Relation with other Projects

The work described here is considered as an important support and extension of the international code validation work, where INTERATOM participates in the code validation program in England (RS 272) and in Euratom/Ispra (RS 139, RS 162).

9. References

a) W. Kellner
APRICOT-Problems 1 and 2
INTERATOM-Notiz 7o.158.5 (1976)

b) Meier (GRS/Köln), Kellner, Doerbecker (IA)
ARES-computations for APRICOT 3
INTERATOM-Notiz 7o.245.3 (1976)

c) The APRICOT-Program: Comparison and bench marking of computational methods for analysis of LMFBR structural response to HCDA pressure loads. Phase 1 report. SAN-1112-1, Oct. 77

d) K. Doerbecker
ARES-Calculations on APRICOT-Problems 5 ÷ 7
INTERATOM-Notiz 7o.877.1 (1978)

1o. Degree of Availability of the Reports

On request

Berichtszeitraum/Period 1.1.78 - 31.12.78	Klassifikation/Classification 20	Kennzeichen/Project Number RS 235
Vorhaben/Project Title Theoretical and experimental investigations on transient velocity- and temperature fields within the reactor outlet plenum Theoretische und experimentelle Untersuchungen zur Bestimmung der Strömungs- und Temperaturfelder im oberen Reaktorplenum bei transienten Vorgängen		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor INTERATOM GmbH
Arbeitsbeginn/Initiated 1.1.77	Arbeitsende/Completed 28.2.79	Leiter des Vorhabens/Project Leader Banasch (Koord.)
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. **General Aim**

   Safety Design of Sodium Cooled Breeder Reactors against Transient Thermal Stresses.

2. **Particular Objectives**

   Development of an Experimentally Proved Computer Program Concerning Time Dependent Temperature Fields within a Reactor Outlet Plenum.

3. **Research Program**

3.1 Theoretical Investigations

3.1.1 Computation of a Representative Transient Behavior of Fast Breeder Reactors by the Use already Existing Computer Codes

3.1.2 Discussion of Fundamental Flow Model Desgin by the Use of an Elementary Code

3.1.3 Literature

3.1.4 Development of a New Computer Program

3.1.5 Coordination of Model Parameters and Experimental Results

3.2 Experimental Investigations

3.2.1 Discussion of the Experimental Program and the Specified Parameters

3.2.2 Reconstruction of the Existing Outlet Region Feature Model

3.2.3 Determination and Providing Measuring and Control Technique

3.2.4 Instrumentation

3.2.5 Time of Beginning of Operation

-2-

3.2.6 Test Procedure

3.2.7 Data Processing

4. Experimental Facilities, Computer Codes

Tests on transient behavior of the upper plenum of sodium cooled fast reactors will be run using a 1:4 scaled water model of SNR 300 reactor upper plenum. The velocity profil of the entering sodium as well as the geometric proportions of this reactor are simulated. With this model and the constructed special loop system experimental investigations can be run changing a mean Arichimedes number by ± 80%.

To make the program TIRE compatible with the geometry of the special test facility some parts of the TIRE Code were changed. Prilimary TIRE calculations could ev aluate informations about the characteristic transient SNR 2 outlet plenum behavior.

A new computer code INKØ2T concerning the two dimensional streaming behavior of an incompressible fluid is going to be developed.

5. Progress to Date

ref.3.1.4 The theory of the new computer code INKØ2T has been developed. This code respects the inner heat conduction and the buoyancy forces of a two-dimensional incompressible fluid. In the case of a given entering volocity profil this digital program needs only a short execution time. Cosiderations concerning the improvement of convergence are carried out.

ref.3.2.5 The time of beginning of operation showed the specific behavior of the loop system. To guarantee reproducible test results parts of the loop system had to be changed into automatically controled systems. The coupling of the 60 thermoelements to the PCM-recording station has been tested.

In the following, stationary thermohydraulic mixing behavior has been tested for a temperature range of $30^{o}C$ and a range of mean velocity of $0{,}003 < V$ [m/sec] $< 0{,}08$

ref.3.2.6 First experiments relating to the transient mixing behavior of the upper plenum of sodium cooled reactors are done. Some of the very first experiments showed the damping influence of the model inlet volume. This damping characteristic could be figured out by additional tests. By this, now it is possible to get the exact specific temperature gradients which are necessary to analyse different reactor transient conditions within the experimental program.

6. Results

ref.3.2.4 The convergence of the matrix of iteration has been improved considerably.

ref.3.2.5 All tests of the time of beginning of operation led to best results so that we could start with the special test procedure.

ref.3.2.6 First results concerning the stratification problems within the large upper plenum volume have been found out by a very qualitative discussion of characteristic temperature-time histories. Within the range of a normal reactor scram condition there is a definite formation of stratification in the lower part of the outlet reactor plenum. This stratification diminishes the mixing behavior of the fluid volume so that even some hundred seconds after inition of shut down large temperature gradients exist especially in the near of the outlet pipes. In some cases even 800 seconds after the be ginning of scram conditions there are still relativ large temperature differences to be found at some measuring points in the upper plenum.

7. Next Steps

ef.3.1.4 The computer code INKØ2T will be made available and tested.

ef.3.2.6 Going on with the experimental procedure

Relations with Other Projects

Till this time there are no relations with other projects.

9. References

/1/ Vossebrecker, Reiff
Beschreibung des Rechenprogrammes TIRE und Dokumentation der für den Reaktortank durchgeführten Analysen.
IA-Notiz 69.2.6

/2/ Vossebrecker
Untersuchungen zur Verifizierung des Rechenprogrammes TIRE
IA-Notiz 69.21.1

/3/ Lieberoth, Dr.
Entwurf eines Rechenprogrammes (INKØ2T) zur Berechnung einer instationären, inkompressiblen 2d-Strömung mit innerer Wärmeleitung
IA-Notiz 33.1662.8

/4/ Grönefeld, G.
Ermittlung repräsentativer transienter Abläufe für Großbrüter mit TIRE.
IA-Notiz 69.70.8

/5/ Benemann, Penny
Versuchsprogramm zur Bestimmung der Strömungs- und Temperaturfeldes im oberen Reaktorplenum bei transienten Vorgängen
IA-Notiz 53.2865.5

/6/ Benemann, A.
Transiente Strömungs- und Temperaturverhältnisse im SNR-Oberplenum sowie in simulierten Modellströmungen (Analogiebetrachtungen)
IA-Notiz 53.2971.9

10. Degree of Availability of the Reports

-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31 12.1978	2o	RS 269
Vorhaben/Project Title		Land/Country
Untersuchung zur tankinternen Notkühlung unter Berücksichtigung von Sieden		FRG
		Fördernde Institution/Sponsor
Examination of the in-tank emergency cooling with regard to boiling		BMFT
		Auftragnehmer/Contractor
		INTERATOM GmbH
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.6.77	3o.4.79	Banasch (Koord.)
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	December 1978	

1. <u>General Aim</u>

Higher power densities are aimed at in larger breeder plants. This can lead to sodium boiling in the case of emergency cooling. Short-term boiling may be permitted in the fuel elements, if it can be proved that the capability of the boiling elements is guaranteed for the boundary conditions of the in-tank natural circulation.

The operational capability of the emergency cooling has till now been proved by the INTERATOM-computer code NOTUNG / 1 /. As this program is restricted to the simulation of single phase flow processes an extension of the code is necessary to take account of boiling phenomena in single fuel elements under natural circulation conditions.

The aim of this project is to develop a simplified boiling module which together with NOTUNG will enable predictions regarding flow and temperature fields that may arise in emergency cooling phases if the local boiling temperature of the coolant is exceeded in the bundle area.

A feasibility study has to be carried out in parallel with the above to produce an experimental model which can be used in boiling experiments at a later date.

2. <u>Particular Objectives</u>

- Evaluating of literature concerning previous boiling experiments
- Development of a theoretical model to calculate natural circulation boiling phenomena
- Verifying of the theoretical model by means of present boi-

ling experiments
- Insertion of the boiling model into the code NOTUNG
- Production of a feasibility study for boiling experiments under natural circulation conditions

3. Research Program

   3.6 Program development

   Production of the computer code

   3.7 Calculations

   Checking of existing boiling experiments

4. Experimental Facilities,Computer Codes

5. Progress to Date

to 3.6

A computer Code has been developed, which enables the user to calculate flow and temperature fields of single and two-phase flows.

to 3.7

The input datas of the boiling experiments / 2 / were gathered. Thereafter the test results have been checked by means of the computer code.

6. Results

to 3.6

A computer program has been developed and been checked successfully with regard to its numerical functioning. It is suitable for an arbitrary number of parallel channels. At both ends they are connected by a collective point. At each mesh point one can vary for example: the heat source, heat capacity, coefficient of heat transmission of the channel walls, coefficient of pressure loss. The program caculates the temperature, pressure and mass flow rate distribution of single and two-phase flows.

to 3.7

Using the known input datas calculations have been carried out.

They represent experiment no. 37-2o/53 in / ? / over the whole duration of the test of about 1o min. Thereby, no satisfying agreement was achieved with the test results.
Therefore, the input datas of the experiment, used for the calculations, as well as the interpretation of the experimental runs were discussed in detail by KfK and IA.

Especially the pressure losses in the cold part beneath the real test device are critical. They are mainly noticable at the sieve in the inlet of the test element and at the throttle valve, which is used to reduce the mass flow rate stepwise during the run of the experiment.
Concerning the friction losses up to now only inaccurate values have been available for the calculations.
KfK will try to solve the problem by measurements of the pressure drop which were carried out previously in the specific part of the loop. In addition to experiment no. 37-2o/53 it is being considered to use other KfK experiments with natural convection.

With the aid of the above mentioned tests for example the influence of the bypass, which is parallel to the test bundle, will be determined.

7. Next Steps

to 3.5
Preparation of the constructional and measuring draft

to 3.8
Re-working of the program
Program modification resulting from the information gained from the checking

to 3.9
Module insertion
Insertion of the boiling module into NOTUNG

to 3.1o

Final report

Description of the program. Evaluation of the results obtained and analysis to see if more tests are necessary with the aim of further modifying the program.

8. Relation with other Projects

-

9. References

/ 1 / F. Rösgen, F. Timmermann, H. Vossebrecker
Wärmetechnische Berechnungen zur Notkühlung des SNR-3oo
KTG-Fachtagung der Fachgruppen Reaktorsicherheit und Thermo-Fluiddynamik in Stuttgart, Januar 1975

/ 2 / A. Kaiser, W. Peppler, M. Straka
Decay Heat Removal from a Pin Bundle
International Meeting on Fast Reactor Safety and Related Physics
October 6 - 8, 1976, Chicago

1o. Degree of Availability of the Reports

/ 1 / Fa. Interatom
5o6o Bergisch Gladbach 1
Friedrich-Ebert-Str.

/ 2 / Institut für Reaktorentwicklung
Kernforschungszentrum Karlsruhe
75oo Karlsruhe
Postfach 364o

Berichtszeitraum/Period 1.1.78-31.12.78	Klassifikation/Classification 2o	Kennzeichen/Project Number RS 242
Vorhaben/Project Title Verifizierung von ..chenprogrammen für Tank und Tankeinbauten in Verbindung mit dem englischen COVA-Programm (explosion containment code validation tests) Verification on Computer Codes for Tank and Tankcomponents with the UK Containment Code Validation Tests		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Karlsruhe (KFK)
Arbeitsbeginn/Initiated 1.12.1976	Arbeitsende/Completed 31.12.79	Leiter des Vorhabens/Project Leader H. Knuth
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 31.12.78	Bewilligte Mittel/Funds

1. General Aim

Within the frame of safety considerations for sodium cooled fast reactors hypothetical power excursions have to be treated. To study the mechanical impacts of these excursions on the primary containment computer codes are used. A verification of the computer codes used is necessary for the safety evaluation of the reactor vessel. There has not been a fully validated computer code until now in the Federal Republic of Germany. An agreement has been concluded between KFK in Karlsruhe and the United Kingdom Atomic Energy Authority (UKAEA) providing for the exchange of R+D-results. The agreement stipulates that all the experimental results from the English code validation program are transferred to Karlsruhe. The experiments are performed with water filled vessels with and without internals. The power excursion is simulated by an explosive charge. A selection is made among the roughly 4o main experiments. A comparison is performed between the experimental and calculated results. So statements can be made on the reliability and accuracy of the theoretical computer models used. Necessary modifications have to be indicated. Results of these activities are also integrated in the licensing procedure for the SNR-3oo fast breeder reactor.

2. Particular objectives

The British COVA program includes experiments with short tanks (pool type design) and with long tanks (loop type design). Since the SNR line is based on the loop type, the advance calculations and recalculations will concentrate on experiments with long tanks.

3. Research Program

In 1977 and 1978 computations and evaluations were made for the following experiments from test programs under way in Great Britain:

WT 5: long rigid tank
WT 7: long thin tank
WT 8: long thin tank with different charge weight
WT 9: short rigid tank, rigid inner
WT 12: short rigid tank, tank inner, rigid diagrid
WT 15: short thin tank, thin inner, perforated diagrid
FT 4: short rigid tank
FT 6: short thin tank
FT 7: short thin hemisperical bottomed tank
FT 1o: short rigid tank, rigid inner, perforated diagrid.

4. Test Facilities, Computer Codes

A german working group was established which is made up of representatives of KFK, GRS and Interatom. Advance calculations and recalculations were performed on selected experiments on a work sharing basis, using the ARES code developed by Interatom (2-dim. Lagrange-code).

Work performed, Results Obtained, Nuclear Research Center Karlsruhe (KFK)

5. Work performed

Eight parameter calculations for COVA-experiment FT 4 (short rigid tank without internal structures) were made and documented in three calculation reports.
The following parameters were varied:
- smallest allowable pressure (respectively highest possible hydrostatic tension)
- mesh size of rectangular finite difference net
- mesh size of deformed finite difference net around charge

The velocity of sound, the travel times of characteristic waves and the time of water impact against the roof plug were determined for interpretation of measured pressure signals at the bottom and at the wall of the tank. These evaluations were done by using the theory of linear wave propagation considering the qualities of d'Alembert's solution for the wave equation.
Comparison were done between calculated results obtained by the computer code ARES and experimental results. Documentation of this work is underway. Development of a computer program was completed for interpretation of dynamic tensile tests. The method of correcting strain measurements obtained in dynamic tensile test at Ispra

was checked. Material data test program for thin tank proposed by UKAEA was examined in cooperation, with GRS, IA and GFS-Ispra. Activities were initiated to fabricate 4o specimens for dynamic tensile tests at Ispra.

6. Results obtained

It was possible to interprete some characteristic pressure signals at the bottom and at the wall of the tank by application of linear wave propagation theory. This allowed a correlation between measured signals and physical events. Comparison of experimental and ARES results showed: the impulse of direct pressure wave up to the time of arrival of the wave reflected from the tank wall is underestimated by calculation (fig. 1). The impulse-time curve for the plug after impact is overestimated. Whereas measured impulse data represent a plateau calculated results are growing discontinuously. Comparison of impulses during intermediate times show discrepancies. One explanation is that the tank wall is assumed rigid for calculation. A larger wall thickness of the tank would have been more suitable for assessment of hydrodynamic aspects in ARES. Comparison of parameter calculations with experimental results shows that smaller mesh size gives smaller differences. Regarding time-averaged pressures it can be shown that calculated results are sensible against choice of mesh size. Interpolation of dynamic tensile test results of Ispra proved that a visco plastic material law with hyperbolic sinus dependency for strain rate in the range of 3.7 . $1o^{-3}sec^{-1}$ to 19o sec^{-1} is suitable if a strain rate dependency for strain hardening is assumed. Evaluation and interpretation of test results gained by dynamic tensile tests showed the need of different hypothesis for evaluation which must be checked independently of test. It should be stressed that measured strains need better corrections for accounting effects not yet considered.

Work Performed, Results Obtained, Company for Reactor Safety, Cologne (GRS)

5. Work Performed

Following calculations with the computer code ARES were performed:

- Ft 1o short rigid tank, rigid inner, perforated diagrid
- WT 12 short rigid tank, thin inner, rigid diagrid
- WT 15 short thin tank, thin inner, perforated diagrid

These experiments were choseh from english COVA program by setting priorities. The input parameters for these calculations are based on results of former parameter calculations with the code ARES. Calculations of experiments with internals ended up to about 2.5 msec because the expanding bubble touched the diagrid. Consequence of this fact is strong mesh compression of the diagrid causing a rapidly decreasing time step.

Further consequence of perforated diagrids is that passing of the bubble through the diagrid into the downward area of the vessel cannot be simulated and this yields a limitation by phsical arguments.

Experiment FT 1o was calculated up to a problem time of 2.1 msec while the calculations of WT 12 and WT 15 reached times of 2.2 msec and 2.7 msec.

Parameter calculations were made for COVA-experiment WT 15 to model the perforated diagrid. The influence of resistance coefficient and of material data ($\sigma - \epsilon$ curve) on results were studied. For interpretation of calculations and comparisons with experimental results a statistical method was established which allows a statement about the quality of calculation.

6. Results Obtained

At problem time of 2.2 msec in experiment WT 12 the strain process is finished and the impulse valves have reached a stationary value. Therefore at this point of time a comparison between experiment and calculation is possible. Calculated impulses are deviating from experiment within 1o% to 2o%. Differences at the bottom, at diagrid and at inner shield tank are smaller than at tank wall and roof plug. Calculated impulses for experiment FT 1o reach at 2.1 msec a stationry phase with exception of outward plug area. Comparing calculated with the test results differences of the same magnitude occur as in experiment WT 12. Calculations for experiment WT 15 were done up to 2.7 msec. It can be shown that after about 1.5 msec strain history of inner tank was finished whereas final strains of outer tank were reached at times 3 msec. For parameter calculations following input were varied:

- resistance coefficient of diagrid ξ = 5o; 177; 35o; 86.25
- perforation range of diagrid, first a perforation was assumed for the complete area and afterwards for the real hole area
- boundary condition of the diagrid, clamped fast at first assumptions, pinpointed as second assumption,
- assumption of material data with 15% higher values as in documented reports.

Parameter calculations were done up to a problem time of 1.5 msec. Two calculations which can be accepted as best simulation of experimental results reached problem times up to 2.7 msec. Following statements can be made by parameter calculations:

- changing perforation range from the real hole area to the complete plate area does not influence the results significantly . Hoop strains of the inner tank as well as impulse curves at selected positions are remaining nearly the same,
- changing resistance coefficients ξ has the predicted influence. Increasing resistance produce an increasing hoop strain between charge and diagrid. Impuls curves show no significant increase,
- simulation of boundary conditions of the diagrid is of great influence. In the case of a clamped plate beneath the charge there are higher hoop strains at the shield tank and higher impulses as in the case of a pinpointed diagrid.

Work Performed, Results Obtained, INTERATOM GmbH, Bensberg

During the calculations made by the German COVA working group there were doubts if the material data by Albertini are relevant for used tank material. It was tried to estimate the material data of the cylindrical part of IT 9 tank with measured impulse data. Additionally to the tank experiments there were calculated with ARES two simple problems coupling fluid with structure. These two cases were calculated too with computer codes ASTARTE and SEURBNUK developed by UKAEA. Calculations of experiment WT 9 (short rigid tank, rigid inner) were started.

6. Results Obtained

Without considering the multi axial behaviour σ (ε)-values for experiment IT 9 are consistent with Albertini's. Corrections for multi axial behaviour move the curves in such a way that they are between the curve of Albertini and the curve gained from experiment IT 7. The inaccuracy of extraction method results in no final statement for material data.

Comparison of calculated results by computer codes ARES, ASTARTE and SEURBNUK shows good agreement of time dependent curves for impulse, velocity and strain. Whereas pressure curves got by Lagrangian codes (ARES, ASTARTE) are spiky the Eulerian code SEURBNUK presents a smooth curve. The difference can be explained by using different solution methods within the codes (explicit and implicit integration of time).

7. Planned Work

Calculations were made with computer code ARES for experiments WT 5, WT 7, WT 8, WT 9, WT 12, WT 15, FT 4, FT 7 ($\hat{=}$ IT 8) and FT 1o of the English COVA-program.

Delays for working procedure came from insufficient material data up till now. Therefore organisations involved in the COVA-program at Ispra and Great Britain started special activities to close this gap. January 1979 a first set of material data will exist.

There exist calculation reports of mentioned experiments but nof final report comparing calculated and experimental results. One of the reasons is the material problem. To change this situation the German COVA working group (cooperation between KFK, GRS and Interatom) decided to write final reports for experiments where calculations exist and where knowledge of material data is not important. This is the case for experiments WT 5, WT 9, FT 4, FT 1o with short rigid tanks (thick wall). Within the next 4 months the group will concentrate to these activities. Because of licensing work for SNR-3oo which has top priority Interatom was not able to work with planned manpower for 1978.

Activities for assessment of ARES code will be finished on 31.12.79. Up to this date COVA working group has to present final reports too for experiments for which material data are needed and not yet available now and has to make statements on the assessment with respect to the computer code ARES and for the analysis of the mechanical consequences of hypothetical accidents in LMFBR's.
Following activities are planned for first half year 1979:
Repeated calculation for experiment FT 4 with assumption of deformable tank walls. Final report for experiments FT 4 and WT 5, FT 1o.
End of calculations with preliminary material data and documentation of calculated results for FT 6. Application of circular theoretical model for parameter calculations with experiment FT 6. End of calculations and final report for experiment WT 9. Parameter calculations and documentation for experiment WT 15.

9. Literature

COVA-Report No. 22/77
Estimate of stress-strain values for the IT 9 vessel
Schäfer, March 1978

COVA-Report No. 3/78
ARES-calculation No. 1 - No. 4 for COVA-experiment FT 4
Y.S. Hoang, March 1978

COVA-Report No. 4/78
ARES-calculation No. 1, No. 5, No. 6 for COVA-experiment FT 4
Y.S. Hoang, June 1978

COVA-Report No. 5/78
ARES-calculation No. 1, No. 7 and No. 8 for COVA-experiment FT 4
Y.S. Hoang, June 1978

COVA-Report No. 7/78
Calculation of COVA-experiment WT 12 with ARES
W. Salz, April 1978

COVA-Report No. 8/78
On the strain corrections for dynamic tension tests
performed with the hydro-pneumatic tension machine
T. Malmberg, May 1978

COVA-Report No. 11/78
Comparison of ASTARTE, SEURBNUK and ARES
Calculations for two coupled fluid-shell problems
K. Doerbecker, May 1978

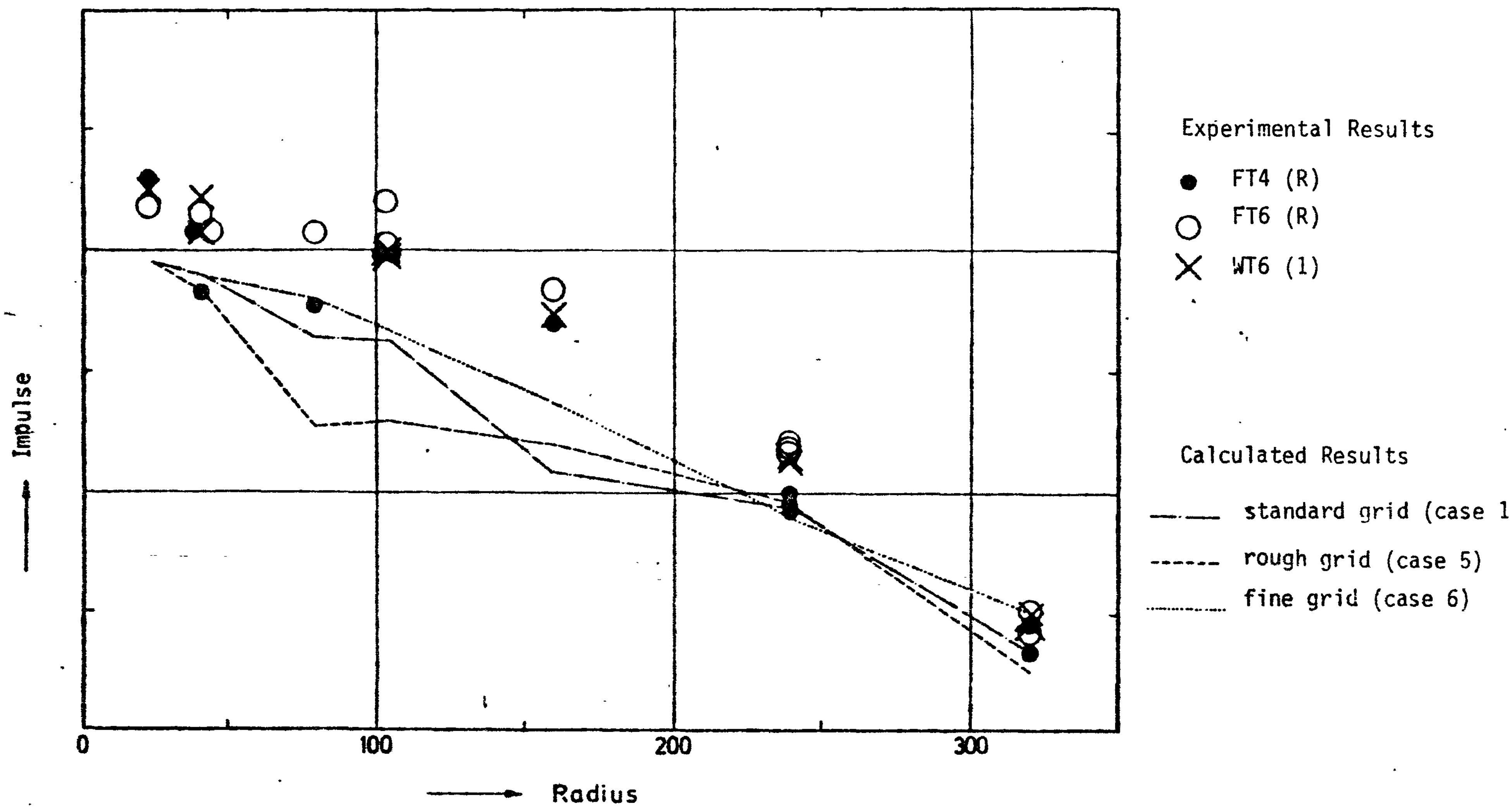
COVA-Report No. 11/78
Calculation of COVA-experiment FT 1o with ARES
W. Salz, April 1978

COVA-Report No. 13/78
Parameter variation for COVA-experiment WT 15
B. Baltes, October 1978

"The validation of the explosion containment code ARES within the European COVA-program", Intern. Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, Oct. 16-19, 1978.

COVA-Report No. 14/78
Calculation of COVA-experiment WT 15, case 1
B. Baltes, December 1978

COVA-Report No. 15/78
Calculation of COVA-experiment WT 15, case 2
B. Baltes, December 1978



Impulse of direct wave at tank bottom before entrance of first reflected wave coming from tank wall

Fig. 1

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	20	RS 276

Vorhaben/Project Title	
Untersuchungen reaktivitätswirksamer axialer Brennstoffverlagerung in der flüssigen und festen Phase Investigations on slumping of the fuel in the liquid and solid phase	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: INTERATOM GmbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.8.77	31.3.79	Banasch (Koord.)
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	225.545,-- DM

1. General Aim

The radial and local fuel relocation within a fuel pin is well understood and described by continuously improved models for the conditions under normal reactor operating conditions. However, the basic understanding of the axial forces and possible relocation mechanisms is still on a rather low level, although it has a non-negligible impact on the reactor safety features if the fuel relocation is significant for reactivity changes. It is the goal of this task to help to understand the mechanisms of axial fuel relocation.

2. Particular Objectives

- preparatory work for the treatment of axial relocation with the fuel pin modelling code IAMBUS;
- evaluation of irradiation experiments and collection of well defined data for calculations with IAMBUS;
- additions to and refinements of IAMBUS models with respect to axial fuel relocation (movement) effects;
- final evaluation and documentation;
- recommendation of an experimental test program, if necessary.

3. Research Program

3.1 Specific analysis of SNR fuel irradiations

3.2 Specific analysis of foreign fuel irradiations

3.3 Evaluation of available published information

3.4 Preparation and collection of data for IAMBUS

3.5 Analysis of experimental findings with IAMBUS, preferably with an improved version

3.6 Final Report

4. Experimental Facilities, Computer Codes

This task is based on the evaluation of experimental findings by theoretical means. The key tool is the code IAMBUS (INTERATOM Model for Burn-up Studies on fuel rods) which describes the behavior of cylindrical fuel rods for power reactors. The target of this code is to interprete fuel pin experimental irradiations and their application to the engineering design of fuel pins of LMFBR's as well as the prediction of the operational behavior of those pins.

5. Progress to Date

The activities concentrated on the investigations of possible fuel relocation in the solid phase by applying and where necessary extension of the models in IAMBUS, and by evaluating the measurements available from the irradiation of two bundles with 37 pins each, irradiated in Rapsodie Fortissimo.

A qualitative comparison was possible by using data from LWR pins. Furtheron, the kind of changes in IAMBUS had to be defined which will be necessary to describe the fuel behavior under fast operational power transients and the expected effects at the transition from the solid to the liquid state.

A certain literature survey had been started, which unfortunately suffered from the huge amount of detailed reports especially in the field of the TREAT experiments.

Higher priority than to the literature was given to the follow-up of experimental irradiations in the SILOE reactor, which cover some side areas of this task.

6. Results

Slow fuel densifications originate via shrinkage and post-sintering of the fuel especially during early burn-up periods.

A reactivity increase can be easily compensated by adequate changes of control rod positions during normal operation.

A spontaneous densification in the solid state will have a positive effect on reactivity, and may be a consequence of an instationary operational step leading to a differential elongation between clad and fuel column. Consequently, a spontaneous contraction of the fuel column to its axially most dense state is possible. This effect becomes relevant for power changes to lower levels and depends on the power level differences. It reaches its maximum value for the power changes from full power to levels between zero to 30 % power. Necessary condition for the maximum contraction is the unfavorable assumption that the fuel was originally sticking to the clad over the total length of the pin and that the contraction occurs spontaneously after reaching the respective power level.

Although these results are promising and already applicable for certain project cases, principal changes in some IAMBUS subroutines have to be made in order to treat correctly the fast (transient) operational power changes, as the method of using a sequence of very short time steps is not applicable anymore. Consequently a redistribution of technical subtasks will be necessary within the working group.

The pin experiments performed in SILOE lead to rather interesting results in a connected area: the penetration of sodium in purposely defected fuel pins leads only to chemical reactions with fuel, which do not have fuel relocations as a consequence although the remarkable but moderate (with respect to reaction speed) chemical reaction leads to the development of further breaks of the clad.

7. Next Steps

Higher priority has to be given to running experiments although they are only losely connected with this task. Consequently, the work for this task will be interrupted until March 31, 1979. Starting in the second quarter of 1979, the subtasks to be performed will be:

- changes in IAMBUS to be able to treat transient power changes;
- development of a simple model for the behaviour of a molten central fuel zone;
- further literature evaluation.

8. Relation with Other Projects

not applicable

9. References

not applicable

10. Degree of Availability of the Reports

-

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1978 - 31.12.1978	20	RS 254

Vorhaben/Project Title

Entwicklung fernbedienter Ultraschall-Prüftechnik für Schnellbrutreaktoren

Development of Remote-Controlled Ultrasonic Inspection Technology for Fast-Breeder Reactors

Land/Country: BRD FRG

Fördernde Institution/Sponsor: BMFT

Auftragnehmer/Contractor: M.A.N. Nürnberg

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.1977	31.12.1979	Otte
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	29.12.1978	

1. General Aim

The objective of the work carried out under this project is to further develop equipment for remote-controlled ultrasonic inspection of the reactor tank of liquid-metal-cooled fast breeder reactors so as to permit remote-controlled volumetric inspection of the austenitic reactor inner tank according to the present status of the art on completion of this work.

2. Particular Objectives

- Development and preparation of complete engineering for inspection equipment with the associated probe module carriers for the reactor inner tank

- Development and preparation of complete engineering for a linked truck train with drive sprocket and chain storage

- Development and preparation of complete engineering for the cooling system of a TV-camera

- Development of a control system for adapting a universal control system to the specific requirements of the inspection facility

- Development and construction of a test rig for testing high-temperature-resistant electric motors under conditions similar to those in a reactor and conductance of the tests

3. Research Program

3.1 Investigations to determine the necessary clearances in negotiating the S-shaped bend of the guide tracks.

3.2 Development of inspection equipment for the area close to the flange of the reactor inner tank, the tank bottom and the nozzle-to-pipe transition welds.

3.3 Development and construction of a motor test bench for testing high-temperature motors under conditions similar to those in a reactor.

3.4 Development of a linked truck train with drive sprocket and chain storage.

4. Experimental Facilities, Computer Codes

Not applicable

5. Progress to Date

Ad 3.1 The investigations aimed at improving clearances in negotiating the S-shaped bend of the guide tracks showed that a reduction of curvature of the S-shaped bend and stiffening of the track system by additional sleepers are inevitable. INTERATOM will make allowance for this requirement in designing the track system.

Ad 3.2 Priority was given to the development of three manipulator systems for the tank area close to the flange, tank bottom and transition welds over the other manipulator systems. Some revised designs are already available.

Ad 3.3 In view of the uncertainties existing regarding the suitability of the proposed drive concept, the work program was modified in that other points on the program were deleted and priority given to the construction of a motor test bench permitting testing of the drive under realistic conditions. The development of the test bench has been substantially completed and fabrication has been started.

Ad 3.4 The development of the linked truck train with drive sprocket and chain storage has also been included as a new item in the work program. A linked truck train has been designed which is separable and substantially free from thermal expansion. Work is in hand on the drive sprocket and drive unit.

6. Results

It has not yet been possible to complete the work in hand. Part results are described under Item 5.

7. Next Steps

The work under 3.1 to 3.4 will be continued.

8. Relation with other Projects

The problem definition has resulted in interfaces with the work done by INTERATOM, Bergisch-Gladbach, - RS 250 and the Bundesanstalt für Materialprüfung (BAM) - RS 244.

9. References

No technical reports have been completed during the period under review.

10. Degree of Availability of the Reports

Not applicable.

20. OTHER TOPICS

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.1978 - 31.12.1978	20	RS 244

Vorhaben/Project Title	
Ultrasonic Testing Techniques for Pre- and Inservice Inspections of Fast Breeder Reactors Ultraschallprüftechnik zur Null- und Wiederholungsprüfung von Schnellbrutreaktoren	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Bundesanstalt für Materialprüfung Fachgruppe 6.2: Zerstörungsfreie Prüfung

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.1977	31.12.1979	Dr.-Ing. E. Neumann
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	1.323.203,-- DM

1. <u>General Aim</u>

The objective of the project is to carry on development of the ultrasonic testing technique for the nondestructive basic and inservice inspection of austenitic welded components of Fast Breeder Reactors. Besides the difficulties in testing austenitic welds with ultrasound because of the grain scattering arising from the special grain structure in austenitic weld metal there is the difficulty that the inservice inspections are performed at approximately 200°C. The developed ultrasonic testing techniques for coarse grain materials have to be qualified for automatic testing at this temperature.

2. <u>Particular Objectives</u>

- Continuation of ultrasonic probes development for flaw detection and flaw evaluation in coarse grain materials.
- Continuation of ultrasonic probes development for complex shaped welded specimens of the primary loop of Fast Breeder Reactors.
- Development of ultrasonic probes for inservice inspection of Fast Breeder Reactors at a temperature of approximately 200°C.
- Development of signal processing methods to increase the signal to noise ratio at ultrasonic testing of coarse grain materials.

3. <u>Research Program</u>

3.1 Continuation of development of ultrasonic transmitter-receiver probes for flaw detection and of focussing probes for flaw evaluation both in coarse grain materials.

3.2 Development of ultrasonic transmitter-receiver probes for thin-walled (6 to 15 mm) specimens.

3.3 Schlieren-optical representation of ultrasonic fields.

3.4 Elaboration of statistical methods for quantitative evaluation of the performance of ultrasonic testing techniques.

3.5 Investigation of the ultrasonic probes materials and measurements of their ultrasonic and mechanical properties as a function of temperature.

3.6 Calculation, design, and manufacturing of ultrasonic probes applicable at temperatures of approximately 230°C.

3.7 Investigation of the scattering of ultrasound in polycristalline materials.

3.8 Fundamental investigations on deconvolution of ultrasonic signals as a means of signal processing methods to get a better signal to noise ratio.

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

- Computer controlled x-y-manipulator for automatic ultrasonic testing and corresponding computer programs.

- Manual controlled x-y-manipulator for defect size estimation based on shape determination using e.g. focussing probes. The manipulator is provided with a motion pickup and a plotter. Recording of the echodynamic curves in different sections will be possible.

- Schlieren-optical bench and device for optical indication of ultrasonic fields.

- Computer programs for calculation of data for design of ultrasonic probes for complex shaped welded specimens.

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

To 3.1 Specification and standardization of ultrasonic probes for the inspection of the cylindrical and bottom part and of the nozzle welds of the vessel of the Fast Breeder

Reactor SNR 300 have been commenced.

(Tab. 1)

Examples of applications of the transmitter-receiver probes in the workshop are given in Tab. 2.

Defect size estimation is based on comparison with reference reflectors via distance/amplitude curves (reference block system).

The correlation between the echo height and the diameter of flat bottom holes as reference reflectors has been evaluated experimentally for transmitter-receiver-probes. The signal amplitude increases by 12 dB, if the diameter of the reference reflector is doubled, thus confirming the same interrelationship for transmitter-receiver-probes as it is valid for usual ultrasonic single transducer probes.

Furthermore, it has been experimentally confirmed that the calculated equivalent flaw sizes of cylindrical holes are consistent with the size of actual flat bottom holes, the maximal deviation being less than $\pm$ 4 dB.

To 3.2 The development of ultrasonic probes for thin walled specimens resulted in a prototype of a 4 MHz broadband transmitter-receiver ultrasonic probe with an angle of incidence of 70° for plane coupling surfaces. Artificial flaws as a 2 mm diameter cylindrical hole and a notch with 1 mm depth and 1 mm width in the austenitic weld material could be detected with a signal to noise ratio more than 16 dB.

To 3.3 A Schlieren-optical bench has been set up. The focal length of the imaging mirrors is 2m, the maximal object size to be illuminated is 90 mm in diameter. The bench is embedded on non-oscillating bearings. Till now ultrasonic standing wave fields have been optically indicated in a water cuvette.

To 3.4 To estimate the performance of an ultrasonic testing technique quantitatively statistical methods of evaluation of testing results have been used. By a statistical evaluation of the echo amplitudes and of the signal to noise ratios of a certain quantity of flaws in a test weld, the flaw detection probability and the false rejection probability were determined. The flaw detection probability is the cumulative echo height distribution of the flaw indications as a function of the sensitivity setting. The false rejection probability is the cumulative echo height distribution of the grain scattering indications. It is the probability that a noise signal is registered as a flaw signal. Three quality criteria can be taken from this statistical evaluation. The first criterion is the total fraction of detected flaws with a certain signal to noise ratio. The second criterion is the fraction of detected flaws at a certain sensitive setting. The third important quality criterion is the area in the diagram between flaw detection probability and false rejection probability giving a direct measure for the signal to noise ratio.

Another criterion by which the performance of a testing technique can be judged are the maximal sound paths which can be penetrated in the austenitic weld metal. This is important for the testing of transversal flaws and in other cases where long sound paths in the weld metal are occurring.

The maximal sound path is obtained by the crossing point of the DAC curve from a certain reference reflector with the noise limit curve due to grain scattering defined by the level which is exceeded by 1% of the noise indications.

To 3.5 The following components of ultrasonic probes for application at 200°C have been investigated:

- Ultrasonic probe housing: steel or aluminum

- Jacks: commercially available jacks with polyimide insulations
- Wedge: polyimide, ceramics and glass-ceramics
- Ultrasonic probe crystals: lead circonatetitanate, leadmetaniobate, lithiumniobate
- Ultrasonic cable: coaxial cables with fiberglass-mica insulation
- Coils: polyimide enamelled taping wire
- Adhesives: ceramic-, silicone- and epoxide-adhesives
- Mechanical damping of the probe crystals: mixture of heavy metals with diverse cements
- Sealing and insulating compound: silicon caoutchouc

To 3.6 Transmitter-receiver Angle beam probes with glass-ceramic wedge have been calculated, designed and manufactured. They are applicable for testing zones from 20 to 80 mm depth using angles of incidence between 45° and 60°. For the testing zones from the surface up to 20 mm depth transmitter-receiver probes with polyimid wedges having angles of incidence more than 60° are used. The ultrasonic probes were tested out at approximately 230°C. Compared with transmitter-receiver probes for room temperature they have higher cross talk and decreased sensitivity and bandwidth.

To 3.7 Investigations on Rayleigh scattering of ultrasound and multiple scattering in the Rayleigh region have been performed and analysis of the ultrasonic backscattered signal concerning surface wave and volume wave positions has been achieved. The wellkown dependence of backscattered ultrasonic amplitude from grain size has been confirmed for austenitic materials.

To 3.8 For ultrasonic single crystal probes working in the pulse-echo mode computer programs have been set up to calculate the impulse response of a reflector from the amplitude and phase information of the ultrasonic echo. These computer programs can be used for deconvolution of ultrasonic echoes from flaws in austenitic scattering materials, achieving an improved signal to noise ratio due to increasing the bandwidth after receiving them.

6. Results

The results already achieved have been described above.

7. Next Steps

- Manufacturing, testing out and improvement of ultrasonic transmitter-receiver probes for the inservice inspection of Fast Breeder Reactors.
- Continuation of development of transmitter-receiver probes for austenitic thinwalled tubes (6 to 15 mm).
- Application of signal processing methods to increase the signal to noise ratio at ultrasonic testing of coarse grain materials.

8. Relation to Other Projects

This project is linked with the projects RS 250 and RS 254 by a cooperation treaty.

9. References

E. Nabel, E. Mundry
Evaluation of Echoes in Ultrasonic Testing by Deconvolution;
Mat. Eval. 36 (1978) 1, Januar, S. 59 - 61 und 77

E. Neumann, M. Römer, T. Just, K. Matthies,
E. Nabel, E. Mundry

Development and Improvement of Ultrasonic Testing Techniques for Austenitic Nuclear Components;
2nd Int. Conference on Non-Destructive Evaluation in the Nuclear Industry, American Society for Metals (ASM)/American Society for Testing and Materials (ASTM)/ American Society for Non-Destructive Testing (ASNT)/ American Nuclear Society (ASN), Salt Lake City, Utah, USA, 13.-15. Feb. 1978;
Erscheint im Tagungsband

T. Just, M. Römer, E. Neumann, E. Mundry

Vergleich der Leistungsfähigkeit verschiedener Ultraschallprüftechniken für grobkristalline Werkstoffe mittels statistischer Methoden;
I. Europäische Tagung für zerstörungsfreie Materialprüfung, Mainz, 24.-26. April 1978, Deutsche Gesellschaft für zerstörungsfreie Prüfung, Vorträge Band 2, S. 323-330

M. Römer, T. Just, K. Matthies, E. Neumann, B. Kuhlow

Erfahrungen bei der Ultraschallprüfung ebener und gekrümmter austenitischer Bauteile;
I. Europäische Tagung für zerstörungsfreie Materialprüfung, Mainz, 24.-26. April 1978, Deutsche Gesellschaft für zerstörungsfreie Prüfung, Vorträge Band 2, S. 361-370

A. Hecht, E. Mundry, E. Nabel, E. Neumann

Korngrößenbestimmung in austenitischen Feinblechen mit Ultraschall.
I. Europäische Tagung für zerstörungsfreie Materialprüfung, Mainz, 24.-26. April 1978, Deutsche Gesellschaft für zerstörungsfreie Prüfung, Vorträge Band 2, S. 395-402

E. Nabel, E. Mundry

Versuche zur Deutung von Ultraschallechos durch Entfaltung;
Berichtsband der DGZfP zur Europäischen Vortragstagung Zerstörungsfreie Materialprüfung, 24. - 26.4.1978, Mainz, erschienen November 1978, Band 1, S. 123 - 129

H. Wüstenberg, E. Neumann

Improved Probe Techniques at the Ultrasonic Inspection of Coarse Grain Materials.
U.S. Nuclear Regulatory Commission (USNRC), 6th Water Reactor Safety Research Information Meeting (WRSRIM), Nov. 6-9, 1978, Gaithersburg, Maryland, USA,
Bilateral Review Group Meeting between the USNRC and the Bundesminister für Forschung und Technologie (BMFT), Nov. 10, 1978, Bundesanstalt für Materialprüfung, Berlin, Fachgruppe 6.2

E. Mundry, H. Wüstenberg, E. Neumann, E. Nabel

Ultraschallprüfung - Stand der Technik und Entwicklungstendenzen
I. Chinesisches Symposium über zerstörungsfreie Materialprüfung, Shanghai 14.-16.11.1978

T. Just, B. Kuhlow, K. Matthies, E. Neumann, M. Römer

Über den Stand der Entwicklung einer Ultraschallprüftechnik für austenitische Schweißverbindungen
3. Int. Coll. Schweißen in der Kerntechnik, 28.-29.11.1978, Hamburg. Deutscher Verband für Schweißtechnik e.V. (DVS); DVS-Bericht 52, Deutscher Verlag für Schweißtechnik (DVS), GmbH, Düsseldorf 1978, S. 237-242

10. Degree of Availability of the Reports

Bundesanstalt für Materialprüfung, Fachgruppe 6.2 "Zerstörungsfreie Prüfung", Unter den Eichen 87, D-1000 Berlin 45

Tab. 1, part I

SEL-probes with plane coupling surface (AE)

(A: austenitic; E: plane)

Frequency: f = 2 MHz

Distance between sound exit point and probe housing edge: x = 20 ± 5 mm

Type of probe	SEL 70 AE1.7	SEL 65 AE2.20	SEL 60 AE3.30	SEL 45 AE3.30	SEL 45 AE4.40
dimension BxLxH (mm)			40x40x30		
testing zone	1	2	3	3	4
angle of incidence α^o	70	65	60	45	45
depth of maximal sensitivity b_o (mm)	7±2	20±5	30±8	30±8	40±10
projection distance at maximal sensitivity a_o	20±5	40±5	50±10	30±8	40±10
applicable in the depth range	0-12	10-30	20-50	20-50	30-70

Tab. 1, part II

SEL-probes with plane coupling surface (AE)

(A: austenitic; E: plane)

Frequency: f = 2 MHz

Distance between sound exit point and probe housing edge: x = 20 ± 5 mm

Type of probe	SEL 60 AE3.30B	SEL 45 AE4.40B	SEL 45 AE 2.20*)
dimension BxLxH (mm)	60x40x30		40x40x30
testing zone	3	4	2
angle of incidence α^o	60	45	45
depth of maximal sensitivity b_o (mm)	30±8	40±10	20±5
projection distance at maximal sensitivity a_o	50±10	40±10	20±5
applicable in the depth range	20-50	30-70	10-30

*) For testing of the second zone with another angle of incidence

Tab. 1, part III

SEL-probes with curved coupling surface

Type: ARI, ARA; AKI, AKA
(A: austenite; R: tube; K: sphere (double curved),
I: testing from the inner side
A: testing from the outer side)

dimension BxLxH:	35x35x30 mm
frequency:	f = 2 MHz
distance x:	x = 17,5 ± 5 mm

1. No curved coupling surface in the direction of sound beam
Type: ARI, ARA

testing zone	1	2	3	3(4)
angle of incidence α^o	70	65	60	45
depth of maximal sensitivity b_o	7±2	20±5	30±8	30±8
projection distance a_o at maximal sensitivity	20±5	40±5	50±10	30±8
applicable in the depth range	0-12	10-30	20-50	30-70

2. With curved coupling surface in the direction of sound beam
Type: ARI, ARA; AKI, AKA

testing zone	1	2	3	3(4)
angle of incidence α^o *)	60-70	60-65	45-60	< 45 **)
depth of maximal sensitivity b_o	7±2	20±5	30±8	40±10
projection distance a_o at maximal sensitivity **)	20-30	30-55	30-50	30-45
applicable in the depth range	0-12	10-30	20-50	30-70

*) dependent on radius of curvature

**) The angle of sound impact at the opposite surface of specimen should be between 55° and 45°. The angle of incidence α is choosen adequately.

Tab. 2: Examples of Applications of Transmitter-Receiver-Angle-Probes (SEL-Probes) on Austenitic Components

Component	Materials	Weld, Weld method	Geometry of the components	Type of inspection	Calibration of sensitivity, S/N-ratio
SNR 300-Vessel and Test-welds	1.4948 X6 CrNi 1811	Double-V welds, V-welds, Manual electrode welded, Submerged arc welded	plain $40 \leq d \leq 60$ mm	Weld, Longitudinal flaws, transverse flaws	3 mm dia. FBH in weld metal, S/N = 20 dB
Pump housing	1.4948 X6 CrNi 1811	V-welds	cylindrical: $600 \leq \emptyset \leq 1400$ mm spherical: $\emptyset$ 1600 mm, $20 \leq d \leq 35$ mm	Weld, Longitudinal flaws	3 mm dia. FBH in base metal, S/N = 24 dB
Steamgenerator, Flange of tube plate	Ferritic base material, CrNi-cladding (Inconel)	Hand welded cladding	cylindrical: $186 \leq \emptyset \leq 288$mm $6 \leq d \leq 9$ mm	Subcladding cracks	2mm dia. FBH below cladding, transv. flaws:S/N = 16 dB, longit.flaws:S/N=12-14dB
Control rod stud (Compound tube)	Ferrite(St52.4) cladded, Austenit 1.4550	V-weld, Manual electrode welded	cylindrical: $\emptyset$ = 112 mm d = 24 mm	Weld, Longitudinal-flaws	2 mm dia. FBH insonified through weld seam, S/N =16-18 dB
Heat exchanger	1.5662 X8 Ni 6	V-weld, Submerged arc and inert gas welded	cylindrical(plain) $\emptyset$ = 2500 mm d = 28 mm	Weld, Longitudinal-flaws	Testing according to ASME-code: 4 mm dia. cylindrical hole, $S/N \geq 30$ dB
Valve housing	X20 CrMoV 121 and GS-12Cr Mo 910 or 10CrMo910	V-welds, Manual electrode-welded	cylindrical $\emptyset$ = 250 mm d = 38 mm	Weld, Longitudinal-flaws	3 mm dia. cylindrical hole through weld metal, $S/N \geq 14$ dB
Preheater for chemical industry	X10 Ni Cr Al Ti 3220	U- and V-welds, Submerged arc and TIG-welded	cylindrical and spherical $380 \leq \emptyset \leq 1200$ mm $28 \leq d \leq 40$ mm	Weld, Longitudinal flaws, Transverse flaws	3 mm dia. flat bottom hole through weld metal, $S/N \geq 12$ dB

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
o1.o1. - 31.12.1978	2o	RS 28o
Vorhaben/Project Title Development and testing of a gas bubble detection system monitoring the reactor inlet pipes by means of ultrasonic sensors for sodium-cooled fast breeder reactors. Entwicklung und Erprobung eines Glasblasendetektors zur Überwachung der Reaktoreintrittsleitung mit Ultraschallsensoren bei natriumgekühlten Brutreaktoren		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor INTERATOM GmbH
Arbeitsbeginn/Initiated o1.o9.1977	Arbeitsende/Completed 31.12.198o	Leiter des Vorhabens/Project Leader H. Banasch (Koord)
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds

1. General Aim

Development of a bubble detection system in order to monitor critical gas contents in the SNR coolant by detecting bubbles in the reactor inlet lines.

2. Particular Objectives

Performance of a study on the conceptual design of a bubble detection system taking into account: safety criteria, plant layout criteria, bubble theory, selection of the transducer types, elaboration of proposals for the transducer design. Investigation of alternative measuring methods such as inductive or ultrasonic methods. Estimate on the detection limits, on the measuring range, on the reparability, on the development state, on the development keypoints and on the test expenditure. Elaborating constructive design for the water tests.

3. Research Program

3.1 Performing preliminary tests at the NW-6oo pipe section submerged in water, using the ultrasonic (US) method

3.2 Investigating the measuring method by sound through-transmission of a NW-9oo-mm pipc section using several transmitter and receiver heads (measurement from the outside of the pipe)

3.3 Investigating the measuring method by sound through-transmission of a NW-9oo-mm pipe section using (measurement from the inside of the pipe)

3.4 Investigating the measuring method using low-frequency crystals (measurement from the outside of the pipe)

4. <u>Test Facilities, Computer Codes</u>

The preliminary examinations as well as the investigations of the different measuring methods have been performed under water at stationary medium. For that purpose such test equipment has been provided, which is adapted to the geometries of the SNR pipework.
For each of the above-mentioned tests adequate pipe sections have been made. The test rings equipped with measuring sensors have been submerged in a water pool. The necessary gas bubbles distributed according to volume and size are generated by means of a bubble generator provided with 2o nozzles of each, of the following diameters: o.1; o.2; o.4 and o,5 mm. According to the used measuring method respective sensors have been provided, each equipped with ultrasonic sensors and magnetostrictive crystal exciters and acceleration pickups.

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

<u>To 3.1:</u> At a NW-6oo pipe section submerged in water the basic data required for detecting the gas bubbles have been measured. Besides the boundary influences, three evaluation methods have been investigated:

- Influence of the gas bubbles on the ultrasonic attenuation
- Influence of the gas bubbles on the ultrasonic scattering
- Influence of the gas bubbles on the ultrasonic transit time.

To 3.2: When investigation the sound trough-transmission of the NW-9oo-mm test ring, the basic data from item 3.1 have been taken as basis. Due to the enlargement of the piping, some test runs had to be repeated. The use of ultrasonic discs having a diameter of 3o mm has also necessitated a new calibration of the heads.

To 3.3: The investigation of the cone-shaped reflectors resulted in other keypoints for the test programme. The sound parameters are evaluated like in the items above. In the main, the refraction and reflection behaviours have been investigated.

To 3.4: At a NW-1oo pice section filled with water two magnetostrictive sensors have been installed at an axial distance of approx. 5oo mm. One of the sensors has been operated as crystal exciter and the other as receiver. The test equipment has also been used for optimizing the frequency range and for determining the measuring data being most appropriate for detecting the gas bubbles. After having confirmed the basic function, this measuring method has also been investigated at a NW-9oo-mm pipe section.

6. Results

At the performed analyses and investigations it has been proven that a bubble detection is meaningful in the range form 0 to 1o vol%. The critical gas contents thereby are in the range from o.1 to 1o vol%; they are largely detected by means of the ultrasonic methods using the sound trough-transmission of the pipe section with several pairs of measuring heads. At these measuring methods the transmitters are installed outside the piping. The measuring method according to 1 is not applicable for the SNR-requirements.

By means of tests it has been proven that gas contents belon o.1 vol% are prevailing as finely distributed bubbles and will thus only degas over very long time periods. It is thereby not to be excluded that this might lead at some spots to accumulations of bubbles. Under these conditions the measuring method using low-frequency crystals might be rather promising for a bubble detection 2 . At this detection only the integral gas volume is of decisive importance.

At the measuring methods the sensors are also attached outside the piping.

7. Next Steps

- The prototyp electronics shall be proven in the available test facilities and the required compensation circuits shall be developed.

- The low-frequenzy crystal method shall be further developed and proven by basic investigations.

- Appropriate methods shall be developed to suppress multiple reflections and interferences.

- As for the coupling of the sensors from the outside of the piping, respective coupling media shall be investigated.

- As for the testing of the measuring methods under sodium, a NW-2oo test section shall be designed. Besides the measuring methods described in item 6, the fuel-element flowmeter shall additionally be applied as bubble detector. In order to test the handling at the sodium test stand, all sensors shall be designed in such a way that they are replaceable without having to open the system.

8. Relations with Other Projects

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9. References

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1o. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	20	RS 256

Vorhaben/Project Title	Land/Country
Versuche zur Prüfung von Reaktorkomponenten mit Wirbelstromprüfverfahren Investigations for inspection of reactor components by eddy-current methods	FRG
	Fordernde Institution/Sponsor BMFT
	Auftragnehmer/Contractor INTERATOM GmbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.3.77	31.12.78	Banasch (Koord.)
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Completed	December 1978	

1. General Aim

In cooperation with KWU, IFR, IzfP and Interatom investigations are conducted for automatic examinations of reactor components by multi-frequency eddy current testing for production control and inservice inspection. Tests carried out by Interatom have to determine the suitability of eddy current methods in components of LMFBR.

2. Particular Objectives

The first part of the investigations has the subject to
- require the access for inservice inspections,
- give the proof that eddy current testing is a successful method for the examination of sodium contaminated ferritic and austenitic heat exchanger internals.

3. Research Program

3.1 Investigation of accessibility

In this part boundary conditions are made up for inservice inspections by eddy current testing. Problems of accessibility are regarded from the aspect of construction, geometrical character of the heat exchangers and their internals, radiological and temperature conditions as well as the influence of sodium deposits.

3.2 Theoretical and experimental tests of eddy current application

By eddy current tests of SNR-300 heat exchanger tubes statements are made about the suitability of this method. The investigations are particularly concentrated on the influence of austenitic and ferritic materials and of sodium deposits on the tube surface.

4. Experimental Facilities, Computer Codes

To 3.2 In basic tests non destructive examinations are conducted in steam generator- and IHX-tubes, which are provided with well-defined artificial discontinuities and wetted with sodium in a test container.

5. Progress to Date

To 3.1 The investigation of accessibility has been concluded. A summary of the result is given in a status report containing the description of works until Oct. 1978.

To 3.2 Austenitic and ferritic test tubes were provided with artificial flaws. After testing the tubes without sodium influence, they were set into a sodium container and wetted on the outer surface with sodium of 540° C. Final eddy current tests were conducted after draining the sodium.

6. Results

To 3.2 The examinations of the test tubes without as well as with sodium influence were conducted at the IzfP Saarbrücken with single frequency eddy current methods. For ferritic tubes a frequency of 5 kHz was used. With an additional magnetisation of the tubes circumferential notches down to 30 % of wall-thickness and longitudinal notches up from 60 % were detected in tests without sodium.

Test frequency for testing the austenitic tubes was 150 kHz. Circumferential and longitudinal notches from 30 % to 90 % were defined with amplitude and phase.

Drilled holes of 0,3 mm diameter could generally not be detected exactly. After exposing the tubes to sodium the eddy current tests were repeated. The first results of these tests showed, that there is no significant interference of sodium on the flaw-signals, but the resolution limit of flaws was reduced.

7. Next steps

A plan for continuing the investigations to show the possibilities for applications of eddy current methods was prepared.

8. Relation to other Projects

RS 255, RS 257, RS 258, RS 299

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	20	RS 248

Vorhaben/Project Title	
Elementverbiegung im Kernverband Element Bowing in Reactor Cores	Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor INTERATOM GmbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.1.1977	31.12.1979	Banasch (Koord.)
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	December 1978	DM 870.740

1. General Aim

In order to fulfill the safety requirements for the core design of large, sodium cooled fast breeder reactors, a program is needed to calculate the three dimensional subassembly bowing in a reactor core.

2. Particular Objectives

This project has the aim to develop a method step by step. So it will be possible to calculate the three dimensional geometrical, mechanical and reactivity behaviour of subassemblies in a reactor core with the consideration of thermal gradients, differential swelling, irradiation enhanced creep and friction between the subassemblies.

3. Research Program

3.1 Three dimensional force equilibrium in a reactor core
3.2 Consideration of swelling and irradiation enhanced creep
3.3 Testing and description
3.4 Consideration of friction
3.5 Calculation of torsion of subassemblies
3.6 Consideration of friction and torsion
3.7 Other mechanical models
3.8 Verification and qualification of the program

4. Experimental Facilities, Computer Codes

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5. Progress to Date

To 3.1 The modul DDT was completed. Two plotting routines were written to show cross sections through the load planes with contact forces and deflections and to show a longitudinal section through chosen elements with contact forces and deflections.

To 3.2 After the first successful tests of DDT, the programing of the next modul DDTAB was started. This modul calculates the free deflections due to neutron induced swelling and creep. These deflections together with the free deflections of DDT are used to calculate the mechanical equilibrium of the elements in the core.
The modul KEPHIR war programed and is being tested to transfer $\overline{E}Ø_{tot}$ from the modul KASYS to DDTAB.

To 3.3 The modules DDT and DDTAB were tested and the descriptions were started.

To 3.4 The test program was supplemented and can now calculate a 30^{o} sector of a core with an arbitrary numbers of elements with friction forces. Some tests were done.

To 3.5 Some reflections were done about torsion of subassemblies

To 3.8 The comparisons between NUBØW-3D and the testprogram were repeated with the modul DDT.
A meeting was held with people of CEA.

The schedule is fulfilled except for point 3.5 and the consideration of friction 3.4 is done in the testprogram and not yet in the moduls DDT and DDTAB. The foreseen budgets were found to be too small for the programing of the data transfer programs TEMTRA and KEPHIR and the plotting routines.

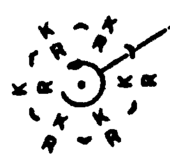
6. Results

To 3.1 The modul DDT is completed and can manage to calculate any configuration of elements using available symmetry properties. The temperature of the subassembly walls are transferred from the thermohydraulic code IACØB by the modul TEMTRA.The results are shown by cross or longitudinal sections and transferred to DDTAB which calculates the burnup situations.

To 3.2 The modul DDTAB receives data from the moduls DDT and KEPHIR. KEPHIR calculates the $\overline{E\emptyset}_{tot}$ or dpa-rates of the subassembly walls using the results of the 3D neutronic modul KASYS. These data are used to calculate the free deflections due to swelling and neutron induced creep.

To 3.3 The tests of the modul DDT are completed successfully and the description has been started. The modul DDT was tested.

To 3.4 An example is chosen, which can be calculated by hand to test the program. The elements are free to move at the upper load plane only and the dimensions are chosen in such a way, that contact occurs at few points only (Fig. 3). Fig. 2 shows a cross section with numbered elements (10,20,30,31,40,41) and the qualitative free bowing due to temperature gradients. In the next figures starting points are marked with • and eventual positions of elements are marked with x; the contact and friction forces are written as shown in figure 1.



positive direction of friction forces

K - contact force

R - friction force

Fig. 1 Legende

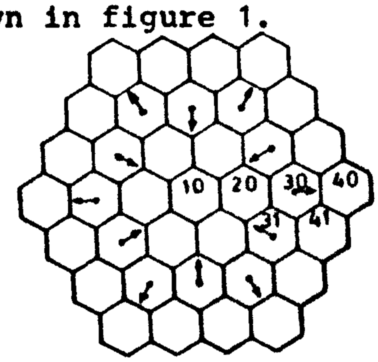
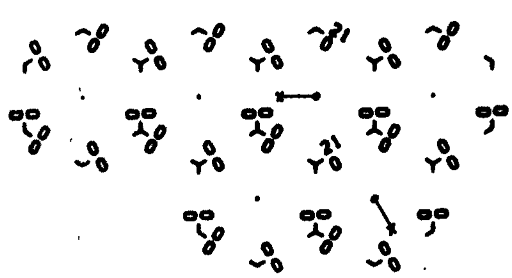


Fig. 2 cross section with direction of bowing due to temperature gradients



scale: ⟼ 2mm

Fig. 3 Start condition

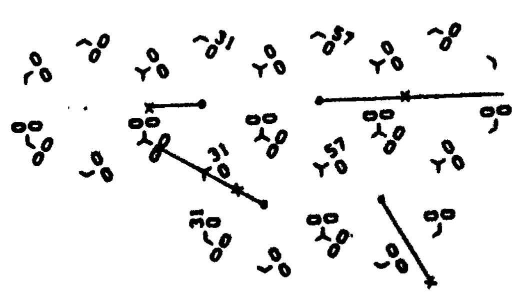


Fig. 4 Test without friction

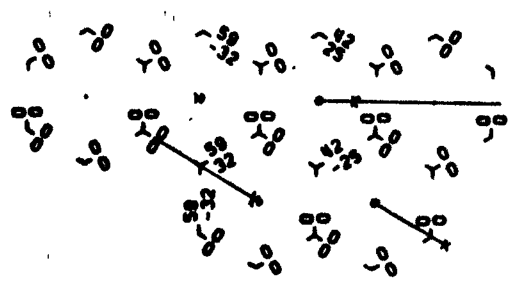


Fig. 5 Test with friction

The equilibrium positions are obtained by adding free bowing and mechanical bowing due to forces. If a friction force is less than the product of contact force and friction coefficient (here μ=.6), the elements stick together (as in Fig. 4 element 20 and 31), otherwise they slip(as in fig.5 element 30 and 41). In fig. 4 the results are shown without friction. The results agree well with hand calculations.

To 3.5 A subassembly has a stiffness of torsion, which is as large as the stiffness of the load plane. Therefore torsion has to be considered only if friction forces are very high.

To 3.8 The three comparison calculations with NUBØW-3D show perfect agreement. The information exchange with France started with success. A comparison is arranged between the two 3D-codes HARMØNIE and DDT.

7. Next Steps

To 3.2 A subroutine will be written to calculate the bulging of elements due to inner pressure and neutron enhanced creep. A routine will be written showing a cross section of the core with deformed and moved elements at chosen axial nodes.

To 3.3 The description and testing of DDT and DDTAB will be completed.

To 3.4 The consideration of friction will be handled in a modul DDTR

To 3.5 After several examples calculated with friction forces, one will decide, if torsion has to be considered or not.

To 3.6 see point 3.5

To 3.7 Considerations of the kind of mechanical ducts moduls. Especially the variation of duct stiffness with loading conditions will be looked into.

To 3.8 Comparisons will be done with HARMØNIE and NUBOW-3D. We shall try to obtain results of experiments done by CEA and of the reactor PHENIX. A principal agreement was already given by CEA representatives.

8. Relation with other Projects

9. References

R. Menssen
3D Mechanics Analysis of LMFBR cores with irradiation effects

and friction
European Nuclear Conference 6.-11.5.1979 Hamburg

10. <u>Degree of Availability of the Reports</u>
The report will be published in the compacts of the ENC'79 of ANS and ENS.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	20	RS 260

Vorhaben/Project Title	
Untersuchungen zur langzeitigen thermischen Gefügestabilität des Stahles 10 CrMoNiNb 9 10 Investigation on long-term stability of steel 10 CrMoNiNb 9 10	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: INTERATOM GmbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.7.77	31.12.80	Mr. Banasch (Coord.)
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General Aim

Determination of parameters influencing the mechanical properties in the long time high temperature range of the ferritic steel 10 CrMoNiNb 9 10 which has been chosen as a structural material for sodium cooled reactor systems.

2. Particular Objectives

- Investigation of the recrystallization behaviour.
- Influence of recrystallization on mechanical properties.
- Evaluation of the results from RS 261 by Mannesmann concerning the importance for the project RS 271 "low cycle fatigue behaviour of steel 10 CrMoNiNb 9 10".

3. Research Program

3.1 Specification, procurement and characterization of test materials for the projects RS 261 and 271.

3.2 Coordination of the projects "thermal stability" and "low cycle fatigue behaviour".

3.3 Evaluation of the results of the program "thermal stability" with regard to the low cycle fatigue strength.

3.4 Evaluation of the results concerning the definition of design values.

4. Experimental Facilities, Computer Codes

Tests are performed by Mannesmann. The present objective includes the activities according to point 3.

5. Progress to Date

To 3.1: The procurement and the characterization of the test materials for the Mannesmann project RS 261 (materials no. 1 - 7) and for the INTERATOM project RS 271 (material no. 6 and an additional melt of steel 10 CrMoNiNb 9 10) have been terminated. As the steels no. 2, 3a and 3b did not reach the necessary strength by the initial heat treatment (1020°C/air cooled + 720°C/air) this procedure had to be repeated using oil quenching after 1020°C annealing. With respect to the two steels for low cycle fatigue tests suitable heat treatment conditions have been defined by the help of an aging programme.

To 3.2: First results from the investigations on thermal stability have been exchanged and compared.

To 3.3 and 3.4: No activities.

6. Results

To 3.1: Presentation of the results see quarterly and annual reports to the projects RS 261 und 271.

To 3.2: A comparison of the hardness measurement data on aged steel 10 CrMo 9 10 (mat. no. 6) showed some inconsistencies the cause of which is not yet known.

To 3.3 and 3.4: No results available.

7. Next Steps

To 3.1: No further work

To 3.2: Continuation of Mannesmann long time exposures with subsequent mechanical short time testing; start of low cycle fatigue tests at INTERATOM; mutual information on the test progress.

To 3.3: Begin of evaluation of long time exposure test results.

To 3.4: No activities

8. Relation with other Projects
This project is closely connected with the projects RS 261 and 271.

9. References
-

10. Degree of Availability of the Reports
-

Berichtszeitraum/Period 1.1.1978 - 31.12.1978	Klassifikation/Classification 20	Kennzeichen/Project Number RS 261
Vorhaben/Project Title Untersuchungen zur langzeitigen thermischen Stabilität des Gefüges des Stahles 8 CrMoNiNb 9 10 Study on the long-term stability of the micro structure of the steel 8 CrMoNiNb 9 10 at elevated temperatures		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor Mannesmann-Forschungs-institut, Duisburg Abt. Metallkunde
Arbeitsbeginn/Initiated 1.4.1977	Arbeitsende/Completed 31.12.1980	Leiter des Vorhabens/Project Leader Dr. H. Fabritius
Stand der Arbeiten/Status continuing	Berichtsdatum/Last Updating 5.1.1979	Bewilligte Mittel/Funds

1. **General Aim**

Determination of the parameters that influence the strength of the steel 8 CrMoNiNb 9 10 during long term exposure at elevated temperatures. This steel is used for components in sodium cooled fast breeding reactors.

2. **Particular Objectives**

The effect of the degree of stabilization (Nb/C) on recovery and recristallization shall be studied as well as the influence of coldworking.

3. **Research Program**

3.1 Materials: see Table 1

3.2 Rolling of the materials:
The casts 1 - 6 are hot rolled to plats 22 mm in thickness

3.3 Heat treatment:
Cast 1: 1020 ^{o}C 30 min/oil and
720 ^{o}C 1 h/air cooling
Cast 2 - 5: 1020 ^{o}C 30 min/air cooling and
720 ^{o}C 1 h/air cooling
Cast 6: 950 ^{o}C 30 min/oil cooling and
720 ^{o}C 1 h/air cooling
Material 7: 700 ^{o}C 5 h/air cooling

3.4 Pre-treatment of the materials:
20 % cold rolling of one half of the materials 1 - 6 (version A). Pre-annealing 20 h at 700 ^{o}C of parts of the materials 5 and 7 (material 5 in both conditions, heat treated and heat treated and cold worked) (version V).

3.5 Cutting of squares for annealing according to 3.6 of the materials according to 3.3 and 3.4.

3.6 Annealing of the squares according to the schedule in table 2.

3.7 Tensile tests are carried out at 20 °C and 550 °C on every specimen, annealed according to 3.6. The initial conditions according to 3.3 and 3.4 are tested with double specimens.

3.8 The impact toughness (Charpy-V-notch, transverse): The A_V-T-curve and the toughness at 550 °C is determined with 18 specimens for every annealing condition. The conditions tested are: initial conditions according to 3.3 and 3.4 and two selected annealing times at every temperature.

3.9 Metallographic examination of the microstructure and hardness-measurement at all annealing conditions.

3.10 Electron microscopy and residual analysis (chemical and X-ray) of selected specimens.

4. <u>Experimental Facilities</u>

to 3.1 Existing laboratory and industrial equipment.
to 3.2 " " " " "
to 3.3 " " " " "
to 3.4 Existing laboratory equipment.
to 3.5 " " "
to 3.6 Partly existing electrically heated furnaces, partly furnaces to be provided.
to 3.7 Existing laboratory equipment.
to 3.8 " " "
to 3.9 " " "
to 3.10 " " "

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

to 3.1 Finished.
to 3.2 "
to 3.3 Finished. The casts 2, 3 and the base material for the weld, No. 7 did not have the specified properties in the air cooled and tempered condition. They were heat treated again therefore by oil-quenching and tempering.
to 3.4 Finished
to 3.5 "

to 3.6 The annealing is finished up to 3000 h.

to 3.7 Tensile test on all materials after annealing up to 1000 h are finished.

to 3.8 Impact tests are finished on all materials after annealing up to 1000 h.

to 3.9 Finished for all annealing times up to 1000 h.

to 3.10 Residual analysis is made for all materials in the initial condition. It must be repeated for the newly heat treated materials.
By reason of the new h... treatment the work is not in time.

6. Results

to 3.1 Finished.

to 3.2 "

to 3.3 "

to 3.4 "

to 3.5 "

to 3.6 Finished up to 3000 h.

to 3.7 The mechanical properties at RT and 550 °C are present for all materials after annealing up to 1000 h.

to 3.8 The A_V-T-curves and the impact-values at 550 °C are present for all materials annealed up to 1000 h.

to 3.9 The hardness-values and the examination of the micro-structure are present.

to 3.10 The chemical composition of the residues of all materials in the initial condition except No. 2 and 3 is present.

7. Next Steps

to 3.1 None

to 3.2 "

to 3.3 "

to 3.4 "

to 3.5 "

to 3.6 The annealing treatment is continued.

to 3.7 Testing of the materials after the annealing treatments.

to 3.8 Testing in the same extend as with 3.7.

to 3.9 Testing in the same extend as with 3.7.

to 3.10 " " " " " " " "

8. Relation with other Projects

Relation with the Project RS 02 605 (Interatom)

9. References

None

10. Degree of Availability of the Reports

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Table 1: characterization of the materials

No.	material denomination according to DIN	remarcs
1	similar 8 CrMoNi 9 10*	0,1 - 0,2 % C, Nb : C = 0
2	8 CrMoNiNb 9 10*	about 0,6 % Nb, Nb : C ≈ 10
3	8 CrMoNiNb 9 10*	0,7 - 0,8 % Nb, Nb : C ≈ 12,5
4	8 CrMoNiNb 9 10*	0,9 - 1,0 % Nb, Nb : C ≈ 16
5	8 CrMoNiNb 9 10**	high degree of stabilization
6	10 CrMo 9 10**	C ≤ 0,10 %, Nb : C = 0
7	similar 8 CrMoNiNb 9 10***	usual analysis

* 1000-kg-laboratory casts

** commercial material

*** weld material deposit

Table 2: annealing schedule to 3.6

ϑ [°C]	annealing time in h									
20	0									
750	1	3	10	30	100	300	1000			
725*	-	3	10	30	100	300	1000			
700	-	3	10	30	100	300	1000	3000	10000	30000	
650	-	-	-	30	100	300	1000	3000	10000	30000	50000
600	-	-	-	-	100	300	1000	3000	10000	30000	50000
550**	-	-	-	-	100	300	1000	3000	10000	30000	50000

* Except with material 5 and 5 A

** The pre-annealed specimens of the materials 5, 5 A and 7 (3.4) are annealed only at 550 °C.

After the first results the annealing schedule was modified for the materials 1 A, 2 A, 3 A, 4 A, 5 A and 6 A. The long annealing times were canceled and replaced by shorter times for a better cover of the range of recristallization.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	20	RS 271
Vorhaben/Project Title Ermüdungsverhalten des Stahles 10 CrMoNiNb 9 10 unter Berücksichtigung von Haltezeiten Low cycle fatigue behaviour of steel 10 CrMoNiNb 9 10 in consideration of hold time		Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: INTERATOM GmbH
Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.7.77	31.12.80	Mr. Banasch (Coord.)
Stand der Arbeiten/Status	Berichtsdatum/Last Updating	Bewilligte Mittel/Funds
Continuing	December 1978	

1. General Aim

Determination of the low cycle fatigue behaviour of steel 10 CrMoNiNb 9 10 and of the influence of hold times for the derivation of design values in the strength analysis for sodium cooled reactor plants.

2. Particular Objectives

- Determination of the low cycle fatigue behaviour at 550°C in different aging conditions.
- Investigation of the hold time effect on the life time in low cycle fatigue test.
- Use and evaluation of damage accumulation rules.
- Determination of the influence of microstructure on the low cycle fatigue behaviour.

3. Research Program

3.1 Pretests to the low cycle fatigue behaviour
3.2 Preconditioning of the specimens
3.3 LCF-tests without hold time
3.4 LCF-tests with hold time
3.5 Creep tests for evaluation of the hold time effect
3.6 Relaxation tests to define the effective stress during hold time
3.7 Use of damage accumulation laws to evaluate the test results.
3.8 Evaluation of structural changes at high temperatures on the low cycle fatigue behaviour
3.9 Definition of design values.

4. Experimental Facilities, Computer Codes

The low cycle fatigue tests are performed with a servo-hydraulic universal testing machine with axial strain measurement and strain-control in the "closed loop system".
For the relaxation tests one testing machine is available.

5. Progress to Date

To 3.1: It was decided to perform low cycle fatigue tests on one melt of steel 10 CrMo 9 10 (material no. 6) and of steel 10 CrMoNiNb 9 10 (round bars, SNR steam generator material). Both materials should be fatigue tested in the as-received condition as well as in two different aging conditions, i.e. heat treatments that cause a partial and complete recrystallization of structure, respectively.
In order to find out suitable heat treatments an experimental programme was carried out including aging of both steels in the temperature range 600 - 800°C with aging times between 0,5 and 1000 hours. The aged specimens were investigated by means of hardness measurements, tensile tests at room temperature and structural examinations.

To 3.2: After evaluation of the test results mentioned above, reference heat treatments for the two steels were defined. Furthermore, the specified aging heat treatments were started on test materials. The parameters for the LCF-tests were defined.

To 3.3 - 3.7: No activities.

Results

To 3.1: From the hardness measurements no concrete statements could be made with respect to the structural changes caused by aging. The results of the tensile tests, however, showed that complete recrystallization occurs after a 1000 hrs heat treatment at ≥ 740°C (10 CrMoNiNb 9 10) and ≥ 760°C (10 CrMo 9 10).

To 3.2: Based on the results of the tensile tests the aging conditions for both steels were defined as follows:

- 100 hrs/760°C (partial recrystallization)
- 1000 hrs/760°C (complete recrystallization)

To 3.3 - 3.7: No results available.

7. Next Steps

To 3.1 - 3.2: Termination of metallographic investigations on aged materials, performance of impact tests on 10 CrMoNiNb 9 10.

To 3.3 - 3.4: Performance of low cycle fatigue testing on test materials in 3 different heat treatment conditions.

To 3.5 - 3.7: Start of work.

8. Relation with Other Projects

This project is closely connected with the projects RS 260 and RS 261.

9. References

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10. Degree of Availability of the Reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1. - 31.12.1978	20	RS 259

Vorhaben/Project Title	
Untersuchung von Na-Meßverfahren auf ihre Anwendbarkeit in NaK-Kreisläufen Test of Sodium Instrumentation and Measurement Methods for Application in NaK-Loops	Land/Country: FRG Fördernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: Technische Universität Berlin, Institut für Kerntechnik

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
1.7.1977	30.6.1977	Prof.Dr.-Ing.U.Wesser
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
continuing	Dec. 31, 1978	

1. General Aim

Measurement methods and liquid metal cleaning techniques will be tested concerning their applicability in NaK-loops. The measurement methods were developed for water and partially employed successfully. Experiences with liquid metal cleaning techniques has already been gained for applications in Na- and K-loops. The measurement and cleaning methods will be tested in conditions in a core-melting-prevention system.

2. Particular Objectives

The tests of measurement methods and liquid metal cleaning are planned especially in NaK-loops. In the starting phase the operaional security of the test-rig is to be tested, parameters are to be measured in stationary states. Nonstationary operating conditions request fast variations of temperature in the range up to 500° C. This situation is expected in a coremelting-prevention system. Determination of attainable temperature-transients is the aim of the planned work. Tests with fluidlevel detecting systems and flow meters may show temperature influence concerning measurement accuracy of these instruments. Information about onset and formation of cavitation may be provided by measuring instruments, mounted outside. Operational security of liquid-metal-loops is dependent on the purity of the liquid metal. For this aim purchasable instruments for determination of oxide concentration and cleaning methods are tested.

3 Research Program

The Research program is devided into six interconnected activities.

3.1 Operational tests NaK-loop

3.2 Operational tests NaK-K-loop

3.3 Investigation of methods for oxide concentration measuring and NaK-cleaning

3.4 Test of flow meters

3.5 Test of fluid level monitoring systems

3.6 Test of detection methods for cavitation

4. Experimental Facilities

The test rig has two closed loops, one of them consisting of a NaK-loop, the other of a K-loop. Both loops may be operated independently. Secondary rigs include systems for cover-gas and cool-air-providing, vacuum stand, cleaning system for argon and filling systems. The test rig is remote-controlled with projected connections to the data processing system.

5. Progress to Date

5.1 Operational tests on NaK loop

The NaK loop was completed and put into operation. A number of trial runs were performed to check the function and handling of components such as EM pump, heating, and emergency cut-out. In order to carry out the experiments, the following were installed: a level indicator, a test section to stimulate cavitation, and a test cell for speed measurements by means of temperature noise-analysis.

5.2 -

5.3 -

5.4 Test of flow meters

The test cell was put into operation and tested under various operating conditions.

5.5 Test of level probes

There are two independent systems of level measurement. Electrodes are used for discontinous and an ultrasonic probe for continous measurement. The investigation proceeded at a constant temperature but with various feed rates.

5.6 Investigation of cavitation detection methods

Cavitation was stimulated at a sharp edged diaphragm at various mass flow rates. Suitable ultrasonic recording devices were made available by project RS 284 (cavitation signals from emergency-cooling pumps).

6 Results

with ref. to 3.4 Test of flow meters

The first experiments have shown that an evaluation of the temperature-noise-signals is possible. Geometry, power and thermocouples must be optimised.

with ref. to 3.5 Test of level probes

Tests with the first ultrasonic probe showed that the prozedure is applicable for NaK as well as for water. Problems with the passage of the sonic conductor through the container wall without signal attenuation or a acoustic short circuit were solved. Experiments carried out so far at constant temperature with various feed rates have shown satisfactory results.

with ref. to 3.6 Investigation of cavitation detection methods

Cavitation signals recorded at the diaphragm were evaluated by methods developed in project RS 284. A comparison with analysis of cavitation signals in water shows a great similarity in the curve pattern.

7. Next Steps

The priority in the case of speed measurement is the optimisation of the measuring arrangement. In the case of level measurement, the behaviour of the probes under very strong temperature fluctuations is to be exammined. The cavitation investigations will be continued with test stretches of different geometries (pressure change patterns in the fluid).

8. Relations with other projects

1. Project RS 284
   "Cavitation signals from emergency-cooling pumps"
2. Project RS 225
   "Density measurements by means of ultrasonic probes"

9. References

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10. Degree of availability of the reports

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Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
01.01.78-31.12.78	20	RS 305

Vorhaben/Project Title	
Untersuchung der Wechselwirkung zwischen Druckwellen und Bauteilen in flüssigkeitsgefüllten Systemen. Investigations on the interactions of pressure waves and components in liquid filled systems.	Land/Country: FRG Fordernde Institution/Sponsor: BMFT Auftragnehmer/Contractor: INTERATOM GmbH

Arbeitsbeginn/Initiated	Arbeitsende/Completed	Leiter des Vorhabens/Project Leader
01.11.77	28.02.81	Dr. Walter (Koord.)
Stand der Arbeiten/Status	**Berichtsdatum/Last Updating**	**Bewilligte Mittel/Funds**
Continuing	December 1978	

1. General aim

LMFBR safety analysis shows a number of plant failures by which strong overpressure transients in the system occur. In the design of components the hydraulic loads due to the resulting pressure waves have to be taken into considerations.

As an "improved status of the art" the description of pressure-time histories inside of an apparatus in which the accident takes place is already possible and proofed. But there is still a great lack of knowledge in the mathematical-physical description of the transmission of pressure waves through a system. This knowledge is necessary in order to describe the behaviour of a component which is exposed to a pressure wave transmitted from another one.

2. Particular objectives

The particular aim is the development of experimentally proofen computer codes which allow a satisfactorily description of the interactions between pressure waves and components. Special attention will be payed to the elastic-plastic behaviour of components during highly hypothetical events.

3. Research program

At INTERATOM two different computer codes for the description of pressure waves have been developped:
HEINKO/C and ROPLAST/2.

The ranges of validity of these codes have to be verified and qualified as well by experiments.

4. Experimental facilities, computer codes

The experimental set up consists of a pipe system of 100 mm nominal diameter. Pressure pulses of different amplitudes and rise times are generated by gas explosions. Transients up to 35 bar/ms and maximum amplitudes up to 60 bar are obtained. The system operates at room temperature, with water as the pressure transmitting liquid. The resulting load functions are picked up by quartz pressure transducers and strain gages. The experimental results are compared with the results of calculations performed with the computer codes.

Two different computer codes are available at INTERATOM:

- HEINKO/C, a one dimensional program for the description of pressure waves. The feed back from the tube wall material to the pressure is approximated by a change of the velocity of sound in the liquid.
- ROPLAST/2, in addition, describes the elastic-plastic behaviour of the tube wall and the interaction between tube wall material and pressure wave. Similar to HEINKO, the hydrodynamics are one dimensional.

5. Progress to date

In detail the following activities have been performed:

- Final design and fabrication of a spherical pressure wave generator.
- Construction of a platform for supporting the pressure wave generator and the test components.
- Numerous test measurements for generating reproducible pressure/time functions of well-defined height and rise times.
- Mounting of the test arrangements for studying the elastic interactions between pressure waves and straight tubes, elbows and T-junctions.
- First experiments and detailed analysis of the experimental data with respect to the effect of tube vibrations on shape and possible deformation of the pressure waves in the system.

6. Results

- A wide range of different pressure/time funktions for generating pressure waves in the water-filled system can be produced by gas explosions within a shperical vessel of 0.3 m diameter. Shape and height of the pressure pulses are well reproducible. Transients up to 35 Kbar/s and amplitudes up to 60 bar can be chosen.
- The effect of large accellerations of the test components (e.g. vibration) on the pressure waves under study has been understood and can almost completely be eliminated by experimental measures.

7. Next steps

- Experiments on stage "a": Elastic interaction between pressure waves and single components such as tubes, elbows, T-junctions, changes of flow area etc.. Comparison with calculated pressure curves and verification or revision of the computer-codes.	up to 30.06.79
- Construction and fabrication of the test components for experimental stage "b": several elbows in series with tubes and vessels.	up to 30.06.79
- Experiments and calculations on stage "b".	01.07.-31.12.79
- Construction and fabrication of thin-walled tubes for studying the elastic-plastic interaction of pressure waves and tube walls (stage "c" of the experiments).	up to 31.12.79

8. Relations to other projects

none.

9. References

none.

10. Degree of avialability of reports

none.

Berichtszeitraum/Period	Klassifikation/Classification	Kennzeichen/Project Number
1.1.78 - 31.12.78	2o	RS 3o1
Vorhaben/Project Title Kreislaufsimulationsprogramm HYDRON Computer Code for Circuit Simulation		Land/Country FRG Fördernde Institution/Sponsor BMFT Auftragnehmer/Contractor INTERATOM GmbH
Arbeitsbeginn/Initiated 1.1.78	Arbeitsende/Completed 31.12.8o	Leiter des Vorhabens/Project Leader Dr. Walter (Koord.)
Stand der Arbeiten/Status Continuing	Berichtsdatum/Last Updating December 1978	Bewilligte Mittel/Funds DM 661.937

1. General Aim

The development of a computer code which simulates nonstationary fluid mechanics in a network with arbitrary junctions.

2. Particular Objectives

The description of nonstationary, incompressible fluid flow in a piping network, where the topological data may be given arbitrarily by input data.

The calculation of the propagation of large gas filled areas in a fluid filled piping system, as for draining and filling-in processes.

The computation of bubble transport, slug flow and gas entrainment in low pessure regions.

3. Research Program

3.1 The theoretical description of networks with arbitrary given junctions. The development of a general formalism to include special physical models (incompressible fluid flow, gas flow, perhaps two-phase flow).

3.2 The formulation of the program organisation and the programming of routines, which describe the topology of the network and the first physical model (incompressible fluid flow).

3.3 The theoretical description of the movement of the interfaces of fluid and gas and the time-variability of the network caused by draining and filling-in processes.

3.4 The programming and testing to point 3.3

3.5 The theoretical formulation and programming of bubble transport and accumulation of gas at special points in the network.

3.6 Improvement of the program, optimisation of the running time and memory occupation.

4. Computer Codes

The integration of the time dependant equations of motion will be done by a INTERATOM-code called IADIS, which makes use of the different Runge-Kutta-methods. The different physical models shall be written in single moduls to get an optimal storage occupation.
The equations of the stationary states will be solved by the Newton-Raphson-method as is done in the CDC-routine NONLIN.

5. Progress to Date

to 3.1 A study of the literatur showed that no comparable program formulation would be accessible. An estimate of the expected computation time schowed that there should be an appreciable advantage to compressible fluid flow caculations (as e.g. in the INTERATOM-code HEINKO, which describes interfaces of gas and fluid in a relatively great variability). A first version of the mathematical (graph theoretical) basic for the code HYDRON was formulated. As far as possible there were made use of the theory of electrical networks. In a further step the physical equations of the incompressible fluid flow were written in a systematic manner and the graph theoretical formulation of the solution procedure were performed.

to 3.2 A sketch of the program organisation was given and the first routines to calculate the network topology were programed and successfully tested.

6. <u>Results</u>

The formulation of the physical model showed that it is possible to write down the equations of motion and the pressure loss relationships for a single section of the piping system in a compact form which is independant of the way choosen through the network. That gives a basic for the construction of the system of equations of motion by the program itself, using the topological input data only. Thereby the program will be able to describe any network with arbitrary given junctions and other constructive elements.

An extension of the definition of the incidence matrix allows a formulation of the essential parts of the solution procedure (e.g. the computation of the incidence matrix, optimal ordering in the sparse matrices, the construction of the matrix of the basic loops) in a form which probably will lead to a fast running program.

to 3.2 see point 5.

7. <u>Next Stepps</u>

to 3.1 It should be tried to get a similar generalizebal formulation of the equations of motion for the gas flow. The theory of the solution procedures should be brought to a point where the first routines may be written.

to 3.2 Final tests of the routines to describe the network topology will be done. A comparison of the solution procedures given in the literature and dealing with sparse matrices will be performed.

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

There exist some hydrodynamic programs with special models of networks (e.g. KRELEC,DRUWA, HEINKO) which can be used for a comparison of the calculated results. That will be possible for a restricted motion of the interfaces of gas and fluid too.

9. References

IA-Notiz Nr. 7o.837.7 (Aug. 1o/78), HYDRON, 1. Teilbericht
IA-Notiz Nr. 7o.852.5 (Aug. 29/78), HYDRON, 2. Teilbericht
IA-Notiz Nr. 7o.881.4 (Oct. 21/78), HYDRON, 3. Teilbericht

1o. Degree of Availability of the Reports

The reports are internal notes of INTERATOM and not generally available.
For inquiry: Dr. H.-J. Walter, OE o333, INTERATOM, Friedrich-Ebert-Str., 5o6o Bergisch Gladbach 1